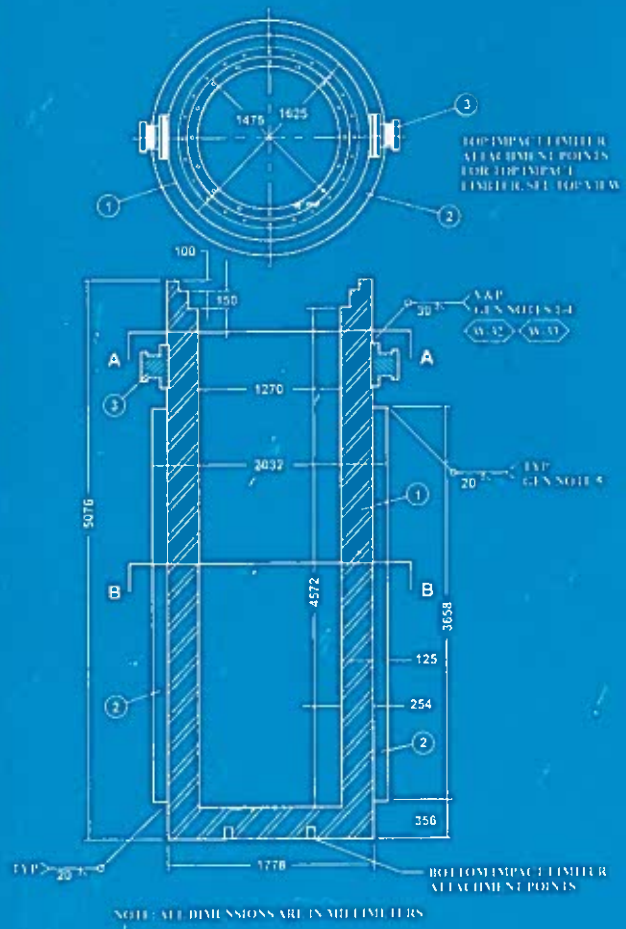
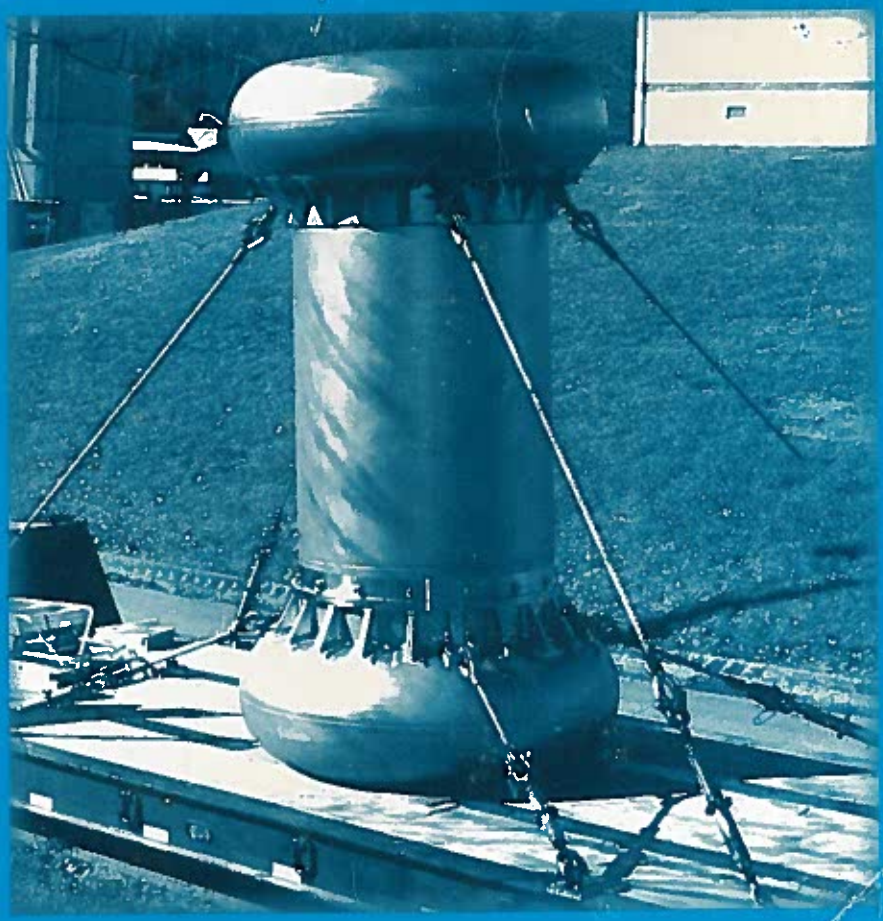
ORNL/M-5003

THE RADIOACTIVE MATERIALS PACKAGING HANDBOOK

Design, Operations, and Maintenance





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THE RADIOACTIVE MATERIALS PACKAGING HANDBOOK

Design, Operations, and Maintenance

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1. Introduction

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As part of its required activities in 1994, the U.S. Department of Energy (DOE) made over 500,000 shipments. Of these shipments, approximately 4% were hazardous, and of these, slightly over 1% (over 6,400 shipments) were radioactive.[1] Because of DOE's cleanup activities, the total quantities and percentages of radioactive material (RAM) that must be moved from one site to another is expected to increase in the coming years, and these materials are likely to be different than those shipped in the past. Irradiated fuel will certainly be part of the mix as will RAM samples and waste. However, in many cases these materials will be of different shape and size and require a transport packaging having different shielding, thermal, and criticality avoidance characteristics than are currently available. This Handbook provides guidance on the design, testing, certification, and operation of packages for these materials.

1.1 PACKAGE TYPES

Only a limited amount of information is provided in this document on those packages for which certification by an appropriate authority is not required (e.g., a Type A package). The Type A package is an example of a "self-certifying" type and is restricted to carrying less than Type A quantities of radioactive material as identified in the U.S. Nuclear Regulatory Commission (NRC) regulations 10 CFR Part 71, Appendix A.* or the Department of Transportation regulations, 49 CFR 173.435.* These packages are often small, often require little shielding, and are therefore given a general license that permits them to be shipped in general commerce. The bulk of this document discusses those package designs that require certification by appropriate authorities, such as a Type B, or a Type AF package. A Type AF package is designed to carry significant quantities of fissile materials, and a Type B package is capable of transporting greater than Type A quantities. Both these package designs must be certified by the appropriate authorities before they can be used.

However, because the design of this latter package often involves complex evaluations dealing with shielding, heat transfer, criticality, structural integrity, and containment, the emphasis of this document is on Type B packaging.

Certification of Type B packages (see Chapter 3) is based on information provided by the designer in a *Safety Analysis Report for the Packaging* (SARP). The time and cost of obtaining a certificate

*The NRC and DOT regulations, as issued effective April 1, 1996, are referenced as appropriate in this document.

for a package are dependent upon the throughness and accuracy of the submitted SARP, and the complexity of the package design. Approval may be complicated because many new tools are available for analyzing the behavior of a particular package under normal and accident conditions, and these tools must be shown to provide accurate information for the SARP.

In 1970, *A Guide for the Design, Fabrication, and Operation of Shipping Casks for Nuclear Applications*, commonly called the *Cask Designers Guide*, was published with the intent to provide, in one place, information that would be useful to support most of the engineering activities required to design a package to transport a variety of RAM sources. At that time, only a few computer-based programs were available to analyze specific aspects of a package design, but many technical studies applicable to packages were then being carried out. Since that period, many improvements in package design have appeared, designers have improved their calculational techniques, and much effort has gone into applying quality assurance (QA) principles to cask development. Many component and packaging tests have been performed. Materials, and their limitations, have surfaced as a very important consideration in the certification process, and better data are now available.

This document has one primary goal: to provide sufficient information and guidance to improve the quality of SARPs that are submitted to the DOE or the NRC in support of the package certification process. The methodology chosen here to fulfill that goal is to utilize groups of experts that have (1) contributed to the development of one or more SARPs which have been submitted to either the DOE or NRC for certification or (2) recently interacted with the regulatory bodies. It was judged that their experience would provide the best guidance to minimize the questions raised in the certification process and maximize the likelihood of successfully obtaining a certificate. An additional benefit of using this methodology is that the reader will also be provided generic information on the design of Type B packages and other package designs requiring authority approval, as well as information on package fabrication, QA, certification, use, maintenance, SARP preparation, and other general topics, most of which will directly influence the package design.

As with many documents of this type whose authors are a group of experts in a number of very specific fields and who come from a wide variety of institutions, the reader will find that each chapter emphasizes the areas that the author(s) consider most important on their topic and how it should be applied to the design of a RAM packaging. As a result, some chapters provide more of a theoretical basis for the topic than do others. However, in both cases, the authors have attempted to provide the best information available on their respective topics, and that should help both the novice or veteran package designer in his or her design efforts and thus minimize the problems that may be encountered in the certification process.

Although the focus of this document is on the design of packages, a number of chapters deal with their operational aspects because many of the operational concerns that have been raised in the past can be traced directly to poor package designs. These concerns can also reflect problems created during fabrication. People who design packages often are not involved in their fabrication or operation and consequently have difficulty in recognizing potential operational problems. However, those who do fabricate or operate packagings, once they encounter difficulties, are always looking for the home phone number of the designer (generally around midnight) to inquire just how he or she expected the packaging to be built or to operate properly under conditions the operator considers quite normal. This document should provide the designer with some insight to many of the typical fabrication and operational concerns from those who do utilize radioactive

material packagings and, as a result, will provide a framework for improving the design before it is completed.

The U.S. regulations applied to RAM transport were first developed in 1947 by the Interstate Commerce Commission (ICC) and promulgated in 1948 as Title 49, *Code of Federal Regulations*, Parts 107–180. In these, their subsequent revisions, the introduction of 10 CFR Part 71 by the NRC, and in the International Atomic Energy Agency (IAEA) regulations,[2] the primary cornerstone of the regulatory philosophy has been to require that the package provide maximum protection to the public with little, or no, reliance on operational controls or human intervention.

During the past two decades, the U.S. regulations have undergone two major revisions, and packaging designs have gone through an evolutionary period, particularly in areas where those employed for spent fuel shipments are concerned. In the 1960s, reprocessing was to be the way spent nuclear fuel (SNF) was handled, and large packages were designed to ship short-cooled fuel. Now, because SNF has remained as long as it has in storage at the various reactor sites, the activity, radiation, and heat load associated with the assemblies have decreased significantly. With the decrease in activity and radiation levels brought on by the longer cooling times, the packagings require less gamma shielding and, therefore, weigh less for a given cavity size. This trend also implies having a larger cavity for a given package weight, which, in turn, results in an increased payload (and higher shipping efficiency) for the packaging. In addition, essentially all SNF packages are now equipped with impact limiters to reduce the shock loading on the package (should it be involved in an accident), thereby providing an increased level of protection to the package and the public. All these factors have enhanced the safety of spent fuel shipments in recent years.

QA has also become a key factor in the package design and in the procurement, fabrication, acceptance testing, operation, maintenance, modification, and repair of RAM packagings. Applicable QA requirements can be found in 49 CFR 173, 10 CFR Part 71, and DOE Orders 460.1A, 460.2, and 5700.6C. If the package design is to be certified by the NRC, the governing regulatory document is 10 CFR Part 71, Subpart H and 10 CFR 830.120; however, the designer's QA plan must conform to any additional requirements of other applicable documents, such as the NRC *Regulatory Guide 7.10*, American Society of Mechanical Engineers ASME-NQA-1, NUREG/CR-5717 and NUREG/CR-3019. Subpart H of 10 CFR Part 71 recommends the use of a graded approach in establishing the level of quality effort (QA requirements) applied to important-to-safety items and activities. In addition, the level of quality effort must be consistent with the category level established by the design criteria which is, in turn, based on the nature of the radioactive contents being transported.

Because of the complexity of most Type B packages, their design is frequently carried out by a group of engineers, many of whom are specialists in very specific technical areas (e.g., heat transfer, shielding, and criticality). However this document addresses them collectively as *the package designer*. Thus, the package designer must be aware of many interrelated technical fields, federal regulations, and operational constraints and limitations of the packaging under consideration. Figure 1.1 summarizes these interrelationships and how they must be established within, and supported by, both the regulatory and QA framework. These relationships were discussed extensively in ref. 3.

Throughout most of this document, the word *packaging* is used to refer to the empty container that is used to ship radioactive materials, and *package* is used to refer to the packaging plus its contents.

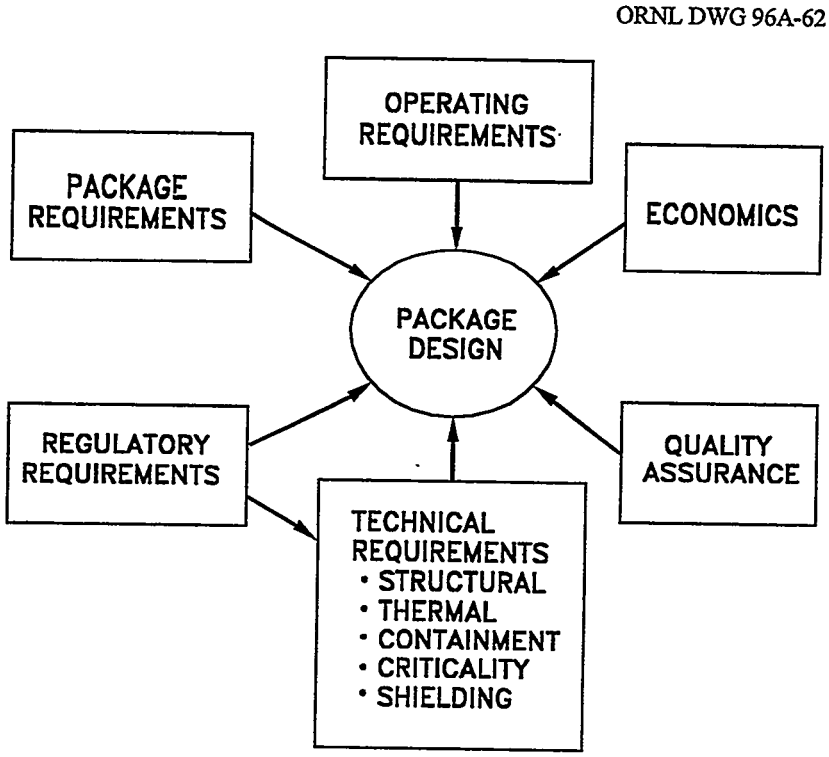

Fig. 1.1. Relationships among package design, regulations, and operating requirements.

However, the designer always considers the contents when carrying out his or her activities, and thus it is the package that is designed. In a number of chapters of this *Handbook*, however, the word *cask* is used interchangeably with *packaging* and generally refers to a large, very heavily shielded packaging typically used to transport irradiated fuel from commercial reactors.

A concerted effort has been made, when at all possible, to present the technical information in this document in the SI system of units. However, this has not always been possible, particularly in Chapter 8, which deals with containment. The bulk of the information in that chapter is based on the American National Standards Institute ANSI N14.5-1987 Standard,[4] which is written using English units. In addition, much of the technical data presented in Appendix A which focuses on materials used in the construction of impact limiters, were generated over the past 20 years. Most of the information that was referenced therein was originally displayed in English units. In cases like this, and, where possible, both the original English units are used along with SI units which have been added.

As mentioned earlier, the *Cask Designers Guide* was published during a period of development of theoretical tools, many of which are now being used to investigate in detail various aspects of package design. Since then, not only have the theories on material behavior and radiation interaction with materials continued to evolve, but also more powerful computers are now available that make detailed studies of heat transfer, structural integrity, and criticality quick and easy. However, even though the design of a package will typically involve extensive use of computer analyses to evaluate the most important aspects of the package behavior under normal and accident conditions, many of these calculations are often not carried out until the design is fairly far along and the designer has many details of the packaging concept. This means that the

designer often employs rules-of-thumb and simplified hand calculations to create the initial package design concepts. For this reason, some of the chapters provide simple methodologies to analyze various aspects of a package design and the detailed analyses methods and computer codes that have been found to produce more accurate and defensible results are simply referenced.

1.2 NEW REGULATIONS AND ORDERS

The last half of the 1990's will be a transition period for U.S. regulatory agencies; since approximately the start of fiscal year 1996, NRC and DOT have modified their transportation regulations, bringing them more in line with the International Atomic Energy Agency (IAEA) regulations, and the DOE has changed their orders. This *Handbook* has attempted to update the references in all chapters to reflect the new numbers, titles, and dates. However, in the case of DOE Orders, implementation of those orders by DOE contractors will take place over a period of time. Specifically, contractor compliance will be required only when it is specifically agreed to in the contract between DOE and that particular contractor. Thus, contractors have been directed to comply with the requirements of Orders that have been canceled by the new series Orders until such time that their contracts are modified to delete the reference to the requirements of the canceled Orders.

In late 1997, the NRC issued a draft report, NUREG 1609, entitled *Standard Review Plan for Transportation Packages for Radioactive Material* for review and comment. This document is available on the NRC website at http://www.gov/NRC/NUREGS/SR1609/SR1609.html. According to the abstract, the objectives of the document are to: (1) summarize the requirements noted in 10 CFR 71, (2) describe the procedures under which the NRC staff determines that these requirements have been met, and (3) identify specific areas of interest that have been investigated by the NRC in previous certification applications for a variety of package types (e.g., unirradiated fuel packages, Type B waste packages, high-enriched uranium or plutonium packages, etc.).

In addition, a separate, but similar, document entitled *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, NUREG 1617, is expected to be issued for review and comment by the NRC in the Spring of 1998.

Recently, the need for certifying the designs of nonfissile and fissile-excepted uranium hexafloride (UF_6) packages was considered on a worldwide basis under the auspices of the IAEA. Internationally, such certification must be obtained by either December 31, 2000, or by December 31, 2003, depending upon the individual package design. Although these requirements are expected to ultimately be included in U.S. regulations, this document does not address or discuss the concerns associated with those package designs.

1.3 REFERENCES

1. *Shipment Mobility/Accountability Collection Summary Report: Fiscal Year 1994*, Transportation Management Division, Office of Environmental Management, U.S. Department of Energy, Washington, D.C., March 30, 1994.

2. International Atomic Energy Agency, *Regulations for the Safe Transport of Radioactive Materials*, Safety Series No. 6, Vienna, 1961.

3. M. S. T. Price, "National and International Transport of Radioactive Materials," Conference on Nuclear Transport Systems, Organized by IBC Technical Services Limited, Industrial Division, London, June 1993.

4. *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment*, ANSI N 14.5, American National Standards Institute, Inc., New York, 1987.

2. Regulations and Standards

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2.1 INTRODUCTION

The design, certification, fabrication, maintenance, and use of Type B radioactive and fissile materials packages are some of the most closely regulated of all of the hazardous materials transportation activities. Although radioactive materials are only one of many hazardous materials classes, they have evolved as the most rigorously controlled. For example, the combined performance requirements for packages containing greater than Type A quantities of radioactive materials are more stringent than for any other hazardous material.

Unlike other hazard classes, class 7 radioactive materials have been regulated on a package-design performance basis for over 25 years. Only recently have other hazard classes been brought under this approach, and the performance requirements for these materials do not approach the standards required for Type B radioactive and fissile materials. A comprehensive approach has been taken to define and adequately control all potential hazards presented by radioactive materials. This approach has proven very successful and has provided radioactive materials transport with an enviable safety record.

The overall regulatory philosophy applied to radioactive materials transport is to require that the packaging provide the primary protection with a minimal reliance on operational controls or human intervention. A graded approach has been applied to the required level of performance of radioactive materials packages which is commensurate with the potential hazard presented by the contents. All radioactive materials packages must meet regulations that require appropriate measures to

1. contain the radioactive contents (containment),
2. limit radiation emanating from the contents (shielding),
3. prevent nuclear criticality (criticality safety), and
4. manage any decay heat generated by the contents.

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The International Atomic Energy Agency (IAEA) has been instrumental in ensuring that appropriate regulations are developed and promulgated for international use. These regulations are recommended to IAEA Member States (i.e., countries) for adoption and provide the basis for the regulations imposed by all major countries. Considerable harmony exists among the regulations of the IAEA, most countries, and all international transport organizations, such as the International Civil Aviation Organization (ICAO) and the International Maritime Organization (IMO). In the United States, transportation and packaging requirements are specified in the U.S. Department of Transportation (DOT) hazardous materials regulations (Title 49 *Code of Federal Regulations* (CFR) Parts 100–180) and the U.S. Nuclear Regulatory Commission (NRC) regulations (10 CFR Part 71). The U.S. Department of Energy (DOE) imposes additional requirements through a series of internal directives applicable to its operations.

All life-cycle phases of Type B radioactive and fissile materials packagings are tightly controlled by the regulations. Specific approval (or certification) is required for the package designs and many aspects of packaging fabrication and use. A very high degree of quality assurance (QA) is mandated by the regulations, including requiring formal QA programs and appropriate quality control (QC) measures for the use of the packagings.

An overall understanding of the organization and applicability of the transportation and packaging regulations will be valuable to the package designer, fabricator, and user. This chapter provides an overview of the regulations and standards,[†] and describes the interactions among the regulatory bodies. Where a more in-depth understanding of specific regulations is required, the reader is referred to the particular chapter in this *Handbook* covering the topic of interest.

2.2 REGULATIONS

The primary regulations governing the design, manufacture, and use of certified radioactive materials packagings in the United States are those of DOT and NRC. DOE also exercises jurisdiction over certified packaging activities that it or its contractors perform. The domestic regulations are heavily influenced by the international regulations developed and promulgated by the IAEA. As an active participant in the IAEA regulatory development activities, the U.S. regulatory agencies work to ensure that the international and domestic requirements remain as consistent as possible. The nature and scope of all of these regulations are described in this section, and a description of their relationships with one another is presented.

Even though this section provides an overview of the applicable regulations, space limitations allow only the highlighting of the major requirements. Because many requirements are not described in this chapter, package designers and users of certified packages must consult the regulations in full to ensure compliance with all aspects.

[†]The overview provided utilizes the international regulations issued by the IAEA and companion international and domestic bodies based upon the 1985 Edition of IAEA Safety Series No. 6 (as amended 1990) and the standards for radiation protection upon which they are based. These requirements are generally in force worldwide at the time of publication of this guide, and are those which will remain in force until after the turn of the century. The IAEA issued new regulations in 1996, and a brief summary of the changes that will result from this action is provided in Sect. 2.2.5 of this chapter.

2.2.1 International Regulations

The regulations that govern the design, testing, certification, and use of Type B and fissile radioactive materials packages have their source in the international regulatory structure. Beginning in the 1950s, steps were taken to develop an international consensus on regulations for the safe transport of radioactive materials; the newly formed IAEA had the responsibility for accomplishing this task. Initially, the synthesis of these regulatory requirements was based on the then-existing set of requirements embodied in the U.S. Bureau of Explosives regulations, as well as other requirements that existed in a few IAEA member states. The first edition of these "regulations" was issued in 1961: *Regulations for the Safe Transport of Radioactive Materials*, Safety Series No. 6 (ref. 1).

Safety Series No. 6 has been periodically reviewed, updated, and reissued as technology has developed, information on transport safety has expanded, and the needs of individual countries and international organizations have changed. The latest changes to Safety Series No. 6 occurred in 1985, when the document was completely revised and reissued. It has since undergone revisions to implement identified editing and minor technical corrections, and the current version is the "As Amended" 1990 edition.[2]

International experts continue to meet under the auspices of the IAEA to address issues related to the continued review and revision of Safety Series No. 6.

The requirements set forth in Safety Series No. 6 are focused upon ensuring safe transport of radioactive materials. Special emphasis has been placed on those features of packaging and operational controls that will provide radiation protection for workers, the public, and the environment. To this end, the recommendations of the International Commission on Radiation Protection (ICRP) have continually been considered and have been adopted into Safety Series No. 6. The IAEA *Basic Safety Standards for Radiation Protection*, Safety Series No. 9 (ref. 3), has provided the reference base for implementing these radiation protection guidelines in the transport regulations.[‡]

One primary goal of those who have worked over the past decades to develop the international regulatory requirements in Safety Series No. 6 has been to present requirements in as simple a fashion as possible while not constraining the designer's flexibility or the regulator's ability to adequately assess a design. This approach has been taken (1) to avoid confusion and at the same time (2) to foster a broad and clear understanding of the requirements and compliance therewith. To accomplish these objectives, the regulatory requirements specified in Safety Series No. 6 are generally proscriptive (specifying what) rather than prescriptive (specifying how). For example, radiation-level limits outside a package are established rather than specifying that, for a given package contents, certain types and thicknesses of shielding shall be provided. This prescriptive approach provides a wide range of flexibility to a designer while imposing the burden of proof on the designer and the designer's documentation concerning satisfaction of the prescriptive requirements.

[‡]New radiation protection recommendations from the ICRP were incorporated into a new IAEA document, *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, Safety Series 115-1, in 1994. These recommendations, in turn, have been considered and included in the new addition of Safety Series No. 6 which was issued in late 1996. Adoption of these new requirements into domestic regulations can be expected around the turn of the century, and additional details are provided in Sect. 2.2.5 of this chapter.

In addition to Safety Series No. 6, the IAEA issues a series of supporting documents, including a document on *why* certain regulatory requirements have been imposed, Safety Series No. 7 (ref. 4), and a document on *how* the regulatory requirements might be satisfied, Safety Series No. 37 (ref. 5).

Finally, other international bodies issue regulatory documents, and these bodies impose requirements (either directly or indirectly) upon the domestic or international design, testing, certification, and use of Type B and fissile radioactive material packages. These documents include regulatory documents from the United Nations (UN) Committee of Experts on the Transport of Dangerous Goods, the IMO, the ICAO, and the International Air Transport Association (IATA). The IAEA and all of the preceding listed organizations, with the exception of the IATA, are UN organizations.

These regulatory authorities have developed their relationships into a fairly well-defined set of interactions; these include not only the international, regional, and domestic regulatory bodies, but also the international standards organizations (the latter is discussed in Sect. 2.3.2). The relationships among the various regulatory authorities, and the international standards organizations, are depicted in Fig. 2.1, which illustrates that the IAEA plays the central role in setting international regulatory requirements for the transport of radioactive materials. This figure shows that the IAEA interacts with four other types of regulatory agencies in fulfilling this role:

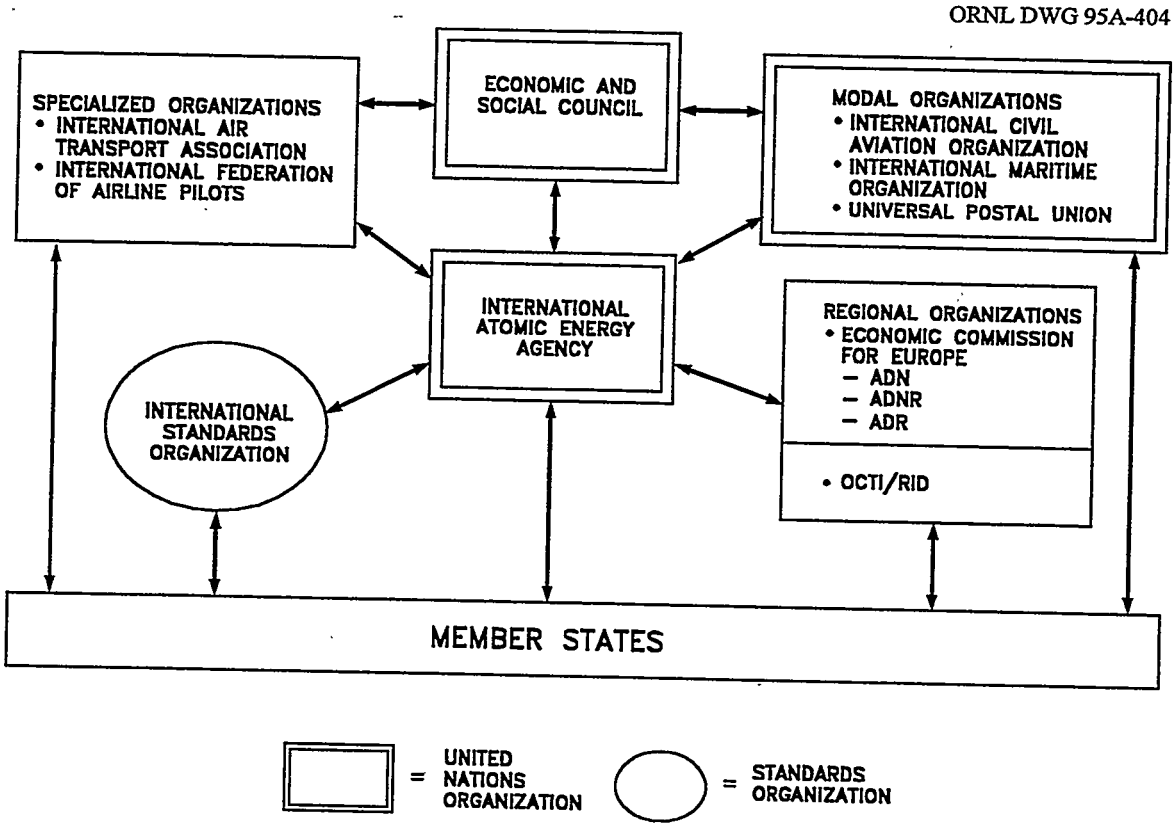

Fig. 2.1. Radioactive materials transportation regulations and standards: international interactions.

1. the UN Economic and Social Council, which is the umbrella organization for the international regulation of all hazardous materials transportation;
2. the organizations that govern modal regulations for hazardous material transportation;
3. regional organizations that govern modal regulations for hazardous material transportation within their region (generally with Europe); and
4. the member states (countries) of the IAEA.

The roles of the other international regulatory bodies and their regulatory documents are elaborated upon in Sect. 2.2.1.3.

2.2.1.1 Role of IAEA Safety Series No. 6

Safety Series No. 6 was developed in close cooperation with various national regulatory bodies and interested international organizations and with their designated experts. Whenever each edition was issued by the IAEA, it was accompanied with a statement that the regulations are recommended ". . . to Member States and to the organizations concerned as a basis for national regulations and to be applied to international transport" (from Foreword to each edition of Safety Series No. 6; see ref. 6).

Thus, each edition of Safety Series No. 6 becomes the model which the national regulators, such as DOT and NRC in the United States, follow in developing modifications to their respective domestic transport regulations. Although the requirements in Safety Series No. 6 will generally be adopted by the national regulators, additional requirements may be imposed by an individual member state, depending upon its specific environmental, social, political, or regulatory needs.

2.2.1.2 Role of the IAEA Safety Series No. 6 Support Documents

Because Safety Series No. 6 specifies *what* must be done relative to a specific regulatory requirement, it was deemed necessary to provide guidance to designers and regulators alike on a way, or ways, through which specific requirements might be satisfied. To this end, Safety Series No. 37 (ref. 5) was prepared; it provides descriptions of *how* one might proceed to satisfy some, but not all, of the requirements specified in Safety Series No. 6. Because the document represents an international consensus, it may not necessarily represent the position a specific domestic regulator may take. However, it does provide a basis from which discussions on design philosophy and compliance analyses or testing may be initiated.

Relative to Type B radioactive and fissile materials packages, Safety Series No. 37 contains fairly detailed discussions on analyses, testing, test procedures, contamination control, QA, acceleration values, and calculation methods for package tiedown forces, the influence of brittle fracture on material integrity, and criticality safety assessments.

To support a better understanding of the primary requirements imposed on the use of a specific type of package, a "schedule of requirements" by type of radioactive material consignments was developed. These schedules, collected in Safety Series No. 80 (ref. 7), identify requirements including

- applicable U.N. number or numbers;
- materials (a listing of the acceptable materials which may be included as package contents in the specific package design);
  packaging/package (a brief listing of the requirements the package design must satisfy);
- maximum radiation levels (a listing of the allowable radiation levels at various locations around the package and transporting vehicle);
- contamination and decontamination (a listing of various contamination limits and decontamination requirements relating to use of the packagings and transporting vehicles);
- mixed contents and mixed loading (a description of limitations on loading of the packagings and mixing of packages in consignments); and
- administrative requirements (a listing of labeling, marking, placarding, transport document, storage, dispatch and carriage requirements for each package).

Relative to Type B radioactive and fissile material packages, Schedule 10 [Type B(U) packages], Schedule 11 [Type B(M) packages], Schedule 12 (fissile material), and possibly Schedule 13 (material transported under special arrangement) apply.

Because Safety Series No. 6 has a long history of development and a multiplicity of requirements, it was also deemed useful to provide information to designers, regulators, and other users of the regulations on *why* specific requirements were established or modified. Safety Series No. 7 (ref. 4) was prepared to accomplish this task. It provides, based on personnel's memory and documents available, a view of why some, but not all, of the requirements specified in Safety Series No. 6 were established. It is a useful background tool for those being initiated into the use of the regulations and also for those who are trying to apply specific requirements and who desire more in-depth background knowledge concerning the requirements of interest.

Relative to Type B radioactive and fissile materials packages, Safety Series No. 7 contains fairly detailed discussions of the basis for A_1 and A_2 values and for allowable release rates from Type B packages.

2.2.1.3 Role of the Other International Organizations' Documents

In addition to serving as the basis (recommendations, model) for domestic regulations, IAEA Safety Series No. 6 also serves as the basis for all other international hazard-class-specific or mode-specific transport agreements. The key international organizations and their relationships were outlined in Fig. 2.1, and these organizations and their documents referencing or otherwise using Safety Series No. 6 as principal input are summarized in the Table 2.1.

In most cases, representatives from the organizations listed in Table 2.1 participate in the structured review process for Safety Series No. 6 and ensure compatibility among the regulatory requirements. In a like manner, representatives from the IAEA participate in meetings with ICAO, IATA, IMO, and other organizations to provide inputs as needed and again ensure compatibility wherever possible. As a result, the development of this suite of international regulatory documents is an iterative process, wherein each document "feeds" the other, and, once published, the documents are generally in concert with each other.

For example, during the preparation of the 1985 edition of Safety Series No. 6, representatives from the ICAO participated in IAEA revision panels; this resulted in the "Additional Requirements for Packages Transported by Air" (paragraphs 515 through 517 of ref. 2) being added during the

Table 2.1. International organizations/regulatory documents controlling packaging and transport of Type B/fissile radioactive material

Mode of transport	Class of hazardous material	International organization	Name of regulatory document	Geographical scope	Applicable to Type B and/or fissile packages
All	Class 7—Radioactive	IAEA	Regulations for the Safe Transport of Radioactive Material (Safety Series No. 6) (ref. 1)	Worldwide	Yes
All	All	UN	Recommendations on the Transport of Dangerous Goods[8]	Worldwide	No—defers to Safety Series No. 6
Air	All	ICAO	Technical Instructions for the Safe Transport of Dangerous Goods by Air[9]	Worldwide	Yes
Air	All	IATA	Dangerous Goods Regulations[10]	Worldwide	Yes
Sea	All	IMO	International Maritime Dangerous Goods Code[11]	Worldwide	Yes
Road	All	ECE[a]	ADR (European Agreement Concerning the International Carriage of Dangerous Goods by Road)	Regional (does not apply in U.S.)	Yes
Rail	All	OCTI[b]	RID (International Regulations Concerning the Carriage of Dangerous Goods by Rail)	Regional (does not apply in U.S.)	Yes
Inland waterway	All	ECE[a]	ADN (European Agreement for the International Carriage of Dangerous Goods by Inland Waterway)	Regional (does not apply in U.S.)	Yes
Inland waterway	All	CCNR[c]	ADNR (European Agreement Concerning the International Carriage of Dangerous Goods on the Rhine)	Regional (does not apply in U.S.)	Yes
Post	All	UPU[d]	Acts of the UPU	Worldwide	No

[a]ECE = Economic Commission of Europe.
[b]OCTI = Central Office for the International Transport by Rail.
[c]CCNR = The Central Commission for Navigation on the Rhine.
[d]UPU = Universal Postal Union.

revision process. This addition brought the new edition of Safety Series No. 6 in line with ref. 9, although it did result in adding some mode-specific requirements to Safety Series No. 6, of which designers must be aware.

Similarly, the development of the schedules[7] was coordinated with and reflects the requirements in Safety Series No. 6. These, in turn, were used in the development of requirements for ICAO, IMO, ECE, OCTI, UPU, and CCNR regulatory documents.

The key documents that must be considered worldwide for Type B radioactive and fissile materials packages are those originating from the IATA and ICAO for air carriage and from the IMO for maritime carriage. If the packages are to be used in the European area, then the regional modal documents originate from the ECE, OCTI, and CCNR. Usually, if the requirements in Safety Series No. 6 have been satisfied relative to package design, they will be satisfied for any of the other regulatory documents listed in Table 2.1. However, if the contents or the package design are materials which are not commonly carried, a review for additional requirements imposed by these other organizations is recommended.

For example, the ICAO Technical Instructions[9] contain listings of variations as notified to ICAO by (1) member states and (2) airline operators. Prior to air carriage of a Type B or fissile material package, these variations should be reviewed to ensure compliance for that specific use.

Finally, it must always be kept in mind that the regulatory documents discussed here, including Safety Series No. 6, are subject to modification at any time. The requirements discussed in the rest of this document could change, depending upon decisions made by the international bodies responsible for maintaining these documents. For example, Safety Series No. 6 has been under review since the current edition was issued in 1985, and a new edition of this document was issued in 1996.

In a like manner, the promulgation of the changes in the regulatory requirements from the international to the domestic level takes time and occurs on different schedules for different countries. Thus, whenever packages are to be used internationally, compliance with the current versions of applicable regulatory documents must be ensured. For example, although the current edition of Safety Series No. 6 was issued in 1985 and many IAEA member states have adopted this edition, others (as of 1996) have not. The United States implemented these requirements effective April 1, 1996.

2.2.2 Department of Transportation

DOT has been granted broad statutory authority by the Hazardous Materials Transportation Act and its successor acts to regulate the packaging and carriage of hazardous materials, including radioactive materials, in commerce. These regulations apply to interstate transportation of hazardous materials and to all companies that ship these materials in interstate commerce.

Although the majority of the DOT regulations are directed at smaller quantities and lower-hazard radioactive materials (excepted quantities, low specific activity (LSA) materials, surface contaminated objects (SCOs), and Type A packages), there are several provisions that must also be met in the use of certified (Type B or fissile) packagings. Some of these are applicable to all shipments (such as contamination limits), and others are specific to certified packagings.

DOT regulations applicable to certified packagings are found in 49 CFR Parts 171 to 178. Like other hazardous materials, packages and shipments of radioactive materials may be subject to specific packaging, marking, labeling, shipping paper, and placarding requirements. These sections are organized as follows:

- Part 171, "General Information, Regulations, and Definitions," contains generally applicable definitions, matter incorporated by reference, and specific allowances for using the international regulations for shipments;
- Part 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements" contains a variety of sections:
  - Subpart A provides general-purpose, scope, and applicability information on Part 172.
  - Subpart B, includes the **hazardous materials table,** which provides a tabular listing of all hazardous materials shipping names and information identifying hazard class or division, identification number, packing group, prescribed labels, special provisions, packaging authorizations, quantity limits and any special modal requirements;
  - Subpart C contains requirements for **shipping papers;**
  - Subpart D presents requirements for **marking** of packages;
  - Subpart E includes **labeling** requirements and exceptions;
  - Subpart F prescribes **placarding** requirements for shipments;
  - Subpart G includes **emergency response**-related information and requirements;
  - Subpart H prescribes **training** requirements applicable to all persons who affect the safety of hazardous materials transportation, which includes those involved in radioactive materials packaging development and use; and
  - Subpart I defines a *Radiation Protection Program* that applies to persons who offer for transportation, accept for transportation, or transport radioactive materials.
- Part 173, "General Requirements for Shipments and Packagings," contains definitions of the various classes and divisions of hazardous materials, packaging requirements, and other requirements applicable to shippers of these materials. Note that Subpart I, 49 CFR 173.401–173.478 is specific to radioactive materials and is explained in greater detail later in this chapter of the *Packaging Handbook.*
- Part 174, "Carriage by Rail," is the first of the four mode-specific parts that contain requirements applicable to carriers operating in each of the four modes of transport and others who perform required functions (such as blocking and bracing);
- Part 175, "Carriage by Aircraft"
- Part 176, "Carriage by Vessel"
- Part 177, "Carriage by Public Vessel"

This section of the *Packaging Handbook* focuses on the requirements of 49 CFR Part 173, Subpart I, which is found in 49 CFR 173.403–173.478. As stated in the scope of Subpart I, these requirements are applicable to both carriers and shippers of radioactive materials and are in addition to the other requirements found in 49 CFR and 10 CFR Part 71 (the NRC transportation regulations).

Shippers of radioactive materials must first characterize the contents to be packaged and transported (as described in Chapter 3 of this *Handbook*). A key to proper characterization of the contents are the definitions found in 173.403. For example, if the material to be transported does not have a specific activity concentration exceeding 70 Bq/g (0.002 μCi/g), it does not meet the definition of a radioactive material for transport purposes and is not subject to the DOT regulations as a radioactive material. Other key definitions are presented, and 173.403 should be

consulted whenever clarification is needed for specific words and phrases. Note also that 171.8 contains definitions that may be applicable to radioactive material matters, and which are also generally applicable to other hazardous materials.

2.2.2.1 Packages and Packagings

In applying the regulations, and in understanding the application of this *Handbook*, it is vital to understand the differences between *packaging* and *package*.

The regulations (49 CFR) specify that *packaging* means:

> the assembly of components necessary to ensure compliance with the packaging requirements... (173.403).

It is further stated that a packaging:

> may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, service equipment for filling, emptying, venting and pressure relief, and devices for cooling or absorbing mechanical shocks. The conveyance, tiedown system, and auxiliary equipment may sometimes be designated as part of the packaging.

In contrast, the regulations specify that *package* means:

> the packaging together with its radioactive contents as presented for transport. (173.403).

Therefore, a package is designed, and the package design is what is certified. Conversely, a packaging is manufactured and maintained. Thus, there may be many packagings of a single package design. When a packaging is loaded with its contents, the containment system is closed, and any additional components that are part of the package design (such as impact limiters) are attached, then the "package" is ready for shipment.

2.2.2.2 A_1 and A_2 Values

The form of the contents plays an important role in determining the packaging requirements. Special-form materials are those that have been demonstrated to meet certain tests to ensure nondispersibility of the material as specified in 173.469. Special-form contents in Type A packages are limited to activities not exceeding A_1 values. Unless the radioactive material satisfies the requirements of LSA materials or SCOs, normal-form** materials (i.e., any form not demonstrated to be special form) are limited in Type A packages to an activity not exceeding A_2.

Many of the requirements in the regulations use the A_1 and A_2 values. These are activity values (expressed in TBq or Ci) that are derived from dosimetric models and provide consistent dose-based references. For example, an A_2 value of any radionuclide represents the same potential internal hazard during transport. An A_1 value of any radionuclide represents the same external

**In the U.S. regulations (10 CFR Part 71 and 49 CFR Part 173) the term "normal form," "other form," and "other than special form" are used interchangeably.

hazard during transport. The A_1 and A_2 values are listed in 49 CFR 173.435. For radionuclides that are not listed in 173.435, a table (Table 10) is provided in 173.433(g) for general A_1 and A_2 values. Similar tables are provided in 10 CFR 71, Appendix A.

For mixtures of radionuclides, it is necessary to ensure that the total quantity is within the Type A limits. For special-form radioactive material, the "ratio rule" is used. The activity of each radionuclide is divided by that radionuclide's A_1 value, and the resulting fractions are summed. If the summation is less than or equal to unity, the total activity of the mixture is within the Type A limits for special form.

This equation is

$$\sum_i \frac{B(i)}{A_1(i)} \leq 1 \ . \tag{2.1}$$

The $B(i)$ is the activity of radionuclide i in the mixture, and $A_1(i)$ is the A_1 value for radionuclide i. Alternatively, an A_1 value for the mixture of special form radioactive material may be determined using

$$A_1 \text{ for mixture} = \frac{1}{\sum_i \frac{f(i)}{A_1(i)}} \ , \tag{2.2}$$

where $f(i)$ is the fraction of activity of nuclide i in the mixture.

For mixtures of radionuclides in normal form, two approaches may be taken to determine if the quantity present is a Type A quantity. The ratio rule can be used—that is, the activity of each radionuclide present is divided by that radionuclide's A_2 value—and the resulting fractions are summed. If the summation is less than unity, the total quantity is within the Type A limits for normal-form material. In this case, the equation is:

$$\sum_i \frac{B(i)}{A_2(i)} \leq 1 \ , \tag{2.3}$$

where $B(i)$ is the activity of radionuclide i in the mixture and $A_2(i)$ is the A_2 value of radionuclide i. Alternatively, an A_2 value for mixtures of other than special form radioactive material may be determined using

$$A_2 \text{ for mixture} = \frac{1}{\sum_i \frac{f(i)}{A_2(i)}}, \qquad (2.4)$$

where $f(i)$ is the fraction of activity of nuclide i in the mixture.

2.2.2.3 Excepted Packages

Smaller quantities of radioactive materials may be shipped in excepted packages, which require minimal packaging and shipping documentation. Excepted packages are the easiest type to prepare and transport and include four subcategories of materials as contents:

1. limited quantities of radioactive material which are small fractions of the Type A quantities (see 173.421 and 173.425).
2. instruments and articles having radioactive materials as component parts (see 173.424 and 173.425).
3. manufactured articles containing natural thorium or uranium (see 173.426).
4. empty packagings (see 173.428).

Note that excepted quantity radioactive materials that present multiple hazards (meet the definition of another hazard class) must also comply with 173.423.

2.2.2.4 LSA Materials and SCOs

Materials with sufficiently limited specific activity (LSA materials per 173.403) and certain contaminated objects (SCOs per 173.403) can also be more easily packaged and transported than Type A and Type B quantities of material. LSA material and SCO contents require less stringent packaging, known as Industrial Packagings (IP), which meet the packaging requirements of 173.411 (which requires that packaging be designed and constructed in accordance with 173.410 and 173.411). IP integrity is graded in relation to the hazard posed by its contents. There are three industrial package types: Type 1, Type 2, and Type 3 (abbreviated IP-1, IP-2, and IP-3, respectively). IP-1 packages must meet general design requirements 173.410. IP-2 packages must also meet the free-drop and stacking tests, and IP-3 packages must in addition satisfy water-spray- and penetration-test requirements. The overall regulations applicable to LSA and SCO are found in 173.425.

The preceding general requirements, which became effective April 1, 1996, apply domestically and also outside of the United States. Because these represent significant changes to the way in which LSA materials and SCOs were handled, alternatives to these requirements are allowed by the U.S. regulators (DOT and NRC). Table 2.2 presents a summary of the minimum requirements for various forms of radioactive material contents, and compares the current regulatory requirements (effective April 1, 1996) with the previous regulatory requirements.

Table 2.2. Minimum requirements for packaging radioactive materials

Regulations effective April 1, 1996				
	0.002 µCi/g ⇓	Limited Quantity ⇓	New A_1/A_2 ⇓	1 rem/h @ 3m, Unshielded ⇓
Non-LSA/SCO		Excepted	Type A	Type B[b]
Domestic or non-domestic LSA-I Solid LSA-I Liquid Transported Under Exclusive Use SCO-I		Excepted	IP-I	Type B[b]
LSA-I Liquid Transported other than Under Exclusive Use LSA-II Solid LSA-II Liquid and Gas Transported Under Exclusive Use LSA-III Transported Under Exclusive Use SCO-II		Excepted	IP-II	Type B[b]
LSA-II Liquid Transported other than Under Exclusive Use LSA-II Gas Transported other than Under Exclusive LSA-III		Excepted	IP-III	Type B[b]
Domestic only LSA-I, LSA-II, LSA-III, SCO-I, SCO-II		Excepted	Strong-tight[a]	Spec 7A Type A / Type B[b] / NRC Type A[b,c]
Regulations in effect prior to April 1, 1996				
Non-LSA		Excepted	Type A	Type B[b]
LSA		Excepted	Strong-tight[a] or Spec 7A Type A	NRC Type A[a,b]
	⇑ 0.002 µCi/g	⇑ Limited Quantity	⇑ Old A_1/A_2	⇑ 1 rem/h @ 3m, Unshielded

Notes:
[a]Exclusive use is required.
[b]Subject to conditions in Certificate of Compliance if an NRC package.
[c]Exclusive use is required, and use expires April 1, 1999.

2.2.2.5 Type A Packages

If the material to be shipped cannot qualify as an excepted quantity, LSA, or SCO, the next higher level of requirements is applicable to Type A quantities. Type A quantities are defined in 49 CFR 173.403 as those not exceeding specified total activities (A_1 or A_2 values, listed in 173.435 or calculated in accordance with 173.433). If the material is in normal form, the A_2 value is applicable; if the material is in special form, the A_1 value is applicable.

Type A packages are authorized by 173.415. The majority of authorized Type A packages are designed, tested, self-certified, and used as DOT Specification (Spec) 7A. Each user of a

Specification 7A (the shipper) is required to maintain complete documentation of tests and other data showing that the packaging complies with 178.350. Basically, Type A package designs must meet the general design requirements applicable to all packagings (173.410) and the additional design requirements applicable to Type A packages (173.412). The performance requirements specify that the design must be able to withstand the Type A packaging tests of 173.465, including:

- water spray test of 5 cm/h (2 in./h) for at least 1 h;
- free-drop test onto an unyielding target from 1.2 m (4 ft) [graduated down to 0.3 m (1 ft) for heavy packages];
- stacking test for 24 h under a compressive load of five times the mass of the package or 1300 kg/m^2 (1.9 lb/in.^{2}) times the vertically projected area of the package; and
- penetration test of a 3.2-cm (1.3-in.)-diam bar weighing 6 kg (13.2 lb) dropped from 1 m (3.3 ft).

Package designs for liquid and gaseous contents must also withstand the free drop from 9 m (30 ft) and the penetration test from 1.7 m (5.5 ft).

Detailed guidance for the application of the new LSA material and SCO requirements will be published jointly by DOT and NRC as NUREG-1608/RSPA Advisory Guidance 97-005 entitled *Categorizing and Transporting Low Specific Activity Materials and Surface Contaminated Objects*. This document was published in draft form for comment in June 1997.

2.2.2.6 Type B and fissile packages

Larger quantities of radioactive material shipped in a single package, known as Type B quantities, are defined as those quantities exceeding the Type A limits. Fissile radioactive materials are defined in 173.403 and include most of the radionuclides capable of undergoing fission (^{238}Pu, ^{239}Pu, and ^{241}Pu and ^{233}U and ^{235}U). Certain exceptions are made in the definition, and specific exceptions are listed in 173.453 for materials that are criticality safe in unlimited numbers under all conditions of transport. Packages for fissile materials are also categorized as Type IP (i.e., IP-1, IP-2, or IP-3), Type A, or Type B, depending on the total activity of their contents, and must meet the performance requirements applicable to those categories.

Once it is determined that the radioactive material to be shipped is a Type B quantity and/or fissile, the shipper must find authorization in Title 49 for proper packaging of the contents. Section 173.416 lists authorized Type B packages, and 173.417 lists authorized packagings for fissile materials. The authorized packagings of most interest in this handbook are those authorized in 173.416(b) and 173.417(a)(4) and (b)(3). These paragraphs and subparagraphs authorize the use of packaging that meets the applicable NRC regulations and has been approved by the NRC. Paragraph 173.7(d) contains a provision that authorizes the use of DOE-approved packagings when they are evaluated, approved, and certified by DOE to standards equivalent to those of the NRC. DOE-approved packagings are generally limited to use for shipments made by or under the direction of the DOE.

Other requirements of Part 173 of particular interest to certified package users include:

- 173.431—limits the contents of Type A packages to A_1 or A_2 and the contents of Type B packages to that which is specified in the applicable approval certificate,

- 173.441—specifies package and conveyance radiation level limits,
- 173.442—limits the exterior temperature of packages,
- 173.443—specifies limits on both fixed and non-fixed contamination on package surfaces,
- 173.451 to 173.459—specifies requirements for fissile packages and shipments of fissile materials,
- 173.467—requires Type B packaging and packaging for fissile materials to meet the NRC requirements for ability to withstand accident conditions,
- 173.468—specifies tests for LSA-III materials,
- 173.469—identifies tests for special form materials (applicable if the certified package design requires special form materials),
- 173.471—states requirements for NRC-approved packages (note that under 173.7, DOE packages must meet requirements equivalent to these),
- 173.474—identifies QC for construction,
- 173.475—identifies QC prior to each shipment,
- 173.476—identifies approval requirements of special-form material (applicable if the package design requires special-form material),
- 173.477—identifies approval requirements for export shipments, and
- 173.478—identifies notification requirements for export shipments.

Consequently, many aspects pertaining to the use of certified packagings is regulated closely by the DOT regulations. The authorization to use the packages is found in Title 49, but the technical specifications (tests, design requirements, etc.) are incorporated by reference to the NRC regulations, although many Type B and fissile material packages are authorized as specification packages (49 CFR 178).

2.2.2.7 Transport Index

One operational aspect which the package designer should be familiar with is the Transport Index, or TI. The TI is a dimensionless numerical value that provides information on loading and storage limits based on the number of packages which can be safely accumulated into a single group. In the case of fissile materials package designs, the TI is based on package design characteristics, whereas for non-fissile packages the TI is based on general radiation protection principles (see 49 CFR 173.403). Specifically, the TI is the greater of

1. the maximum radiation level measured in mrem/h at 1 m (3.3 ft) from the package surface or
2. for fissile package designs, the number 50 divided by the allowable number of packages which may be accumulated in a group (see Chapter 10).

The TI is used by operations personnel and carriers to limit accumulations of packages to provide both radiation protection and criticality safety. By limiting the number of TIs in a single group of packages to 50, the resulting radiation level around the group of packages is controlled. By limiting the TIs for fissile packages on board a vehicle or conveyance, criticality safety is provided by limiting the amount of fissile material present to a safe amount.

For a non-fissile package, the TI is the maximum radiation level, in mrem/h, at 1 m from the package surface. The actual TI for a non-fissile package will depend on the quantity and decay characteristics of the contents loaded into the packaging at the time of transport (since the radiation level will depend on this). Specifically, the TI is the value determined by measurement

of the radiation level at 1 m from the package surface after it is loaded and sealed with its radioactive contents and ready to be presented for transport; as the radioactive contents decay, the TI of the package will not be modified.

For fissile package designs, the designer must first determine the allowable number of packages in accordance with the conditions, as described in Chapter 10. Package users then assign the TI on the basis of the actual radiation level at 1 m (item 1 above) or the assigned criticality control value (item 2 above), whichever is greater.

For both non-fissile and fissile packages, the package user needs to ensure adequate shielding to avoid exceeding the maximum allowable radiation levels (173.441).

2.2.3 Nuclear Regulatory Commission

NRC regulations provide detailed design, test, fabrication, approval, and QA requirements and are found in 10 CFR Part 71. They are applicable to all NRC and agreement-state licensees. Unlike the DOT regulations, which are universally applicable to all interstate and most intrastate shippers of radioactive materials, the NRC regulations are applicable only to licensees and only when the shipments involve licensed material (source, by-product, or special nuclear material). In practice, however, the NRC regulations are also nearly universal in application because DOT includes specific requirements that the NRC regulations be met when certified packages are used. Even DOE-certified package activities are heavily influenced by the NRC regulations because DOT requires that DOE package approvals and uses be equivalent to those required by NRC.

The scope of 10 CFR Part 71 includes both transportation and packaging requirements. However, since this *Handbook* focuses on certified packaging (Type B and fissile), this section emphasizes the items of importance to these types of packagings.

2.2.3.1 Excepted Quantities, LSA, SCO, and Type A Packages

NRC requirements for these materials are essentially the same as the DOT requirements which contain specific requirements that the DOT regulations be followed when packaging and transporting these types of materials and packages. Note that only packages containing LSA materials or SCOs with an external radiation level no greater than 10 mSv/h (1 rem/h) at 3 m from the unshielded material are allowed without specific design approval. LSA and SCO package designs for materials or objects having higher unshielded radiation levels must be certified by the NRC as Type B package designs.

2.2.3.2 Type B Packages

The framework of Type B packages is divided into three categories (not including combination Type B and fissile package designs) under the NRC regulations. The first category—Type B package designs approved and fabricated prior to August 31, 1986—generally uses designations of "Type B ( )" with the parentheses empty because the package approval was given under the NRC regulations, which were consistent with the 1967 edition of the IAEA regulations. Some of these package designs may still be encountered, but their use will continue to dwindle as the designs are updated or modified. As specified in 71.13, no more of the packagings may be

fabricated, and only modifications that are "not significant with respect to the design, operating characteristics, or safe performance of the containment system" will be allowed without a demonstration that the design meets the current performance requirements (which are consistent with the 1985 IAEA regulations).

The second and third categories of Type B package designs are those that meet the current 10 CFR Part 71 performance requirements. These designs are categorized as Type B(U) or Type B(M). The "U" designates a package design requiring only unilateral approval—that is, approval by the country of origin only. In spirit, these package designs are intended to be designed and fabricated to such high standards that countries which would be receiving these packages in international transport (imports) would not feel compelled to review the designs. The approval of the country of origin would suffice. In practice the receiving country revalidates the certification of the country of origin. The "M" designates a package design requiring multilateral approval—that is, approval by all countries into or through which the package is transported. Initially, there were many differences between the two designations, but the current NRC regulations have reduced the differences to those specified in 71.4, under the definition of "Type B package." A Type B(M) package design is one that meets the Type B requirements but has:

- a maximum normal operating pressure less than 700 kPa (100 lb/in.^{2}) or
- a pressure-relief device which would allow the release of radioactive material to the environment under the hypothetical accident condition tests (71.73).

In other respects, Type B(M) and Type B(U) are identical. In addition, relative to domestic transportation, "there is no distinction made in how packages with these designations may be used" (10 CFR 71.4).

2.2.3.3 Fissile Packages

Fissile materials are defined exactly the same way by the NRC and the DOT. Specific package design and performance requirements are applicable to fissile packages, especially those found in 71.55 and 71.59. Note that the fissile package requirements are in addition to the requirements which are based on the radioactivity of the contents. This means that both IP, Type A and Type B packages may also be approved for fissile contents, resulting in what is known as Type IF, Type AF and Type BF package designs.

2.2.3.4 NRC Regulations

The NRC regulations in 10 CFR Part 71 are organized into eight subparts. Each of these subparts are listed below, along with a brief discussion of the focus and major requirements of the subpart:

- Subpart A, "General Provisions," defines the scope and applicability of Part 71 and includes the major definitions used. Also contained are specific requirements to comply with the DOT regulations.
- Subpart B, "Exemptions," provides exemptions for physicians and for licensees transporting LSA, SCO, and Type A packages when certain requirements are met. Exemptions in the NRC regulations are defined situations in which some or all of the regulations are waived, meaning that compliance is not required. For example, the exemption for materials with a specific

activity not exceeding 0.002 μCi/g is all-encompassing. The exemption for Type A packages is partial and conditional on compliance with 71.5, 71.53, and 71.88.

- Subpart C, "General Licenses," provides general licenses (authorizations contained in the regulations themselves) for "grandfathering" earlier certified package designs using DOT-specification package designs, and for packaging and transporting fissile material under certain conditions. The general licenses prescribe the conditions that must be fulfilled in order to use the license, including actions such as registration. General licenses for fissile material provide a number of specific situations in which fissile material (with a total activity not exceeding a Type A quantity) may be packaged and transported without compliance with Subparts E and F. Note that the NRC definition of the term *limited quantity* (such as in the titles of 71.18 and 71.22) and the DOT definition of the term (meaning a small fraction of a Type A quantity) are different.
- Subpart D, "Application for Package Approval," specifies the information required in an application for package design certification and for the process for renewing certificates. These sections detail the requirements for the contents of the approval application (71.31), including required detailed information on the description of the package (71.33); evaluation of the package to the design and performance requirements (71.35); and QA for the design, fabrication, assembly, testing, maintenance, and use of the package (71.37). The process for renewing certificates of compliance are presented in 71.38. A provision in 71.39 also allows that the NRC may at any time require additional information in order to determine if a certificate of compliance should be granted, modified, or otherwise acted on. This emphasizes the need to provide complete and thorough information in an application for package design approval.
- Subpart E, "Package Approval Standards," specifies the design requirements for Type B and fissile package designs and other operational requirements. Section 71.41 contains the requirement that a package design must be tested to demonstrate compliance with the normal conditions of transport and the hypothetical accident conditions, unless another method acceptable to the NRC is used in lieu of testing. It also provides that the vehicle design and operational controls may be considered along with the package design if the NRC determines that this is appropriate for the shipment This subpart is wide-ranging and covers requirements concerning package tiedown devices which are a part of the package, external radiation-level limits, and Type A package requirements. It specifies exemptions for LSA materials and SCO's and for certain fissile materials (consistent with the DOT fissile exceptions). Specific requirements are provided for fissile materials package designs, including specifying how arrays of fissile packages must be considered. Special requirements for irradiated fuel and for packaging high-toxicity radionuclides (i.e, the double-containment requirement), and specific requirements for plutonium shipments by aircraft (including packaging) are also contained in this Subpart.
- Subpart F, "Package and Special Form Tests," provides the test requirements for normal and hypothetical accident conditions of transport and plutonium-accident conditions and for special-form materials. It is worth noting that the normal and hypothetical accident conditions of transport (71.71 and 71.73) specify also the initial conditions (e.g., ambient air temperatures prior to and following testing) which must be used.
- Subpart G, "Operating Controls and Procedures," includes requirements for shipment controls, record keeping, and advance notification for certain waste shipments; and
- Subpart H, "Quality Assurance," specifies the QA requirements applicable to all facets of certified package utilization.

Appendix A to Part 71 provides the A_1 and A_2 table, rules for calculating A values, and specific activity values for radionuclides. This information is consistent with the DOT regulations on the same topics.

2.2.3.5 Package Design and Test Requirements

The transportation regulations, as reflected in 10 CFR 71, provide a graded approach to package design and testing; the greater the hazard of the material contained within a packaging, the greater are the design and test requirements for the package design. For very low-hazard materials, only general requirements are applied which are intended by those developing the regulations to apply to routine conditions of transport. In contrast, for high-hazard materials such as Type B quantities and fissile materials, a complex set of requirements applies including the requirement to design for the hypothetical accident conditions (10 CFR 71.73). A summary of the graded approach to design and testing requirements is provided in Table 2.3. The appropriate sections of 10 CFR and 49 CFR have been cited in this table.

Table 2.3. Design and test requirements for the transport of radioactive materials

Requirements	Excepted Packages	Industrial Packages			Type A	Type B
		IP-1	IP-2	IP-3		
General Requirements (71.43-71.47; 173.410)						
Lifting, stowing, vibration, materials	X	X	X	X	X	X
Additional requirements if carried by air [173.40(i)]						
Temperature, pressure		X	X	X	X	X
Normal conditions of transport (71.71; 173.465)						
Free drop 0.3-1.2 m			X	X	X	X
Stacking			X	X	X	X
Water spray				X	X	X
Penetration 1.0 m				X	X	X
Normal conditions of transport for liquid contents (173.466)						
Free drop 9 m					X	
Penetration 1.7 m					X	
Accident conditions (71.73)						
9.0-m drop						X
9.0-m dynamic crush						LLD[a]
1.0-m puncture						X
Thermal						X
Immersion						
15 m						X
External pressure 2 MPa (290 psi)						INF[b]
Water leakage evaluation for criticality (71.73)		- - - If fissile material - - -				

[a]For lightweight, low-density (LLD) packages [71.43(c)(2)].

[b]For Irradiated Nuclear Fuel (INF) packages with activity greater than 37 PBq (10^6 Ci) a pressure test equivalent to a 200-m immersion test (71.61).

2.2.4 U.S. Department of Energy

2.2.4.1 Introduction

DOE has a unique role in the United States in radioactive materials transportation. It conducts operations that produce radioactive materials as a by-product, some of which are used in medicine for diagnosis and treatment of cancers. Other materials are used to make fuel for nuclear reactors. Many materials are simply radioactive wastes that must be processed, stored, or disposed of by DOE and its contractors. DOE's production of radioactive materials requires DOE to ship and carry tens of thousands of packages annually.

Because it is a U.S. government agency, DOE is not subject to the transportation and packaging regulations of DOT. Additionally, DOE is exempt from the regulatory control of NRC. Because of their relationship with DOE, contractors are also exempt from NRC regulatory control. However, DOE contractors are by law, subject to the regulations of DOT.

To ensure that DOE transportation and packaging operations are conducted safely, DOE has developed internal directives or Orders that address transportation and packaging safety and incorporate the requirements of DOT and NRC. This action assures the general public that DOE complies with the same requirements as the commercial sector. These Orders apply to all DOE elements and contractors involved in transportation and packaging operations. DOE contractors meeting the requirements of the DOE Orders will satisfy their obligations under the DOT requirements.

2.2.4.2 DOE Order 460.1A

DOE Order 460.1A establishes (a) requirements for packaging radioactive and other hazardous materials and (b) DOE's approval process for the acceptability of using some types of packaging. Before approval of a Type A packaging, DOE conducts actual tests on the package to verify that its performance meets acceptable standards. Analogous to the NRC, DOE conducts a certification program for Type B and fissile packagings. Finally, DOE certifies the design for special-form radioactive material for conformance with the requirements of DOT and NRC.

2.2.4.3 Type A Packaging

Packaging used to ship a Type A quantity of radioactive material must be designed and constructed in compliance with the requirements of 49 CFR Part 173, Subpart I. To demonstrate that the packaging meets regulatory requirements, DOE conducts a program of testing. A prototype is submitted to the approving authority for use authorization. The approving authority causes the prototype to be tested for conformance with the regulatory requirements. When a prototype has been shown to meet the regulatory requirements, an approval record is issued and compiled into the *Test and Evaluation Document for DOT 7A Type A Packaging*.[12] Any DOE facility or contractor may use a packaging found in this document as long as the packaging is used in accordance with its intended parameters. DOE-tested Type A packagings are approved for a period of about 7 years, after which time additional information verifying that the packaging design conforms to existing standards may be needed.

Packagings purchased from commercial sources do not require retesting by DOE. However, a commercially purchased Type A package must be shown to meet the same performance and QA requirements as a DOE-manufactured package. Additionally, the user must obtain and maintain on file, documentation that shows the packaging meets the DOT requirements.

2.2.4.4 Type B and Fissile Packaging

Packaging used by DOE and its contractors to ship radioactive materials in quantities greater than an A_1 or A_2 except for such materials satisfying the requirements for LSA materials and SCOs, must be designed and constructed in compliance with the requirements of 49 CFR Part 173, Subpart I and 10 CFR Part 71. Specific packagings authorized to be used by DOE are found in 49 CFR 173.416 and 173.417. Additionally, pursuant to 49 CFR 173.7(d), DOT permits DOE to certify its own packagings as long as the certification program is equivalent to the one conducted by the NRC. This exemption allows DOE to use packagings which are not available to non-DOE shippers.

49 CFR 173.416 and 173.417 authorize certain packagings for which no additional certification is required. DOE Order 460.1A (ref. 13) generally authorizes the use of these packagings for no additional certification. For example, packagings certified by the NRC may be used without prior DOE approval as long as DOE is a registered user of the package.

However, DOE Order 460.1A requires that DOT Specification 6M packagings meet a requirement not found in 49 CFR Part 173, Subpart I. DOT Specification 6M packagings used by DOE and its contractors must provide double containment for packagings containing more than 0.74 TBq. This double-containment requirement is based on the requirement of 10 CFR 71.63. Approval for specific use of doubly contained packages for plutonium must be obtained from the Department's Headquarters Certifying Official.

In addition to DOT specification packaging and NRC certified packaging, DOE and its contractors are permitted to use packaging designs certified by DOE. The packaging certification program conducted by DOE is equivalent to that of the NRC. DOE closely monitors the guidance given by the NRC to its licensees and incorporates the applicable parts into its review procedures. The packaging that is certified by DOE provides a level of protection at least equivalent to that of the NRC. DOE's Certificate of Compliance may also be used as the basis for a DOT Certificate of Competent Authority for international transportation.

2.2.4.5 Special-Form Radioactive Materials

Some DOE-approved or certified packagings depend on containment as provided by special-form radioactive material. To verify that containment credit can be taken, DOE certifies the design of special-form radioactive materials in a manner similar to that of DOT. The design and tests requirements for special-form materials are found in 49 CFR 173.403 and 173.469, and 10 CFR 71.4, and 71.75. DOE's Certificate of Compliance provides a pedigree for the special form material. The Certificate of Compliance may be used as the basis for a DOT Certificate of Compliance for international transportation.

2.2.4.6 Summary

Because DOE is not subject to DOT and NRC requirements and because DOE contractors are exempt from NRC's requirements, DOE has developed packaging and materials approval and certification programs that are necessarily consistent with the regulatory program of those two agencies. These approval and certification processes required by DOE ensure that DOE's transportation and packaging activities provide the same level of safety as that of the commercial sector.

2.2.5 Future Changes in Regulations

To keep the world's transport regulations abreast of recent scientific and technological developments, a major review takes place at intervals of approximately 10 years. This time interval allows international organizations and IAEA Member States to schedule rule-making activities regarding the revised Regulations while maintaining an acceptable level of regulatory stability. In continuation of this process, the 1985 edition of the IAEA Regulations[2] has now been followed by a 10-year review process and the publication of updated regulations (as Safety Standards Series No. ST-1)[14] has been completed and was published by the IAEA in late 1996. In addition, the new regulations issued by the IAEA have combined what was formerly Safety Series No. 6 (ref. 2) with the schedules.[7] It is the intent of the IAEA to issue in 1997 revised explanatory and advisory material, where a new document (Safety Standards Series No. ST-2) will replace the former Safety Series No. 7 (ref. 4) and Safety Series No. 37 (ref. 5). The following summarizes some of the changes which have been made in the regulations which can be expected to be promulgated into future domestic requirements.

2.2.5.1 Changes Related to the International Basic Safety Standards

One of the major topics considered in the Revision Process is the incorporation of the new Basic Safety Standards (BSS) for radiation protection.[15] The BSS have been revised to reflect the consensus surrounding the latest recommendations of the International Commission on Radiological Protection (ICRP 1991) and the new Regulations call upon the BSS as a general provision for radiological protection. Consequently, this revision of the transport regulations needed to take into account the revised BSS requirements.

The "General Provisions" on Radiation Protection have been changed, including the need to establish Radiation Protection Programs (RPPs) for the transport of radioactive material. RPPs emphasize the importance of the General Provisions which provide the justification for maintaining the current regulatory limits for radiation levels around packages and conveyances. It is important to recall that these limits have led to low individual and collective doses to both workers and members of the public.

Dose assessment programs for occupational exposures arising from transport operations will be prescribed on the basis of likely annual doses. For occupational exposures which are unlikely to exceed 1 mSv/y, no special actions such as special work patterns, detailed monitoring, nor individual record keeping are required. Workplace monitoring is required for exposures expected to be in the range of 1-6 mSv/y. Individual monitoring is required for exposures likely to exceed 6 mSv/y. To assist operators in estimating the exposure of their workforce, advice has been provided that correlates exposure with the number of packages handled and the radiation level at

1 m from the packages. It is believed to be unlikely that carriers handling less than 300 TI per year will exceed the 1 mSv threshold. This graded approach is an adaptation of the current Regulation which has proven to be practicable in transport applications. Workers engaged by consignors and consignees can be expected to be covered by radiation protection programmes administered at the fixed-site. Carriage, on the other hand, is a transient operation for which the classification of work areas can be difficult to apply.

The Regulations have always contained an exemption criteria which defined materials subject to their requirements. The current Regulations define radioactive material as any material having a specific activity greater than 70 Bq/g. The BSS, however, use a radionuclide-specific approach which leads to derived exemption values spanning seven orders of magnitude, and straddling 70 Bq/g in the case of activity concentration. The BSS also present exemption values for total activity quantities (Bq). After considerable deliberation, it was agreed by the experts involved that the BSS derived exemption values should be adopted directly into the transportation regulations. As a result, the new Regulations include both activity concentration and total activity per consignment exemption values. For mixtures of radionuclides, the "ratio rule" must be applied so that sum of the activities (or activity concentrations) present for each radionuclide divided by the applicable exemption value is less than or equal to 1. Examples of the new exemption values are shown in Table 2.4.

Table 2.4. Exemption values for selected radionuclides

Nuclide	Activity concentration (Bq/g)	Activity per consignment (Bq)	Nuclide	Activity concentration (Bq/g)	Activity per consignment (Bq)
Co-60	1×10^1	1×10^5	Ra-226	1×10^1	1×10^4
Sr-90	1×10^2	1×10^4	U-nat	1×10^0	1×10^3
Cs-137	1×10^1	1×10^4	Pu-239	1×10^0	1×10^4

The A_1/A_2 Values, which are based upon the Q System (ref. 4) were re-evaluated. The "Q System" is a dose-based set of models which is used to derive the A_1 and A_2 values in the Regulations. The A_1 and A_2 values are activity quantities, calculated for each radionuclide, that set the limits on contents for Type A packages and for specifying other activity limits. The fundamental assumptions in the Q System constrain the detriment to an individual in the event of serious damage to a single Type A package by restricting the dose to the order of 50 mSv. The impact of the BSS on the Q System is limited since the Q System falls in the domain of potential exposures. Potential exposures are not expected to be delivered with certainty, and can result from an accident or events of a probabilistic nature. Since potential exposures are not subject to the dose limits applying to normal exposures (20 mSv a^{-1}, in general), the reference dose of 50 mSv can continue to be used in the context of the Q System. However, a group of specialists have calculated revised A_1 and A_2 values based on complete spectral emissions from radionuclides and also taking into account new radiation weighting factors, new tissue weighting factors, and the latest metabolic models incorporated into the BSS. Even though there is an underlying desire to keep the Regulations as stable as possible, maintaining the scientific rigour of the Q System was felt to be an overriding consideration for ensuring that the regulations remain abreast of current thinking in radiological protection.

2.2.5.2 Other Key Changes to the Regulations

A new package type, known as Type C packages, has been added. The Type C package is a more robustly designed package type, which will be required for use for certain high-activity shipments transported by aircraft. Any package design for more than 3,000 A_2 for normal-form material, or 3,000 A_1 (not to exceed 100,000 A_2) for special-form material, intended for transport by air, will be requireed to meet the Type C requirements. The performance requirements include:

- those applicable to Type B(U) packages and, if appropriate, packages for fissile materials;
- a puncture/tearing test consisting of a truncated cone-shaped probe which is either dropped 3 m onto the package or the package is dropped onto the probe, depending on the mass of the package specimen;
- an enhanced thermal test, with the same technical specifications as the Type B package thermal test but with a duration of 60 minutes;
- a 200-m water immersion test; and
- an impact speed of 90 m/s for the "drop" test.

Type C package designs will require unilateral Competent Authority approval unless they contain fissile material which requires multilateral approval. There are no "grandfathering" provisions for Type C packages which means that as soon as the 1996 revision is implemented by Member States or the International Civil Aviation Organization, they will be required.

There is recognition that NUREG-0360 (ref. 16) is more stringent than the requirements contained in the Regulations. NUREG-0360, inspired by political mandate, seeks almost absolute protection irrespective of the probability of occurrence of an accident. It was recognized that there are differences, such as the required impact velocity and the sequencing of tests. Following a re-evaluation of the data provided by France and the United States, experts at the IAEA agreed that both data sets support an impact speed of 90 m/s as contrasted with the NUREG-0360 requirement of 129 m/s.

Since the primary hazards being addressed in Type C requirements are dispersion and radiation levels, provisions have been made for materials which exhibit limited dispersibility, solubility, and radiation levels. These provisions are contained in a material category known as "low dispersible material" (LDM). It was accepted that material (without any packaging) that has limited radiation levels, which when subjected to the Type C impact and thermal test would only produce limited gaseous, fine particulate, or dissolved aqueous activity (less than 100 A_2) should be excepted from the Type C packaging requirements. Test specifications for LDM material are included in the regulations and Type B packages are authorized for their transport by air with the limit on total activity only being that specified in the approval certificate for the Type B package. Multilateral Competent Authority approval of the Type B package design and the design of the LDM is required.

There will be new packaging requirements for uranium hexafluoride UF_6. The decision to draft regulations for a specific material reflects the importance of UF_6 within the nuclear fuel cycle, the very large quantities being shipped, and the peculiar physical and chemical properties of the material. After considerable deliberation, it was agreed that the new requirements would be that packages:

- must withstand an internal test pressure of at least 1.4 MPa, but cylinders with a test pressure less than 2.8 MPa require multilateral approval;
- must withstand the "Type A" drop test, with graduated heights from 0.3 to 1.2 m, depending on package mass;

- designed to contain 0.1 kg or more but less than 9,000 kg of UF_6 must meet the "Type B" thermal test of 800°C for 30 minutes;
- designed to contain 9,000 kg or more must either meet the thermal test requirements or have multilateral approval;
- containing fissile UF_6 must meet the test conditions applicable to fissile packages ("Type B" impact and thermal tests) with no contact between the valve and other normally noncontacting parts of the packaging, have no leakage from the valve, and meet other operational requirements before the designer can assume no in leakage of water for the safety analysis; and
- have at least unilateral Competent Authority design approval after 31 December 2003, with certain exceptions (test pressure less than 2.8 MPa or not meeting the thermal test) where multilateral approval is required after 31 December 2000.

The new regulations creates two package indexes. The transport index (TI) for radiation protection is unchanged and continues to be based on the radiation level at 1 m. A new criticality safety index (CSI) is based on the allowable number of packages that can be transported together. Currently, the transport index fulfills a dual function in controlling both hazards. Separation of the two indices will allow shipments to be controlled on the basis of the specific value of concern. For example, fissile packages with low radiation levels will not have to be segregated from persons on the basis of a high TI. Additionally, several tables in the Regulations will be simplified, since two simple rules are being substituted for the more complex combination of possible consignment make-ups. The changes introduce clarity which should enhance compliance with the regulations.

New UN numbers will be assigned. The primary purpose of displaying UN numbers on packages, and on conveyances when appropriate, is to key into emergency response procedures in a language-independent way. It was felt that an expanded set of UN numbers would provide emergency workers with more specific response guidance.

Under the new system a UN number is assigned to each of the Schedules appended to the Regulations, with a set of numbers for packages containing fissile material. UN numbers are retained for uranium hexafluoride because of its importance as a commercial substance and its subsidiary (corrosive) risk. The new UN numbers will facilitate emergency response procedures and help with compliance checks and controls through a numerical link with the Schedules. The 1995 Edition of the UN Recommendations for the Transport of Dangerous Goods[8] includes the new UN numbers. Some UN numbers become redundant, but as none of the deleted numbers are re-used the transition process will be eased.

The provisions for criticality safety were also comprehensively reviewed and several significant changes were adopted. These changes include revisions of existing provisions and several new packaging performance requirements. Specifically, the revised requirements include:

- revised fissile exception limits, including a "fissile mass per consignment" limit for packages containing: less than 15 g of fissile material; hydrogenous solutions with an X/H ratio less than 5% by mass; and, material with no more than 5 g of fissile material in any 10 litre volume;
- only one type of fissile exception being allowed per consignment;
- a requirement for a measurement to be performed after irradiation but prior to shipment in the case of using "burn up credit" in the package design;
- if intended for air transport, the package must remain subcritical under conditions consistent with the Type C tests, but without consideration of water in leakage; and

- applicability of the Type B "crush test" for low density (less than 1,000 kg/m^3) lightweight (less than 500 kg) packages designed for non-excepted fissile material.

2.3 STANDARDS

Section 2.2 described the regulations that are imposed by law or other legally binding convention on the packaging and transportation of radioactive material. The regulations represent the *minimum* requirements for making legal shipments. Sometimes there is a need for more application-specific information on certain aspects of packaging or transportation operations, including taking voluntary actions that surpass the regulatory requirements.

Standards-setting bodies have provided the means to develop this more detailed type of information.[14] Standards related to radioactive materials transport are typically developed by accredited organizations such as the International Standards Organization (ISO), the American National Standards Institute (ANSI), the American Society of Mechanical Engineers (ASME), and the American Society of Testing and Materials (ASTM). The work of these organizations includes the development of consensus among its members concerning the need for a particular standard and the technical content of the standard. The organizations also ensure that the standards are periodically reviewed and revised as needed to keep them current.

This section of the *Handbook* provides summary information about the major standards settings organizations involved in radioactive materials transport topics. It also describes the interactions between the regulations and the standards.

2.3.1 Relationship Between Regulations and Standards

Within the United States a number of interactions influence the development of standards and regulations. DOT, the U.S. Competent Authority to the IAEA for the transport of radioactive materials, is the primary interface with the IAEA. As shown in Fig. 2.2, the interactions between NRC and DOT are covered by a Memorandum of Understanding, and DOT provides the focal point for U.S. interaction with the IAEA. These are the primary interactions between the regulatory agencies for the development of regulations. Of course, the affected public, NRC licensees, and DOE, which has some activities regulated by DOT, have opportunities to interact with both NRC and DOT. It is crucial to an effective regulatory process to allow ample opportunity for continuing dialogue between the regulators and the affected parties. For NRC and DOT rule-making, this process occurs through the publication of notices of Proposed Rule-makings in the *Federal Register* and the ensuing opportunity to provide comments to the agencies. The comments are considered and addressed (if appropriate) in the notice of a final rulemaking.

Standards developed under the auspices of ANSI are published and are available to the public at large. If NRC or DOT believes that a standard is appropriate for incorporation into a regulation, it may do so. In some cases, these agencies even encourage and directly support the development of specific standards. If a voluntary consensus standard can be developed on a topic that the agencies have found to be technically controversial, it simplifies implementing the requirements into regulation. It is generally easier for an agency to incorporate an ANSI standard than independently to develop and justify specific requirements. Consequently, Fig. 2.2 shows the relationship between NRC, DOT, and ANSI as a two-way relationship. Of course, many areas of regulation are not amenable to consensus standards, and vice versa.

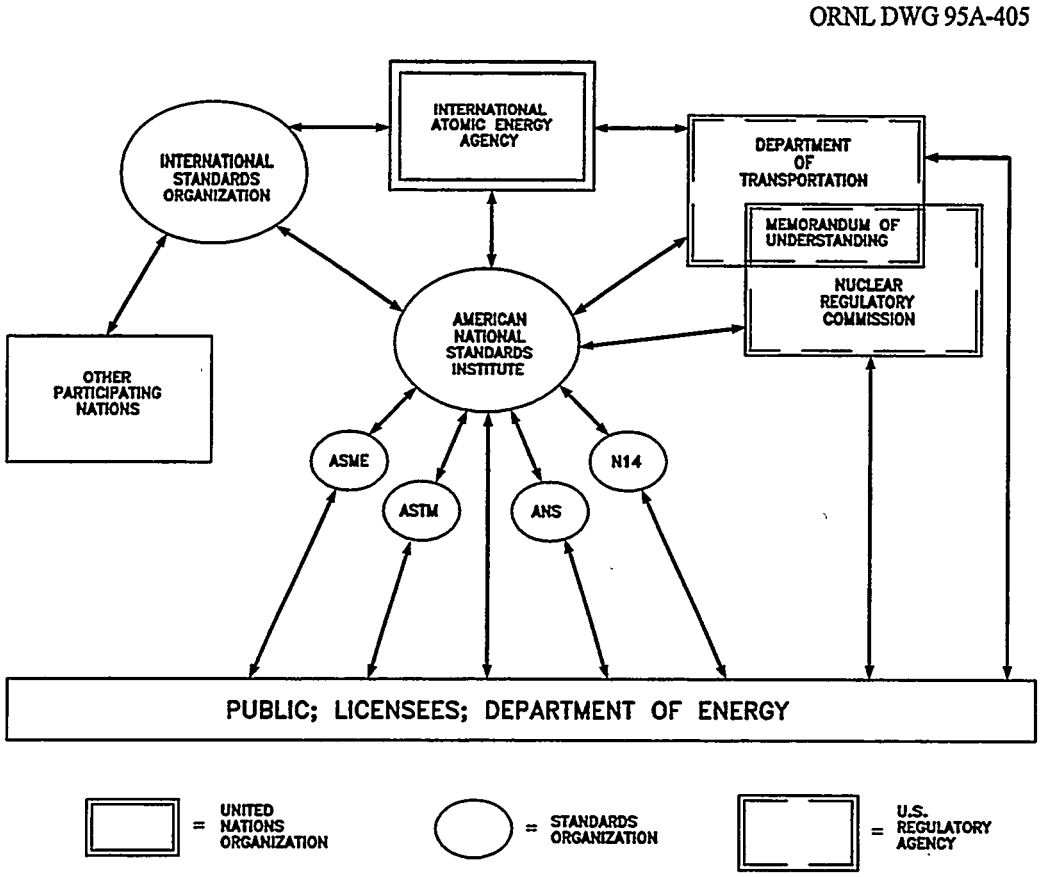

Fig. 2.2. Radioactive transportation regulations and standards: interactions of the United States.

As the primary U.S. interface to ISO, ANSI serves to channel input from all interested parties, including the public and the regulatory agencies. International standards are developed under the ISO charter with the participation of other national organizations and interested parties. These standards are then available for adoption by nations, international organizations, and the IAEA. In some cases, the ISO standards have been adopted by IAEA; in other cases, IAEA is interested in standards that are under development.

The interactions among the regulatory agencies (IAEA, NRC, and DOT) and the standards organizations (ISO and ANSI) are ongoing and can be viewed as circular. A requirement may be developed by any one of these and eventually be adopted or addressed by any of the others. As a result, package designers and users should maintain awareness of the activities in all of these organizations.

2.3.2 International Standards Organization

In parallel with and serving a similar role with the IAEA's advisory material provided in Safety Series No. 37, the ISO has been working to provide internationally accepted standards relating to specific issues dealing with the packaging and transport of radioactive materials. The ISO, which was originally formed in 1946 following an initiative of the U.N., formed Technical Committee 85 (TC 85), "Nuclear Energy," in 1956 (ref. 18). Under this committee, it has issued, or is

working on, standards in areas such as lifting equipment, handling and storage, transportation and packaging, criticality safety, and material control and inventory. All of these standards can be used to facilitate compliance with packaging and transportation regulatory requirements.

The ISO is the focal point for providing internationally accepted standards, and TC 85 has the responsibility to provide such standards that relate to the transport of radioactive material. The secretariat of TC 85 was originally with ANSI, but is now with the German Standards Institute (Deutsche Institut für Norman, or DIN).

TC 85 is divided into five subcommittees with secretariats in the United States, France, Sweden, Poland, and Germany. It coordinates its activities closely with many international organizations, and has frequently provided representatives to the IAEA's regulatory revision panel meetings.

Examples of ISO standards that help guide the international application of transport regulations include:

ISO 1677-1977	Sealed Radioactive Sources—General
ISO 2855-1976	Radioactive Materials—Packagings Test for Contents Leakage and Radiation Leakage
ISO 2919-1980	Sealed Radioactive Sources—Classification
ISO 4826-1979	Sealed Radioactive Sources—Leak Test Methods
ISO/TR 4815-1979	Sealed Radioactive Sources—Leak Test Method

The four standards dealing with sealed radioactive sources cited above may have applications in Type B package design if credit is taken in the package design for the containment capability of the sealed sources, and could apply to Type A Fissile (AF) packages if the radioactive contents are sealed capsules.

The ISO is currently working on other standards which could apply to the design and use of Type B and fissile material packages. Included are potential future standards on trunnions for spent fuel element transport packages; leakage testing on packages; principles of criticality safety in storing, handling, and processing fissile materials; and a qualification program for a long-term waste packaging for heat-generating radioactive waste.

2.3.3 American National Standards Institute

In the United States, ANSI (ref. 19) plays approximately the same role domestically that ISO plays internationally. In some cases, DOT or NRC may encourage and directly support the development of specific standards. It is always helpful for a package designer to be aware of the status of applicable consensus standards.

ANSI standards which are under development or available for application to the design and use of Type B and Fissile Material packages include:

ANSI N14.2	"Tiedowns for Transport of Fissile and Radioactive Containers Greater than On-Ton Truck Transport,"under development.
ANSI N14.5-1987	"Leakage Tests on Packages for Shipment," issued in 1987.
ANSI N14.6-1993	"Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials."
ANSI N14.23	"Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater than One Ton in Truck Transport," under development.
ANSI N14.24-1985	"Domestic Barge Transport for Highway Route Controlled Quantities of Radioactive Materials," issued in 1985 and in 1993.
ANSI N14.25	"Tiedowns for Rail Transport of Fissile and Radioactive Material Containers," under development.
ANSI N14.26	"Fabrication, Inspection, and Preventative Maintenance of Packaging for Radioactive Materials," under development.
ANSI N14.27-1986	"Carrier and Shipper Responsibilities and Emergency Response Procedures for Highway Transportation Accidents."
ANSI N14.30-1992	"Design, Fabrication, and Maintenance of Semi-Trailers Employed in the Transport of Weight-Concentrated Radioactive Loads."

A significant amount of information is exchanged between the ISO and the ANSI. For example, the ISO efforts on leakage rate measurements were initially based upon the ANSI N14.5 standard, and when the ISO standard is completed, it should significantly influence the revision of ANSI N14.5.

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3. Packaging Life Cycle and Certification

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3.1 INTRODUCTION

It is very important for package designers and owners and users of packagings to understand the typical life cycle of a packaging. There is considerably more to packaging than simply design and use. Package designers and owners and users of packagings assume broad responsibilities in many areas, particularly those related to quality assurance (QA). In recent years, industry, particularly the nuclear industry, has formalized the application of QA to package design, and to fabrication and operation of packagings and, in fact, has committed to a program of total quality management (TQM). This commitment has led to the recognition that every phase of a project must ensure that safety becomes a cornerstone in transport operations and that health and the environment are protected.

Whereas the regulations instruct the designer, owner, or user as to what requirements must be met to obtain package design certification, this *Handbook* provides an understanding of how to achieve these requirements. The life-cycle phases, as applied to a packaging, is shown in Table 3.1. The division of these phases may differ somewhat from package to package, but the phases shown are considered basic. It must be realized that these phases are not strictly sequential and that there may be considerable overlap of requirements for phases—particularly for package designs requiring regulatory certification.

Table 3.1. Packaging life-cycle phases and corresponding activities

Concept	Definition	Production	Operations	Shutdown
Determination Development	Package Design SARP Preparation and review Certification	Packaging Fabrication Acceptance	Shipping Maintenance Modifications Certification renewal	Decommissioning Decertification Disposal

No average time period has been identified for the various phases given in Table 3.1. This omission is intentional because many factors can affect the time required for any one phase. However, some estimates of time can be made for the various activities that occur during any one phase. For example, determination of the package type required for the service envisioned and acceptance of a packaging once it has been built could be measured in weeks. Development of

the SARP and actual fabrication of the hardware could be measured in months. SARP review and certification could be measured in years. The complexity of the packaging can affect the time required to complete the prototype demonstration tests and product tests. If a packaging is used widely, its design may remain as a certified design for a long period of time.

A general rule for planning a new packaging development project is that the time required from concept to operation is about 2 to 7 years, depending upon the complexity and the methods used to evaluate the package, as reported in the SARP. The associated costs could vary up to $10 million, depending upon the packaging size, type, material of construction, etc. A large spent fuel cask may cost even more.

The packaging activities mentioned in Table 3.1 are discussed in the following subsections.

3.2 PACKAGE DETERMINATION

Transporting radioactive material over public highways using packages designed for the purpose is highly regulated by federal and international agencies. Anyone who knowingly or willfully violates provisions of the regulation is subject to civil and criminal penalties. Therefore, it is essential that the package design be selected correctly and the packagings be used properly. In general, regulatory guidelines for packaging design ensures that radioactive material dispersion, radiation, and decay heat are limited to acceptable national standards and that a nuclear criticality event is prevented from occurring.

Procedures to ensure safe packaging for transporting radioactive materials include (1) categorizing the materials according to activity levels and material form and (2) requiring the preparation and use of packaging appropriate for the type and quantity of material. The choice of packages is based on the form, quantity, and activity that is related to the radioactivity of the material to be shipped.

As discussed in Sects. 2.2.2 and 2.2.3, a number of factors can affect the type of package design required for transport of radioactive material. If a material has a specific activity less than 70 Bq/g (0.002 μCi/g), it is not classed as radioactive, and the packaging and operational requirements of 49 CFR 173 and 10 CFR 71 do not apply.

The type of package required for radioactive material can range from excepted, industrial, Type A to Type B. Additional design requirements will apply if the material is classified as fissile. The factors which can influence package type include the amount of radioactive material in terms of the specific radionuclides involved and their individual activities, the form of the material (solid, liquid or gas), whether the material is in special form or is other than special form (also denoted in the regulations as "normal form"), the fissile nature of the material, the radiation levels posed by the material, and possibly its specific activity level.

The A_1 and A_2 values (see Sect. 2.2.2.2) are critical to defining package design requirements. The regulations clearly define methods for determining the appropriate values for A_1 and A_2, for individual radionuclides and mixtures of radionuclides (see 49 CFR 173.433-435 and 10 CRR 71, Appendix A).

For a material to qualify as LSA material, it either must be an ore containing only naturally occurring radionuclides, uranium concentrates of such ores, solid unirradiated natural or depleted

uranium or natural thorium or their solid or liquid compounds or mixtures, or Class 7 materials other than fissile material for which the A_2 value is unlimited; or materials which satisfy well-defined limitations on the specific activity of the material and whether it is uniformly distributed (173.403). For an object to qualify as SCO, it must satisfy clearly defined requirements on fixed and non-fixed contamination on both accessible and inaccessible surfaces (173.403). Finally, for a material or object to qualify as LSA or SCO respectively, it must also satisfy a requirement that the external dose rate must not exceed an external radiation level of 10 mSv/h (1 rem/h) at 3 m from the unshielded material [173.427(a)(1)].

In general, the total amount of material to be shipped in a packaging will determine which package design requirements must be used. Transporting small amounts may reduce the requirements on the package, but this may be offset by requiring either more packages or more shipments to transport all of the radioactive material. Thus, the package designer and user of packagings must determine an optimum balance between transportation and packaging costs.

Once the type, form, and quantity of the radioactive material has been determined, the package can be selected. The first choice could be to use an existing certified package, if one can be found that matches (or almost matches) the requirements of the shipment to be made. A number of packages have been certified by the DOT, NRC, and DOE and by competent authorities from other countries. In the United States those package designs that have been certified for shipment from, or to, NRC licensees and DOE facilities are listed in NUREG-0383 (ref. 1). In addition, DOE certifies package designs for use by the Department. A potential user must petition the appropriate certifying authority for authorization to use a particular package. If it is available, a user must stay within all limitations identified in the certificate, and meet QA requirements, operational constraints, and DOT and NRC regulations. The QA and operational procedures must be in place at the shipping facility before the packagings are used.

The second choice could be to use an existing package that had been certified at one time, but whose certification had been allowed to lapse. However, this may require a complete recertification if the QA records have not been kept or if the package design could not meet current standards.

The third choice could be to petition DOT, NRC, or DOE for authorization to allow a different material or quantity of material in a currently certified package. This option may reduce some of the safety documentation that would otherwise have to be generated, but it might save the effort of designing and certifying a new package.

A fourth choice could be to request a DOT or an NRC exemption or a DOE alternative. Such an exemption would be limited to one-time-only shipments or for shipments of national security and would still require the package to be approved by the appropriate federal agency. The user would be required to demonstrate the various levels of administrative controls required to ship safely. The exemptions and alternatives are not easily obtained and probably should not be included in one's plans. These are generally held for very special circumstances and require a very high level of authority (DOE Secretary/NRC Commissioner) signature.

If none of the options for package selection is satisfactory, then a new package design would become the only recourse.

3.3 PACKAGE DESIGN AND DEVELOPMENT REQUIREMENTS

If no options are available to use existing packagings, then the applicant must develop a new package design to fulfill the given mission. The term "applicant" in this context refers to the organization that will ultimately apply to the competent authority for the certification of the new package design. Before developing the package design, the applicant must understand all of the packaging requirements, including (1) federal, state, and local regulations and (2) his own requirements. These requirements are often extended or supplemented at the applicant's facility to meet more stringent safety requirements, such as operating constraints, material limits, or personnel safety. Also, any operation that may be unique to the facility and material to be shipped may add special requirements to the package design. These should be documented in a system requirements document to include all functional and operational requirements. The system requirements document should be baselined within the configuration management system of the packaging and shipping program and should be revised to reflect changes as they occur.

The system requirements should be prepared by project management and provided to the project participants as the basis for their work. The number of requirements should be minimized to allow the designers and analysts as much freedom as possible. The systems requirements should include the following, as a minimum:

1. *Material characteristics of the source.* Material characteristics of the source should include any parameter that may influence design decisions. This description of the source should include important assumptions about the source and be documented through drawings, data sheets, procurement certificates, dimensions, material log books, operating time in a reactor (if spent fuel is the source) or any first-hand accounts of the data. The data should be signed and dated by a witness. This information should be kept in the records system for the life of the certified package design. Some material parameters of interest are

   - chemical form,
   - amount and type of radioactivity,
   - physical form,
   - mass,
   - density,
   - volume, and
   - time out of the reactor (if spent fuel).

2. *Transportation campaign requirements.* Criticality, shielding, and thermal evaluations of the source and the packaging; federal shipping restrictions; and the campaign information will influence the maximum quantity of material allowed per package. This information will also help the designers to decide between developing a reusable or a single-use package. Some examples of shipping campaign information that should be identified include:

   - quantity of material to be shipped,
   - mode of transport to be used,
   - proposed shipping schedule, and
   - number of shipments to be made.

3. *Special operational requirements.* Expected material and packaging operations may require particular design features on the packaging or impose special operational procedures or more stringent safety standards. Some special requirements that might be considered include:

   - absorption of any liquids that might be transported,
   - ullage requirements,
   - cost of the packaging and attendant operations,
   - handling,
   - inner packagings needed,
   - maintenance of the packaging,
   - moderation of any fissile material that is carried,
   - QA,
   - radiation shielding,
   - security measures during transport, and
   - tiedown.

4. *Verification plan.* It is prudent to assess the activities needed to verify that all federal regulations and system requirements have been met before the designer and analyst prepare a verification plan. This assessment is based on the plan and consists of a team effort of management, designers, analysts, test engineers, operators, quality, and procurement. Each must examine the federal regulations and system requirements and prepare a statement of how the requirements will be met by the package. The core of a verification plan could consist of a matrix that identifies each appropriate federal regulation with its applicable requirement, who is responsible, when the verification activity should be completed, and how the requirement will be achieved. The verification plan should be baselined within the configuration management system. It should be revised to reflect changes as the project proceeds. It is important that this plan be completed early to avoid overlooking any system requirements.

Integral with the federal regulations are quality requirements. Total quality management (TQM) is essential from the time packaging ideas are conceived until the packaging passes through its complete life cycle. Each federal agency espouses its own brand of QA (see the following); however, a close look at these will show that they are all basically the same:

Federal Agency	Quality Regulations
DOT	49 CFR 173
NRC	10 CFR 71, Subpart H/NQA (ref. 2)
DOE	Order 5700.6c (ref. 3)/QC-1 (ref. 4)
IAEA	ISO 9000 Safety Series No. 6 (ref. 5)

QA plans should address all of the elements applicable to the packaging development effort. This plan should identify what procedures will be used to achieve the quality requirements and, if possible, should be submitted to the federal agency before initiation of the package design and development effort. This allows the certifying official to assess the adequacy of the packaging

program and to ensure that the package will meet all applicable federal regulations. QA is discussed in Chapter 11.

3.4 PACKAGE DESIGN AND DEMONSTRATION

The design and demonstration phase of the packaging life cycle consists of very important activities that translate the system requirements, verification plan, and QA plan into a package design that satisfies the customer. The designer should develop criteria based on all requirements. This document should describe the functional and physical requirements of the packaging, define the developmental and engineering evaluations necessary to establish the design basis, and provide guidelines to evaluate, demonstrate, or substantiate the fulfillment of the design objectives. As a minimum the following information should be contained in the design criteria:

- the scope and general description of the proposed packaging;
- the method of accomplishing the project;
- applicable codes, standards, and specifications and requirements for verification of design calculations and any software used to perform design calculations;
- reference documents, drawings, and sketches;
- packaging test plan requirements;
- design parameters and design requirements;
- special requirements and provisions (quality, environmental, safety, security, health physics, nuclear safety, reliability, maintainability, and handling); and
- appropriate cost and schedule information.

From the design criteria, the designers will prepare the drawings and equipment, and material specifications. The procurement instructions and specifications personnel will use the information from these documents for items purchased to perform verification activities and, subsequently, to develop the packaging.

The selected method of verification may include testing, analysis, or a combination of both. A detailed verification plan should be developed that addresses the methods of achieving the packaging requirements. This plan may be a further refinement of the verification activity, which was discussed in Sect. 3.3. Either way, this plan should include the specific test or analysis plan requirements. The test or analysis plan should then be followed up with a description of the detailed tests or analytical procedures. The test or analysis acceptance or rejection criteria should be stated in the plans and procedures.

The project should conduct design reviews several times during the design phase. The information that follows should be covered in each review:

- design criteria,
- design drawings,
- equipment specifications,
- appropriate arterial data sheets,
- test verification plan and procedures,
- analysis verification plan and procedures,
- all design calculations,
- all analytical results,

- all tests results, and
- all tests reports.

The last four items listed above should be reviewed by the project management team in a timely manner. The data should demonstrate compliance with requirements. The results should be presented in calculation data sheets; tests results data forms, photographs, and videos; and computer-generated graphs, plots, and charts. The test results should be well documented in a test report that will be used in the SARP directly or by reference. The information noted herein should be baselined within the configuration management system documents which should be revised as necessary.

The ultimate goal of the design and verification effort is to obtain a package design certificate to ship the material identified. Achieving this goal will require the preparation of a SARP and its subsequent review and acceptance by the certifying official. It is important that the certifying official and his or her agents be cognizant of the design and verification results as early as possible to reduce the possibility of problems that may occur in the certification process. Thus, all baseline documents should be forwarded to the certifying official. Furthermore, the certifying official and/or his or her agents may be invited to participate in formal design reviews as appropriate.

The culmination of the design and verification phase will lead to the preparation of the SARP. The SARP, which is discussed in Sect. 3.5, should clearly demonstrate that all of the requirements have been met with a clear margin of safety to protect the public, the workers, and the environment.

3.5 SARP PREPARATION

The SARP is the repository of all information that demonstrates that a particular package meets all imposed requirements. It is very important that the SARP provide convincing information that demonstrates to the certifying official that the package meets all of the requirements with a clear margin of safety to protect the public, the workers, and the environment.

The basis for the safety document comes from the requirements given in 49 CFR Parts 171–180 and 10 CFR Part 71. The package is also subject to other parts of Title 10 (e.g., Parts 20, 21, 30, 39, 40, 70, and 73). Requirements for packaging for international shipments are provided by IAEA Safety Series No. 6 (ref. 6) and 9 (ref. 7). Packaging certified through DOE must adhere to the safety requirements of DOE Order 460.1 (ref. 8) or 5610.12 (ref. 9). Both of these orders refer to the NRC, DOT, and IAEA requirements where applicable. In cases where these are not applicable, DOE provides additional guidance for transportation safety risk assessments.[10]

The information in the SARP is unique to the packaging and content; however, the format of the SARP has been provided by NRC in the *Regulatory Guides 7.9* (ref. 11) and *7.10* (ref. 12), which were established over the years from experience in order to standardize the SARP preparation and review process. Even though these are only guides (not requirements), it is strongly recommended that they be followed with no variation. By using these guides, the SARP is reviewed more easily which, in turn, should ultimately reduce the certification cycle.

A number of other documents published by the NRC or other authoritative bodies may be used by the applicant to prepare of the SARP. These include the following:

- NUREG/CR-3854, *Fabrication Criteria for Shipping Containers*,[13]
- NUREG/CR-5717, *Packaging Supplier Inspection Guide*,[14]
- NUREG/CR 4775, *Guide for Preparing Operating Procedures*,[25]
- ANSI N 14.5, *Leakage tests on packages for shipment*.[26]

The certifying officials or their agents review the SARP following the guidance provided in the reviewer's guide [UCID-21210 (ref. 17)]. In a reverse process, the SARP can be prepared to provide at least the information requested in the reviewer's guide. This should result in providing at least the minimum information required by the certifying official, but more may be requested. However, this guide does give the applicant a clear indication of the direction of the reviewer's intentions and the depth of demonstration needed for each of the SARP sections.

It is also prudent for the applicant to approach the certifying official early with basic information about the packaging and SARP. This may include a brief meeting of the participants to establish lines of communications, to present the SARP preparation schedules, and to discuss technical issues and potential trouble areas. The applicant should not expect to receive design guidance or explicit directions for analysis. The applicant may receive information taken from other SARP reviews, concerning appropriate analytical techniques and documentation which is often helpful to the reviewers to reduce their evaluation cycle time.

Regulatory Guide 7.9 outlines what information goes into the SARP. The following are additional refinements that reduce the burden on the reviewers and improve the likelihood of receiving a swift review:

- a complete set of design drawings with tolerances and weld symbols;
- a complete set of material specifications with certification and testing;
- an exploded view of the packaging with contents; and
- operating, acceptance testing, maintenance, and QA requirements.

The SARP preparation may take many different routes. One that has proven to be successful is to develop a team of highly motivated professionals who are expert in their field. It is recommended that everyone on the SARP team be experienced in preparing SARPs; be knowledgeable about transportation and packaging regulations; be trained in QA procedures; and be knowledgeable in scientific, business, and legal communications. A description of a proposed SARP team and the respective responsibilities of the members follow:

Coordinator. The coordinator should be knowledgeable on all SARP subject matter. The coordinator must be able to blend all of the SARP sections into a single, cohesive document. The coordinator is responsible for identifying and securing all personnel to prepare the SARP and to manage the SARP effort. The coordinator should establish SARP policy, arbitrate technical differences, and be the focus for text preparation and publication. The coordinator should prepare the sections on introduction, operations, acceptance testing, maintenance, and QA with support from the respective facility staff. The coordinator should lead a subteam consisting of the technical editor, electronic publisher, and graphic designer.

Designer. The designer is the responsible engineer for the design of the package. The designer must understand the American Society of Mechanical Engineers (ASME) Boiler and Pressure

Vessel Code, material standards, equipment specifications, weld standards, test procedures, and transportation and packaging regulations. Also, the designer must understand the requirements for containment, radiation shielding, and nuclear criticality. The designer should lead a subteam consisting of a computer-aided drafting operator, metallurgist, and test engineer. The designer should also be responsible for evaluating the containment criteria for the package because containment is directly related to closure, such as welds, seals, flanges, and valves.

Structural analyst. The structural analyst should be well trained in statistics, dynamics, fatigue, fracture mechanics, and nonlinear and large displacement analysis techniques. This training usually requires experience with finite-element computer programs. The analyst should also be knowledgeable in material properties at low and high temperatures.

Thermal analyst. The thermal analyst should have experience in high- and low-temperature structures that are made of steel, brittle materials, molded products, combustible materials, and radioisotopes. These skills generally require a knowledge of finite-difference and finite-element computer programs and experience with material properties at low and high temperatures. The thermal analyst should work closely with the structural analyst and designer to optimize model development and information gathering. The thermal analyst should work with the radiation shielding analyst to evaluate the radioactive source decay heat.

Radiation-shielding analyst. The radiation-shielding analyst should be very experienced with radiation shielding of the radioactive material. This requires a knowledge of source calculation methodology, which is generally evaluated using computer programs. The analyst should know transportation and packaging regulations for minimum dose rates and for implementing "as-low-as-reasonably-achievable" (ALARA) principles for radiation exposure. It is important that the analyst review other SARPs for techniques and data-reporting requirements.

Nuclear criticality analyst (fissile material only). The nuclear criticality analyst should be very experienced with criticality analysis techniques and transportation and packaging regulations. Such experience often requires several years of work with complicated computer programs analyzing materials similar to those which are to be carried in the package.

Operations and maintenance. Facility personnel who will be responsible for the packaging operations and maintenance should provide the respective requirements to the coordinator for incorporation into the SARP and should review the pertinent sections of the SARP. The coordinator should also contact persons from other facilities who may wish to use the packaging in order to reduce operational and maintenance requirements to a minimum and minimize operational problems.

Procurement. Procurement should interact with the designer to procure items he or she has specified. Procurement should prepare the various requisitions, select vendors, and participate in acceptance reviews. For items made in-house, a similar function may be performed by shop schedulers, production control, or other manufacturing organizations. Procurement should provide to the coordinator the procurement documents to incorporate the acceptance criteria into the SARP.

QA. The QA coordinator should have experience with packaging regulations, facility QA plans, vendor surveillance, auditing, and self-assessments. The quality department should provide the quality requirements to the coordinator who would incorporate them into the quality assurance criteria and should prepare or support the preparation of the packaging QA plan and procedures.

All of the SARP team previously identified should receive training in transportation and packaging regulations, design techniques, review procedures, technical writing, and QA. A number of courses sponsored by various federal agencies provide basic and advanced training in SARP-related technologies.

Because the SARP is a lengthy, technical document, it is important that it be assembled with great care. Experience has shown that in the final week before submittal to the competent authorities, it will be important to plan an in-depth review of each section of the report. A large room that can accommodate a variety of people who have helped generate the document should be reserved for this activity. It is at this meeting that last-minute problems will be identified and worked out; the problems will cross-cut all technical and nontechnical activities and will involve not only the various analysts and the coordinator, but also those involved with word processing, reproduction, editing, and graphics. A check list should be generated of the items to be covered, and all contributors should be prepared to be called on at any time during this exercise. Management should be notified if signatures and transmittal letters are required. Once the first draft of the SARP has been assembled, the designer, the coordinator, each analyst, and the editor should review it.

The draft SARP should be reviewed by an independent team with SARP experience. This team should be comprised of analysts who were not involved in the original preparation of the SARP. For example, a qualified subcontractor might be appropriate to review specific sections. This review should be limited to 1 to 2 months and consist of a comparison of the SARP with the information requested by *Regulatory Guide 7.9* and *7.10*, a review of the calculations and results using, to the greatest extent possible, simple confirmatory calculation. The results of this effort should be documented in an independent review report and then should be reviewed by each of the appropriate analysts, the designer, and the coordinator. Corrections to the SARP should be made, and the SARP should be readied for publication. The record of the independent review and information as to its subsequent effect on the SARP should be maintained in the packaging file, where it would remain available for review by the appropriate federal agency.

All sources of information that were referenced but not incorporated into the SARP should be collected and delivered, along with the SARP to the appropriate federal agency. Sources such as reference books, computer manuals, and national standards should not be included. Published reports (e.g., results, plans, procedures, obscure journals, and out-of-print text) should be provided in as much as is possible. This effort may require a lengthy request by the applicant for the rights to reprint copyrighted material.

3.6 SARP REVIEW AND CERTIFICATION

In theory, the process to receive a certificate for a package design from the federal agencies is basically simple; however, in practice, the process is quite often lengthy and subject to an ever more rigorous application of the regulations. This longer review is a product of the increased safety consciousness being applied to all hazardous activities: What was acceptable yesterday may not necessarily be acceptable today. This fundamental credo is why everyone in the SARP process must be continually retrained in the regulations. The federal agency reviewers will take considerable time to review the SARP and all accompanying information before the package design is considered for certification.

The review team prepares an initial set of questions, which are called Q zeros, or Q_0s, if they determine that the SARP does not include fundamental information or is not prepared in the recommended format. They may also request additional data for their evaluations. The Q_0s, are provided within 1 or 2 months, subject to the reviewers schedule on other reviews. These Q_0s, should be evaluated by the SARP team and the appropriate member resolve the questions. The coordinator should prepare a document that details the reviewer's questions and the applicant's response. The document should be provided along with proposed changes to the SARP to the reviewers. The mechanism for the SARP changes should be discussed with the reviewers and the certifying officials. Some certifying officials want only the revised pages, while others want to concur with the proposed changes before the SARP pages are actually changed. The latter tends to reduce the amount of revisions to the SARP and reduces the question and answer cycle.

The SARP team must be prepared to change the SARP to accommodate the certifying official and his or her team of reviewers. The certifying official is relying on the reviewers to provide recommendations for certification; therefore, repeated arguments will generally not resolve findings. The applicant must be able to objectively present the facts. Although the certifying officials may arbitrate when agreement cannot be reached, this is not a desirable situation. The applicant and the reviewers should be able to come to an agreement, the outcome of which will provide the proper degree of safety required by the regulations.

The reviewers continue to review the SARP while the applicant prepares answers to the Q_0s. At some time interval agreed to by all parties, Q_1s are submitted to the applicant. Although the applicant should respond to these in a similar manner as the Q_0s, the questions will be more technical in nature and may require significant re-analysis and changes to the SARP. Experience has shown that if the SARP was poorly prepared, changes to the package design may be required to complete the certifying process. Such changes are generally very expensive and time consuming at this stage. The use of experienced personnel in the preparation of the SARP is the best way to avoid this problem.

The question-and-response process continues until the certifying official and his or her reviewers are satisfied that the packaging design and SARP adhere to all federal regulations. The certifying official prepares the safety evaluation report (SER) that documents this process.

The certifying official prepares the certificate of compliance (COC) or off-site transportation certificate (OTC). The packaging and content descriptions and operational restrictions are taken from the SARP and included in the COC. If the review process uncovers some problems, the certifying official may choose to add some restrictions to the COC. The COC is signed by the certifying official, a certificate number is assigned, and an expiration date is determined. The COC is usually assigned an expiration date 5 years from the approval date; however, this expiration date is at the discretion of the certifying official.

The applicant must be authorized by the federal agency to use the package design, which may require a federal agency audit of the applicant's facilities, QA plans and procedures, and training.

In a logical sequence, the applicant receives the COC before the packaging is built. This step ensures that the QA elements are correctly implemented for the fabrication and assembly. It also reduces the potential costs; however, it may impact project schedules. An applicant must decide what risk is acceptable to its project.

3.7 PACKAGING FABRICATION

Methods for fabricating the packaging will usually depend on the quantity of packagings to be built, the schedule for completion, and funding. Complicated designs may require intricate fabrication techniques. This section will discuss how the fabrication activity relates to the packaging life cycle.

The applicant's QA plan should detail its approach to the control of purchased items and services. Corporate and personnel evaluations are performed before a vendor is selected to fabricate a packaging. The applicant should complete the following activities regarding the vendors:

- evaluate and select packaging component vendors,
- establish controls to be imposed on these vendors,
- perform audits at vendors' facilities (as required), and
- establish conditions for the final receiving inspection at applicant's facility.

Each vendor's QA plan should specify the procedures for performing inspections and preparing the necessary certification papers on the items to be delivered. The certification papers should be delivered with the item(s) in question and should become part of the applicant's QA record. In some instances, if a part is critical to an important function (e.g., containment) of the package, the applicant may specify that he/she, rather than the vendor, perform the acceptance inspection and generate the certification papers; these also should become part of the QA record. If those critical components are manufactured in-house, then similar inspection and certification steps should be identified and conducted for those items.

The applicant should conduct a first-article evaluation for large-quantity purchases. Each evaluation may include a repeat of the demonstration tests specified in 10 CFR 71.71 and 71.73. Although this is not required, it does establish the ability of the design to be manufactured in quantity by a vendor.

Package fabrication is discussed in more detail in Chapter 13.

3.8 OPERATIONS

The shipper and receiver should conduct readiness reviews with the packaging before its use. This review provides on-the-job training for the operators and handlers and for handling-equipment shake-down. The operating procedures (which may require nuclear criticality safety approvals) should be thoroughly evaluated to ensure that all steps are appropriate. Furthermore, each packaging user should conduct training exercises, identifying potential abnormal events and any necessary corrective measures. These training exercises may involve all facility organizations, such as hazards response, nuclear criticality safety, contamination control, health physics, safeguards, security, fire department, medical staff, and plant management. It may be useful to involve off-site organizations. Such organizations could include city or county emergency response teams and the media, including radio and television personalities. The carrier could also participate in these exercises.

The operating procedures should be as simple and straightforward as possible. Everything related to the package operations should be documented to prevent the loss of valuable information. Video tapes of operations and training exercises are excellent methods of documentation.

Users should periodically review their operating procedures. New, improved ways to reduce radioactive exposure to the workers and the public should be incorporated into the procedures as these methods become available. Consignors should be surveyed periodically to determine if the packaging operations are being followed and if new ways are needed to enhance safety or quality.

The applicant must notify the proper facility managers and the appropriate federal agencies immediately if an unusual event occurs during packaging operations (loading, transport, unloading, storage) or if the packaging design becomes suspect. If the packaging design is questioned, then the applicant must issue a stop-use notice to anyone using the affected packaging.

Packaging operations are discussed in more detail in Chapter 12.

3.9 MAINTENANCE

All packagings shall be maintained to the level of certification required by the drawings, specifications, SARP, and COC. The packaging owner or user shall prepare refurbishment procedures consistent with the requirements outline in the SARP. Persons performing the refurbishment shall be trained accordingly.

Refurbishment may follow a replace-or-repair philosophy. If packaging parts are replaced, then only certified parts may be used. If packaging parts are repaired, then those parts must be recertified according to the drawings, the specifications, and the SARP.

The packaging owner or user must recognize the potential problems associated with replacing damaged parts with bogus parts. Of particular concern would be the use of noncertified bolts, nuts, and seals because these components are available from many places and may have the proper markings so as to appear proper. In addition, the refurbishment program shall consider the shelf life of life-limited parts, such as seals. These should be examined before use and discarded if the data listed on a part indicate that it has exceeded its shelf life.

Package maintenance is discussed in more detail in Chapter 14.

3.10 PACKAGING MODIFICATION

If a packaging needs to be modified, the owner must evaluate the proposed modification and present the facts to the certifying official. These proposed modifications to the packaging must be authorized by the certifying official, in terms of changes to the package design, before they are completed and before the packagings may be used. The COC must be revised and appropriate training conducted to make the modifications and to use the modified packagings.

There are a number of reasons why modifications to a packaging may occur. For whatever reason, some part of the packaging must be modified. The following steps, as a minimum, should be followed in processing a modification to the packaging.

1. *Modification.* Packaging modification can occur, for example, because of unacceptable damage during normal use, supply of certified parts or material is no longer available, newly discovered material information that indicates the packaging is no longer acceptable, or new information on materials may disclose hazardous or toxic affects previously unknown. An

investigation should be conducted to discover the root cause of the need to modify the packaging design. This investigation should be well documented and conclude with recommendations for action. If there are safety problems associated with the modification, the certifying official should be notified immediately. The certifying official may recommend a hold on all activities associated with the respective packagings (or similar packagings) until all the facts are discovered.

2. *Design.* The owner shall prepare drawing and specification changes to reflect the proposed modifications to the package design. The drawing and specification revisions shall follow the owner's configuration management procedures. The documents shall not be issued for use until the certifying official authorizes such modifications.

3. *Changes to SARP.* The proposed modifications to the package design shall be evaluated and reported in a manner like that originally conducted for the package design and reported in the SARP. The modification may result in the need for new analyses or tests. All of the impacts of the modifications shall be determined (e.g., changes to a new material may result in changes to weights, material compatibility, radiation shielding, criticality moderation, handling, stresses, dynamics, and thermal response). Revisions (changes to pages) to the SARP shall be prepared and submitted to the certifying official. These revisions will be evaluated commensurate with their importance to safety. If the modification is simple and does not change a safety class item, then the review by the certifying official may not require an extensive analysis. However, if a safety class item is modified, the certifying official will probably require a thorough review of the changes, or perhaps a review of the entire SARP. Following the discussions and review, the certifying official shall revise the SER and the COC appropriately.

4. *Field Change.* When the proposed modifications are authorized by the certifying official and the revised CoC is obtained, the owner of the packages may perform the proposed modifications. Old parts shall be discarded and removed from service to prevent them from being used by mistake. The modified packages shall be marked to identify the change, as appropriate. The modified packages shall be inspected and must pass the appropriate certification tests, such as radiographs, surface smoothness, dimensional, pressure tests, leak tests, or thermal test. When the package design is recertified, the external marking shall be revised to indicate the date of certification. These packagings are now certified for use.

5. *Repair records.* The records of the modification shall be entered into the appropriate files for safe-keeping. The records for all packagings that were modified shall be amended.

Package modifications are discussed in more detail in Chapter 14.

3.11 MAINTAINING CERTIFICATION

The original package certificate is usually authorized for 5 years. The owner of the package design shall submit a renewal request with about 3 years remaining on the certificate. This allows 1 year for the revision to the SARP, 1.5 year for review, and 0.5 year for approval. The time intervals may be overestimated if there are no proposed design changes, no regulatory rules changes, and no problems encountered in using the package. Each one of these present unique problems that must be resolved.

The significance of a design change is proportional to the safety class of the item being modified. If the change is related to the containment boundary, radiation shielding, or nuclear criticality, the respective evaluations in the SARP must be done again to account for the proposed change. This change would be incorporated throughout the SARP, which would probably be subjected to a complete review (much like the original review) by the certifying official. Thus, a design change to an important safety-class item might take several years for renewal. A change to an item that is not a safety-class item would probably not result in a full SARP review. In either case, the change would be assessed, and the SER revised accordingly.

If new or revised rules are implemented before the expiration date of the certificate, then the package design may be reassessed and the SARP revised accordingly. The extent of the rule changes will dictate the level of SARP revision and review. An example of a rule that could have significant impact on the packaging is the NRC change (effective April 1, 1996) to require certain light-weight, low-density package designs with greater than 1000 A_2s to survive an additional dynamic crush test. The hypothetical test sequence for these packages requires a 9-m free drop onto an unyielding surface, followed by a 9-m free drop of a 500-kg mass onto the previously damaged package. The remaining tests are the same.

A listing of all packagings serial numbers that were used under the certificate and their refurbishment history should be available for review by the certifying official. If the refurbishment history indicates an abnormal repair record of specific packaging components, exists then changes may be warranted, calling for more robust components. Like the design changes described above, the level of the change to the safety class items will determine the level of change to the SARP and probably require a review by the certifying official.

3.12 USEFUL LIFE OF PACKAGE

When a radioactive material package design is no longer certified or a packaging is no longer usable for its intended purpose, it may be used to contain and store other hazardous or radioactive materials or be decommissioned and disposed. A Type B package may be downgraded to a Type A package. A Type A package may be downgraded to a 1P or an excepted-quantity package. Any of the packages may be downgraded to a strong, tight container or industrial package. If such packagings are downgraded, the previous markings should be removed to prevent confusion with its use. Also, the records should be amended to reflect this change in status.

3.13 DECOMMISSIONING AND DECERTIFICATION

If the owner of a packaging no longer needs the certified package design, the owner must notify the proper federal agency of intent to decertify that design. If there are other users, the federal agency may choose to transfer the COC ownership to another user. The initial owner may transfer all packagings applicable to that certificate to the new owner.

When a federal agency transmits a notification of decertification, all users must record the decertification and ensure that they are not used for off-site shipments. Each affected function at the facility should be notified of the decertification, and the appropriate measures should be taken to ensure that it is no longer used for off-site shipments.

The user should segregate all the decertified packagings and remove the certification markings and affix the appropriate empty label and other hazardous material labels. Considering the purpose for the decertification, the packagings may be used for other contents, especially for on-site shipments under certain conditions or as strong-tight container. If the packagings are no longer needed, they should be disposed as waste material characterized to the appropriate hazardous waste material. The packagings should be decontaminated first to remove the surface contamination to the lowest level practicable.

The certified package data base should be updated to reflect the latest information on those packages that were downgraded or destroyed. It is important to note that the records for a decertified package must be kept for at least 3 years past the date of destruction.

3.14 DISPOSAL

Decertified packages must ultimately be disposed of in a manner consistent with the national policy for radioactive waste or hazardous material disposal.

The level of surface contamination and imbedded radionuclides should be determined. The surface contamination should be removed to the lowest level practicable. If the contamination level remains high, the packaging may need to be disposed of in one of the national radioactive burial grounds.

3.15 REFERENCES

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4. Testing

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4.1 INTRODUCTION

The regulations contained in 10 CFR Part 71 state that a new package design may be certified via analysis or testing, and a combination of both analysis and testing is often used. Analysis methodologies provide the ability to evaluate package performance in a wide variety of conditions by adjusting modeling parameters. Testing is somewhat more limited since evaluation of even small variations in conditions may require a complete retest. Testing is preferred for conditions that are difficult to analyze, such as dynamic impact behavior and leakage of seals. Confirmatory tests are often used to validate analyses. Each package design may require a different strategy of testing combined with analyses to obtain the optimum combination.

Tests (e.g., engineering tests, certification tests, package acceptance tests, and maintenance tests) may be employed to serve a variety of purposes.

- Engineering tests are performed to gather knowledge about the behavior of a system or component. Engineering tests may be (1) exploratory, where a component's behavior is not known prior to the test, or (2) confirmatory, where verification measurements of expected behavior are sought.

- Certification tests are performed with the express purpose of obtaining data which will be included or referenced in a Safety Analysis Report for Packaging (SARP). Certification tests should be performed under a 10 CFR 71-approved quality assurance (QA) program or U.S. Department of Energy (DOE) equivalent.

- Acceptance tests are performed before a package is placed into service; these tests are usually described in the SARP.

- Maintenance tests are scheduled for annual or other periodic maintenance of packages.

Tests to confirm normal operating conditions may involve an examination of the heat transfer capability of a package to verify that temperature limits for the contents of the package are not exceeded. Seal leakage tests are performed as acceptance tests for many packages.

Other potential tests for new packages include verification of shielding capability through gamma scans or radiography. Because of the difficulty of handling neutron sources of sufficient strength to make shield testing practicable, the presence of neutron shielding is often verified through sampling of the neutron shielding material during installation. Blackness testing for criticality

safety components is sometimes performed by conducting a neutron transmission test or a neutron radiograph.

Packaging components that are purchased from commercial sources such as seals, valves, coatings and paints, and shielding materials may be tested by the product vendor. However, the responsibility for demonstrating the adequate performance of the purchased component remains with the cask design organization if the component serves a safety function in the package. Current practice calls for testing package seals over the range of expected temperatures (-40°C upwards) because the seal performance is dependent upon its proper installation in the package. Valves that are enclosed by valve port covers which provide containment do not require testing over the range of conditions of transport packages since the valves are used only during handling and do not serve a safety function during transport.

Hypothetical accident tests which are commonly performed for packages include:

- Nine-meter free drop—a free drop of 9 m (30 ft) onto an essentially unyielding surface in the worst possible orientation of the package. Tests are often performed on the package closure end, corner, and side to establish the worst condition. Oblique (slapdown) drops may be required to evaluate secondary impact conditions.

- Nine-meter dynamic crush—for a package which has a mass not greater than 500 kg (1100 lb), an overall density not greater than 1000 kg/m^3 (62.4 lb/ft^3) based on external dimensions, and radioactive contents greater than 1000 A_2 not as special form radioactive material; a free drop of a 500-kg (1100-lb) mass from 9 m (30 ft) onto the package specimen. The dropped mass shall consist of a mild steel plate 1 m by 1 m, and it shall fall in a horizontal attitude. For this test, the package shall be positioned on a flat, essentially unyielding, horizontal surface so as to suffer maximum damage.*

- One-meter pin puncture test—a free drop of 1 m onto a 15-cm (6-in.)-diam mild steel pin is performed at various package locations such as closures, valve port covers, and joints which might be compromised by a puncture event.

- Fire test—a 30-min fire with a temperature of 800°C (1475°F). A package surface emissivity of 0.9 is assumed for calculational purposes. Fire tests are generally carried out less frequently than are drop tests because of the passive heat-transfer nature of modern packages and because computer calculations can reliably predict package temperatures.

A characteristic of all testing is the need to determine the initial conditions of the test specimen so that dimensional or property changes which occur during the test can be adequately documented. This usually means that metrology of the test article is required to define the dimensions of the test article prior to and after the test series.

The usefulness of certification tests may be compromised by lack of verisimilitude. The conditions of a test and the features of the test articles must be representative of the certification conditions being verified. Verification is difficult to achieve when accurate scale models of components can

*Because most package designs covered in this Handbook are of greater mass or higher density, the testing of a package to the dynamic crush test is not addressed further herein.

not be fabricated. Questions may arise as to whether the test is valid when components are not to scale or when test conditions are not bounding. The number of possible combinations of test parameters complicates testing; with hot and cold temperatures and various orientations of impact cause the greatest difficulty.

A key to any successful test is the proper identification of the desired performance of the test article so that adequate performance can be determined after the test. Identifying the proper performance allows an answer to the question: Did it work? The range of acceptable performance parameters is usually written into the drop-test procedures.

4.2 TEST MODELS

The physical testing of packages can be expensive. Costs are incurred by virtue of planning; instrumenting the test piece; reducing and evaluating the data; and, of course, destruction of (at least part of) the test piece. With small, simple packages (e.g., the size of 200-L drums), the expense of testing is often not as great as it is in the case of larger packages. The packages themselves are less expensive, instrumentation is generally minimal, and often the tests are intended to determine something very specific (e.g., whether the lid remains on the package and protects the contents from direct flame impingement). For these reasons, small packages are more frequently tested than are larger ones. However, packages with complex structures which may be difficult to analyze [e.g., the double-containment domed structure of a Radioisotope Thermal Generator (RTG)] may incur excessive analysis costs which would still require validation through tests.

In the case of large packages such as one designed to transport irradiated fuel, scale models are often used if testing is deemed desirable. Models should not be smaller than quarter-scale to ensure validity of the scaling laws. Models should have sufficient detail to ensure that all important structural features of the package are represented. Features which are not being tested, such as neutron shields for irradiated fuel casks, may be omitted from the test package; but the mass of features which are not modeled should be incorporated conservatively. For example, steel blocks may be welded to the exterior of a spent-fuel-cask model to represent the mass of the missing neutron shield material, but the use of separate blocks avoids the addition of nonscale strength to a section of the cask exterior.

The scaling laws are identified below. They provide some insight as to how the reduced scale model of a prototype affects the results of the tests as seen in the model. In these equations, S_m refers to the scale of the model as compared to the prototype, and the subscripts m and p refer to the model and prototype, respectively. These relationships do not account for strain rate effects or gravity.

Velocity	V_m	=	V_p
Weight	W_m	=	$S_m^3 W_p$
Deformation	d_m	=	$S_m d_p$
Applied force	F_m	=	$S_m^2 F_p$
Acceleration	A_m	=	A_p/S_m
Duration	t_m	=	$S_m t_p$
Momentum	MV_m	=	$S_m^3 MV_p$

Note that in a 1/4-scale model, the weight of the model is 1/64 that of the prototype; the duration of the impact is 1/4 as long; the deformation in the model scales directly (i.e., is 1/4 as great as it would be in the prototype); and the applied force is 1/16 of that which would be expected on the prototype. These relationships also imply how important is the need to scale everything that can affect the results of the test. Note that the size of the bolts used in scale models (e.g., to attach the lid to the body) scale directly; the stresses that are seen by the bolts are preserved, allowing the experimenter to determine if they are sized properly.

4.3 NINE-METER FREE-DROP TEST

The 10 CFR 71 regulations specify that a 9-m free-drop test should be performed in the worst-case orientation to demonstrate the capability of the package in accidents involving impact. For some packages, the worst case is not a single event since the worst case for the closure and the worst case for the body of the package may not occur in a single drop. For this reason, most package drop tests consist of a series of tests at various drop angles and possibly with various orientations about the package long axis. Scale-model or full-size testing of the package are acceptable.

The impact pad upon which the package is dropped must simulate an unyielding surface. Typically, this requires a hard steel surface over a concrete block which is embedded in the ground. The mass of the block should be at least ten times the mass of the test article to ensure that the impact pad does not recoil significantly during the impact.[1] A gridded backdrop (or stadia board) is sometimes placed behind the impact pad to provide perspective for high-speed photography of the drop event. A stadia board background is helpful in the determination of rebound or "bounce" of the test package, a parameter that is important to the assessment of secondary impacts.

Drop tests are often performed at ambient conditions. This is usually adequate because the structural strength of American Society of Mechanical Engineers (ASME) code materials is well-documented over the range of expected temperatures. Cold tests at - 29°C (-20°F) have been required when fracture toughness concerns apply to the package. Elevated temperature tests are usually not required to confirm package-structure capability if the package-structural materials are not significantly sensitive to temperature; although this is true for steel components, it may not be true for components fabricated from aluminum or other temperature-sensitive materials.

The rigging of a package for drop testing should not cause lifting cables or fixtures to interfere with the desired test. Because test equipment, such as accelerometer cables, is sometimes damaged during drop tests, care should be taken with the location and method of attachment to the test piece. Safety is a concern during drop tests because failure of components, such as impact limiter attachment devices, can result in hazardous projectiles with unexpected energies and ranges.

A drop-test series imparts high stresses on package structures, and damage to components often results. Impact-limiter attachments and lifting lugs added to a package to perform the tests are often deformed by the tests. Such components should be returned to their original configuration after each test so that any subsequent test is performed with an "as-manufactured" test article. Sacrificial shock absorbers such as impact limiters that have been damaged should be replaced, although it is customary to reuse these items for more than one test if it can be determined that the crushing, which occurred in one portion of the impact limiter, will not affect its performance at the drop angle and orientation of the subsequent drop. If multiple drop tests are carried out, it

is also prudent to verify the torque on any bolt on the test piece following the first test, and bolts which have detorqued should be returned to the specified torque value.

4.4 DROP-TEST OUTCOME

The test article may be inspected after the completion of each drop to verify that deformation of the package structure is within allowable limits. The deformation of sacrificial impact-limiting structures should be measured, and the shape of the crush volume should be documented by sketches and photographs. Careful attention should be given to measurements of deformation of any containment components. Strain gauges and accelerometers are particularly useful to determine the magnitude of any deformation and the associated stresses. Such data allow actual measured stresses to be compared with the calculated stresses of the SARP, which may show that the analytical models are conservative and adequate predictors of package performance during impacts.

The containment function of a package should be verified after each drop in a series. In many cases, verification can be achieved by a pressure test using a valve which is a normal component of the package or by a valve added to the test article. Packages which provide double containment, such as plutonium packages, may experience damage to the exterior containment. This damage can make it difficult to disassemble the package by removing the exterior secondary containment and thus can complicate the testing of the inner primary containment integrity after each test. In such cases, it may be more practical to verify containment after completion of a test series.

Closure lid bolts are designed to be sufficiently strong to survive drop impacts without tensile yielding, but bolts have been observed to detorque during drop impacts because of the dynamic loadings which occur. The magnitude of any detorquing which has taken place may be determined by measuring the break-out torque as the bolts are loosened with a torque wrench. If detorquing is observed to occur in an impact, all bolts in the test piece should be inspected to ensure that this effect was not caused by dynamic-tensile overloading of the bolts. Match-marking of the bolts and package body before a drop test can allow a quantitative measurement of angular rotation of bolts caused by impact loads so that it can then be determined if bolts have turned out and thereby have released the torque preload. Closure lid surfaces should be inspected for scarring or denting by the hard steel of the bolt, which can also result in loss of bolt preload.

Some packages require that impact limiters remain attached after drop impacts to provide thermal insulation. Whether an impact limiter will remain attached to a package is often difficult to prove analytically because of the large deformations involved; consequently, testing is frequently needed. Other performance requirements that cannot be ensured by analysis should be verified by testing.

4.5 PIN-PUNCTURE DROP TEST

The 1-m drop of a package onto a 15-cm-diam steel pin demonstrates the resistance of a package to puncture events. Plastic deformation of the structure in the region of the puncture is unavoidable, but containment of the material to be carried should not be affected. The pin-puncture drop requires that the impact be at a specific predetermined location, and to achieve this, precision is required in the rigging of the package prior to the drop. Most puncture tests are carried out with the package oriented in such a way that its surface is perpendicular to the main axis of the steel punch prior to the drop.

Areas of a package which may be tested via a pin-puncture event include closure lid or joint, closure bolts, valve port, impact-limiter attachments, and the body of the container at the location likely to cause the most damage (e.g., the cylindrical side of a cask). Only one end of a cask needs to be tested for puncture resistance if the ends are similar, but in the case of some packages (e.g., the RTG cask, which has a flat circular closure bottom end and a domed upper end), both ends should be tested.

Shock-absorbing materials such as impact limiters may be removed for puncture tests if their presence is not required for the event. Alternatively, the retention of impact limiters during a pin puncture may be a requirement for certification tests to demonstrate that the impact limiters are not stripped from the package by the pin during impact.

4.6 PUNCTURE-TEST OUTCOME

Plastic deformation in the area of the pin-puncture event is unavoidable, but the deformation should be limited, and containment should be maintained. Ductile steels are preferred for packages because of their ability to withstand the localized deformation of a pin puncture without rupturing or cracking of the structure. Multiwalled casks may suffer deformation of only the exterior wall, but after the 9-m drop and 1-m puncture test, the inner shell should be inspected regardless.

4.7 THERMAL TESTS

If a package needs to be subjected to the 30-min fire test, a full-size article should be tested because the various heat transfer mechanisms are different functions of temperature [radiation (T^4), convection ($\sim T^{1.33}$), and conduction (T^1)], and are dependent upon different geometric parameters. Testing of large packages can become quite expensive given the need to test full-size test articles. The reliability of modern computer codes has made it unnecessary to thermally test many packages that rely upon only passive cooling during hypothetical accident scenarios. However, when using analysis methods special care must be taken to account for unusual occurrences, such as decomposition of materials of construction with possible subsequent production of off-gases.

Testing of packages under normal thermal operating conditions may be required if calculated internal temperatures are determined to be significant and if the methods of construction involve poured materials whose thermal performance should be verified in situ. Examples include shields of lead or concrete which may be poured into the package structure for shielding purposes.

4.8 IMPACT-LIMITER TEST

Impact-limiting structures are usually tested concurrently with the package during drop and pin-puncture testing. Tests may also be required to evaluate the properties of impact limiters or crushable materials for drum containers. Materials that form part of the packaging may be studied by testing 8- to 10-cm cubes of the material at a variety of angles and temperatures. Such tests are especially important for nonisotropic materials because the axis of crush may alter the effective crush strength significantly. Testing of material samples can be performed quasistatically (very slowly) and the same test repeated dynamically (with a stroke speed of the crushing cylinder equivalent to a package which has fallen 9 m). The results of quasistatic and dynamic tests provide the static-to-dynamic crush strength ratio required for analyses.

The dependence of foam strength on temperature should be verified through testing. Foam materials generally soften at elevated temperatures and thus require a greater crush stroke to absorb the desired drop energy. At low temperatures, foams stiffen and a reduced crush stroke is needed, but the deceleration load applied to the package is greater. The variable properties of foam may be simulated, if the effect upon the package performance warrants, by using the stiffest foam crush strength and minimum crush stroke. Drop tests may be performed at ambient temperatures, thus avoiding the need to chill the package, by increasing the foam strength to simulate the stiff foam. Caution must be exercised with the adjustment of foam properties because of the loss of verisimilitude.

4.9 QUASISTATIC CRUSH TESTS

It is often desirable to test materials of construction used in impact limiters quasistatically as a function of drop angle in order to validate computer codes used to predict load-deflection curves for those materials. Tests are usually performed on test pieces or scale models in various orientations, and agreement between calculations and tests at these angles provides confidence that the calculations will be adequate at other drop angles. This allows a series of structural analyses to be performed to determine the worst-case drop angle without requiring an excessive series of impact limiter tests.

Quasistatic tests of scale impact limiters allow the performance of an impact limiter to be evaluated and verified prior to 9-m drop tests, which are typically much more expensive than quasistatic tests. The quasistatic tests may be performed with a standard tensile-test machine, which is operated in the compressive mode to crush the test impact limiter. The applied force is measured as the crushing progresses, allowing the plotting of the load-deflection curve which is required for 9-m drop analyses. The rebound energy may also be measured by slowly unloading the test impact limiter and measuring the force which the test limiter exerts upon the test machine.

Another benefit of such tests is that the behavior of the envelope or skin of the impact limiter may be evaluated. Ductile materials such as stainless steel are preferred materials for the envelope because of their capability to accommodate large deformations of the impact limiter without splitting at manufacturing joints, which can be a problem for aluminum and Kevlar™.

Quasistatic tests may verify the design of the impact limiter attachment methodology, allowing confidence to be gained through less expensive tests before the drop tests are performed. Impact-limiter retention is more easily demonstrated with quasistatic tests because the crush progresses more slowly and the attachment area may be inspected during the crush event.

Caution should be used in the interpretation of quasistatic test results because the behavior of some materials may be different in static and dynamic situations. Wood is a common material for impact limiters which can behave properly during a quasistatic test, but it may fail in the dynamic environment of a 9-m drop because the wood tends to split as it is crushed. Therefore, if the impact limiter envelope does not contain the wood in a dynamic impact, the wood may shatter and be expelled from any split in the limiter skin before it can do its job of absorbing drop energy.

4.10 SEAL TESTS

Should it be necessary to test seals of a particular package design, the tests should be performed at temperatures within which the package is expected to operate in service, especially at the cold (-40° C) temperature and at the hot (fire scenario) temperature. Tests must be performed with full-size seals, preferably the model which is expected to be used in the package. The test fixture for the leakage tests should mock-up the area around the seal region sufficiently to reproduce the thermal environment of the package. Leakage tests should be in accordance with American National Standards Institute (ANSI), Standard N14.5, which allows helium leakage to be measured externally when the test vessel is pressurized with helium, or internally (through a test penetration) when the test vessel contains a vacuum and the region outside the vessel is flooded with helium. The first test configuration is normally used for ambient- and low-temperature testing, and the second may be used for fire-temperature testing, which might be performed in an oven filled with helium.

4.11 CONCLUSION

The variability of packaging needs and designs prevents a definitive series of tests from being categorically specified herein, but a well-planned strategy of tests can provide a balance between analysis and testing of package performance. The containment function of a package may be demonstrated through analysis with confirmatory testing, but some capabilities of the package may be difficult to demonstrate analytically. The purpose of a test, whether for engineering knowledge or for certification, affects the test plan and the quality assurance requirements.

4.12 REFERENCE

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5. Structural Analysis

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5.1 INTRODUCTION

The regulations which govern shipping casks for radioactive material or spent fuel permit a variety of approaches to the task of demonstrating the structural adequacy of the shipping package. The most straightforward method is to perform a series of structural calculations to evaluate the package performance in the regulatory scenarios, which involve normal operation conditions and a set of hypothetical accident conditions, primarily a 9-m drop impact and a 1-m pin-puncture drop impact. These scenarios must not result in excessive stresses under a variety of hot and cold temperature combinations.

The analysis of the structural integrity of a shipping package differs from the analyses of the thermal, shielding, and criticality safety in one important respect: No single computer code or methodology for structural evaluation that can be applied to all parts of the cask in a single analysis. The structural analysis is performed in pieces; for example, the lift trunnions may be analyzed by hand calculations or a computer code, while the cask body shells are usually analyzed by a finite-element computer code. The acceptable stress levels are different for these components, and different *Regulatory Guides* apply. This makes the task of integrating the pieces of the structural analysis into a cohesive chapter of the *Safety Analysis Report for Packaging* (SARP) more difficult, and often results in inconsistencies. A further complication is that various components of the package are analyzed at different times during the SARP preparation. For example, a component such as lift trunnions may be analyzed during the early stages, perhaps with preliminary temperature calculations. In an effort to avoid recalculation as temperature fields are refined, a conservative temperature may be chosen. The final SARP will then have discontinuities between the calculated temperatures for the trunnions and the temperatures used in the structural analysis, and the (temperature-dependent) allowable stress limits will be different from other components fabricated from the same material. These inconsistencies are not critical, but they will result in lengthened review time as the regulatory agency attempts to put the various components, temperatures, and stress limits into proper perspective. Questions asked by regulators often include requests for explanations of such discrepancies. The preparation and review of the SARP can be expedited by careful planning to limit the number of discrepancies to the unavoidable minimum. One approach to improve consistency is to trace the load path in normal operation from the transport vehicle (which is not part of the package), through the cask supports or tiedowns (which are part of the package), into the cask body and its contents. The same procedure can be applied to the hypothetical accident-scenario loads.

The structural analysis of shipping casks requires special attention to two areas: (1) the loads developed in impacts and (2) the strength and stability of the structure that resists those loads. Two types of loads are developed in impacts: loads produced in the 9-m drop scenario by deformation of impact limiters (shock absorbers) or by contact of the cask with the ground, and loads produced in the pin-puncture scenario. These loads cause stresses in the cask body and closure lid, and the stresses must be calculated and shown by analysis to be less than the allowable stresses for the material of the structure. The allowable stress limits for cask materials must be specified by code (such as the American Society of Mechanical Engineers [(ASME Boiler and Pressure Vessel (B&PV) Code)] or must be documented adequately with test results. Noncode materials may be used for structural components, but the material properties under all operating and hypothetical accident conditions must be known. Shielding materials also contribute to the weight (and stresses) of the cask and may cause unusual loading upon the structural shells of multiwall casks. For example, lead slump is an important phenomenon in lead-shielded casks because lead slump produces a dynamic pressure upon the cask shells in an impact.

Stability of the cask structures is the ability to resist buckling stresses for the cask shell or shells, and internal structures (baskets) which hold the cask contents must be capable of resisting buckling stresses as well. Structures should not experience increased stresses as the geometry changes because of elastic or plastic deformation; this ability to resist such stresses is sometimes called geometric stability.

The materials which form the containment boundary of a cask must be qualified by elastic analyses; that is, they may not yield in hypothetical accident scenarios.* Noncontainment components may function in the elastic/plastic regime. An example is impact-limiter attachments, which may deform substantially in a 9-m drop impact without losing their functionality. Ductile materials are preferred for structural components because they can absorb a significant quantity of strain energy in an impact without immediate failure of the component. Materials such as stainless steel can survive elongation of tensile test specimens of approximately 40%, while structural aluminum alloys may be capable of only 10 to 15% elongation before failure. This capability adds a measure of conservatism to cask designs because a design subjected to impacts somewhat more severe than design values would yield without catastrophic failure. (Note that this is, in a sense, taking credit for material properties beyond the elastic region.) Ferritic steels may also be used, and the fracture toughness of such steels must be evaluated at –40°C with methodologies described in two *Regulatory Guides*: one for shells with less than 4-in. thickness[1] and one for shells greater than 4-in. thickness.[2] Materials such as nodular cast iron may possess ductility, but they are not favored for transport casks because the potential for crack growth adds a level of uncertainty which must be evaluated by fracture mechanics methodologies. Further, the uniformity of cast material may be more difficult to ensure than for plate or forged materials.

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*The ASME code allows yield to be exceeded in containment vessels, such as reactor vessels and steam generators, which might be stressed beyond the elastic limit by internal pressurization. To ensure against loss of containment, a stress limit of 0.7 S_u is applied. In such cases, the geometric form of even a highly stressed vessel is preserved. However, rules that apply to a constant geometric configuration do not necessarily apply to a transportation container because plasticity could allow a geometric deformation and alteration of load paths in such sensitive areas as closure lid seal surfaces. Thus, a yield stress limit is appropriate for transportation package containment.

Packages must also be analyzed in normal operation conditions to ensure that the package, cask supports, trunnions and impact-limiter attachments will not be affected by normal operation loads and stresses. Cyclic-applied loads or vibrations must be analyzed, especially for such components as tie-downs. Normal-operation thermal stress loads must also be evaluated. The thermal loads are often treated as a secondary stress in structural analysis, but the relatively high thermal gradients present inside a spent fuel cask can cause much higher levels of thermal stress, which should be added to the stresses induced by normal operation or accident scenarios. Component stress allowables should be evaluated at the appropriate temperature of the component in the scenario. Allowable stress limits may be evaluated at various locations within a component, such as the interior of a fuel basket and at its periphery. The impact stresses are highest at the basket periphery, but the temperature is lower than that at the center, and the allowable stress limit is thus higher than it would be if the basket-center temperature were used for the entire basket.

The impact analysis of casks with classical structural methodologies or computer codes allows the determination of the stresses, and resulting structural margins, in many locations of the cask and at many drop orientations. The regulations of 10 CFR Part 71 specify that the cask be evaluated for end, corner and side 9-m drops, although the U.S. Nuclear Regulatory Commission (NRC) prefers the designer to consider other oblique angles between the corner and side. The locations of greatest interest in the cask are the closure lid, lid bolts, and the cask mid-body, at which the greatest bending moments are developed. The discontinuity stresses produced in multiwall casks, which change from a relatively thin structural wall into a much thicker end, are also of particular interest. The containment boundary of the cask must be qualified under ASME Sect. III or an acceptable alternative in that the calculated stresses must not exceed yield, or a lower stress limit if appropriate. The quantity S_m, equal to 2/3 of the yield stress for steels, is a commonly used stress limit for normal conditions of transport (NCT). The inner shell of a multiwall cask is typically limited by the buckling stress limit. ASME Buckling Code Case N-0284 (ref. 3) may be used to evaluate the buckling stresses of cylindrical shells. Alternative methods are allowed, but the factors of safety required may be larger than if N-0284 were used.

Casks may also be licensed by testing. Drop testing provides confirmation that the structural analyses at tested drop orientations are correct. By extrapolation, the analyses of untested orientations are also likely to be valid. A typical series of tests includes 9-m drop tests on the cask end, side, and corner. Drop tests may also be performed at intermediate drop angles if one (or more) angles can be shown by analysis to be the worst case for a given structural requirement. Small casks with simple geometries and large structural margins may rely heavily on tests in place of detailed analyses. More complex casks, including Type B packages, are typically licensed based upon the structural analyses plus confirmatory tests.

The testing of casks should be guided by the material properties of the cask components. For example, room-temperature testing is adequate if the material properties of the components being studied are not affected by the extremes of high and low temperature. In some cases, the load conditions are affected by temperature. An example is foam impact limiter stiffness, which changes substantially over the range of operating temperatures. Impact testing at high or low temperatures is often very difficult and may be avoidable if the material properties are predictable as a function of temperature, so that behavior at temperature extremes can be extrapolated with confidence from room temperature test results. Dynamic testing also verifies the stability of the structures, which could be difficult to demonstrate analytically.

The principal parts of the shipping cask from a structural perspective are the following:

Cask body (single or multiwall)
Closure lid(s)
Valve covers for containment penetrations
Basket for fuel or radioactive material
Impact-limiters (or shock-absorbing structure)
Impact-limiter attachments
Lifting/rotation trunnions
Cask support structure and tiedowns
Transport vehicle (which determines normal operation loads)

A measure of consistency can be introduced into the analyses of the SARP by tracing the load path from the physical origin of the loading force through the various components of the package. The normal operating loads originate in the transport vehicle and pass through the cask support structures and tiedowns into the cask trunnions (if used) and, thence, into the cask body. The normal operating loads also affect the cask basket and radioactive material contents.

Hypothetical accident loads originate in the impact limiter (not the unyielding surface) as the crushable material deforms. The application of deceleration force caused by the crushing of the impact limiter is very important to the accurate determination of stresses as those forces are transmitted to the cask body. Inaccurate modeling of the force application can result in unrealistic stresses predicted for the cask body. The hypothetical accident loads stress the cask surface and eventually affect the cask basket and contents.

All physical data and analysis methods necessarily contain varying degrees of uncertainty. To ensure that a safety analysis is conservative and defensible, conservative analytical methods and engineering values are often used. The ratio of the calculated value for a stress and the allowable stress limit for a component is usually called the margin of safety. Small margins of safety may indicate a component that is highly stressed and redesign might be appropriate. However, consideration should be given to the conservatism contained in the parameters and analysis methods used to ensure that a component which actually contains an adequate margin of safety is not redesigned, thus increasing weight and increasing stresses in other components. A review of the conservatisms contained in the load path from the impact site throughout a cask can sometimes identify overconservative loads or analyses which contain a higher measure of conservatism imbedded in an analysis. Such overconservative analyses penalize other components and result in components which are overdesigned in comparison to the remainder of the cask. This condition may result in a loss of balance in a package design and exact penalties in weight and functionality.

The materials used to fabricate a cask must be evaluated for adequacy in the particular application. The stress limits of a given component must be determined from the material properties. Material selection should consider whether the component must operate in the elastic range only or whether limited plastic deformation is permitted. Operation in the elastic or moderately plastic range ensures that stress computations are reliable. Gross deformation alters the geometric parameters of a structure and may reduce the predictability of loads and stresses. Materials used for casks should be sufficiently ductile such that fracturing will not occur. Stainless steels are commonly used for this reason, although ferritic steels, some aluminum alloys, and other materials may be acceptable for some components. Materials which become brittle at low temperatures (–40°C) are generally not acceptable.

The ASME B&PV Code, Sect. III, Subsect. NB is recommended by NUREG/CR-3854 as the code to which the shipping cask containment boundary components should adhere. Section III, Subsect. NF may be used internal cask structures, such as a spent fuel basket, and Sect. VIII may be used for the outer wall of a multilayered cask. Article NB-2000 of Sect. III, Subsect. NB requires that pressure-retaining material shall conform to the requirements of one of the specifications for material listed in Sect. II, Part D, Subpart 1, Tables 2A and 2B. Article NF-2000 of Subsect. NF requires that material for component supports shall conform to the requirements of the specifications for material listed in tables of Sect. II, Part D. Certain components are exempt from ASME specifications provided that they are not connected to a structural support or pressure boundary, as stated in NF-2121 (b). A component is "connected" if failure of that component or its attachment could compromise the functionality of other components of the cask which are important to safety.

An example of the effect of structural connectivity is that items such as shielding placed inside the containment boundary of a cask could affect the function of a spent fuel basket should the internal shielding fail in a hypothetical drop accident scenario. Hence, internal shielding would have to be structurally restrained in a manner equivalent to the basket structure. Similarly, the lid of a container transported within a cask would require structural restraint equivalent to any items important to safety located within the container.

5.2 CASK BODY

Casks may be made from single-wall structures, typically steel, or by a multiwall sandwich structure. Multiwall casks are lighter in weight because an efficient gamma radiation shield with a higher density than steel, such as lead or depleted uranium, can be used. Since the thickness t (or r in a cylinder) of a gamma shield for a specific external radiation dose rate is proportional to the density of the shield material, using a higher density material will result in a lower weight cask. This is because, while the shield thickness (R) for a cylindrical cask varies directly with the shield density, the volume, and, hence, gross weight, is proportional to πR^2 for a cylindrical cask. Thus the cask weight is less if the shielding capability of the package remains the same. This effect is especially significant for casks with a relatively small cask-cavity diameter (such as legal-weight spent fuel casks or smaller) and is less noticeable for larger rail casks.

Stresses caused by lead slump must be evaluated. These stresses appear as hydrodynamic pressures on the cask shells of a multiwall cask. Lead slump must be evaluated for side-drop and end-drop orientations. The lead can be treated whether or not it is bonded to the structural walls of the cask. If the lead is bonded, lower stresses result from the slump, but the presence of the bond must be verified after cask fabrication and periodically in service. (Lead is usually bonded to improve the thermal conduction through the cask shells.) The volume of a void which could be formed by lead slump must be calculated because this produces a "window" in the lead radiation shielding.

Quasistatic finite-element analyses typically require the specification of a boundary condition in order to develop a map of the stresses in the package under load conditions. The stresses that are calculated in the cask shell(s) should not be distorted because of the boundary conditions that are applied; this concern is important because the application of some boundary conditions can cause locally high stresses in regions of the cask which are remote from the impact zone, and such high stresses are thus nonphysical. Even in examples in which the stresses are lower than the stress limits, the nonphysical behavior of the stresses near the boundary condition is undesirable.

Adjustments to modeling methods or recalculation of stresses in the region near the boundary condition by using a different model may be used to correct stresses in these regions.

Quasistatic analyses of the cask impact stresses may be less expensive than dynamic analyses, and quasistatic analyses result in calculated stresses which may be directly compared to ASME stress allowables. If dynamic analyses are performed, dynamic allowable stress limits must be calculated and documented. A quasistatic analysis may be adjusted for dynamic effects by scaling the calculated stresses with a "dynamic load factor." Typically, the dynamic load factor for impacts less than 100 *g* range from 1.0 to 1.2. Higher values for the dynamic load factor should be used for loads greater than 100 *g*. The dynamic load factor used in the analyses of the SARP should be confirmed in scale-model drop testing. The stresses in the scale impact tests may be compared with the calculated quasistatic stresses, adjusted by the dynamic load factor, to demonstrate a measure of conservatism.

5.2.1 Pin Puncture

Multiwall casks, consisting of a structural inner wall, a shield annulus, and a structural outer wall, are usually designed to provide required shielding in a package with the minimum weight. A benefit of this construction is that the outer structural wall of a multiwall cask provides protection from pin puncture so that the inner structural wall is not deformed in a puncture impact. Lead shielding effectively isolates the inner structural wall of a multiwall cask. The lead layer acts as the shock-absorbing medium, distributing the energy of the puncture impact, which is concentrated in a 15-cm-diam circle of the outer shell, over a larger area of the inner shell. The puncture shock wave propagates inward from the outer wall and approximates a hemispherical shock pattern. Hence, the energy in the shock wave decreases as the square of the distance traveled inward toward the inner wall. This effect is illustrated in Fig. 5.1. Some nonscale dimpling of the inner structural wall of quarter-scale models can occur in pin puncture because of nonlinear scaling of dynamic shock front. If the scale of a drop test model is 1.4, then the shock transmitted to the inner cask wall is increased by a factor of 16 by the nonlinear effect and reduced by a factor of 4 by the scaling of the puncture pin. The net result is a factor of 4 increase in shock energy transmitted to the inner cask wall. Dimpling of the inner structural wall caused by this effect is acceptable if the deformation does not cause an inteference fit with the cask cavity contents (fuel basket, special form container, or waste container)–which would raise the possibility of damage to the cask contents.

A series of pin puncture tests performed at Oak Ridge National Laboratory were used to develop an empirical equation[4] for the stress in the outer wall of a multiwall cask as a function of the mass of the cask and the thickness of the cask outer wall material. This equation (Nelm's equation) applies to steel-lead-steel cask wall construction. Nelm's equation may be used to demonstrate-pin puncture adequacy for casks with stainless steel or ferritic steel walls, and this equation has been the basis for the puncture analysis of several licensed casks.

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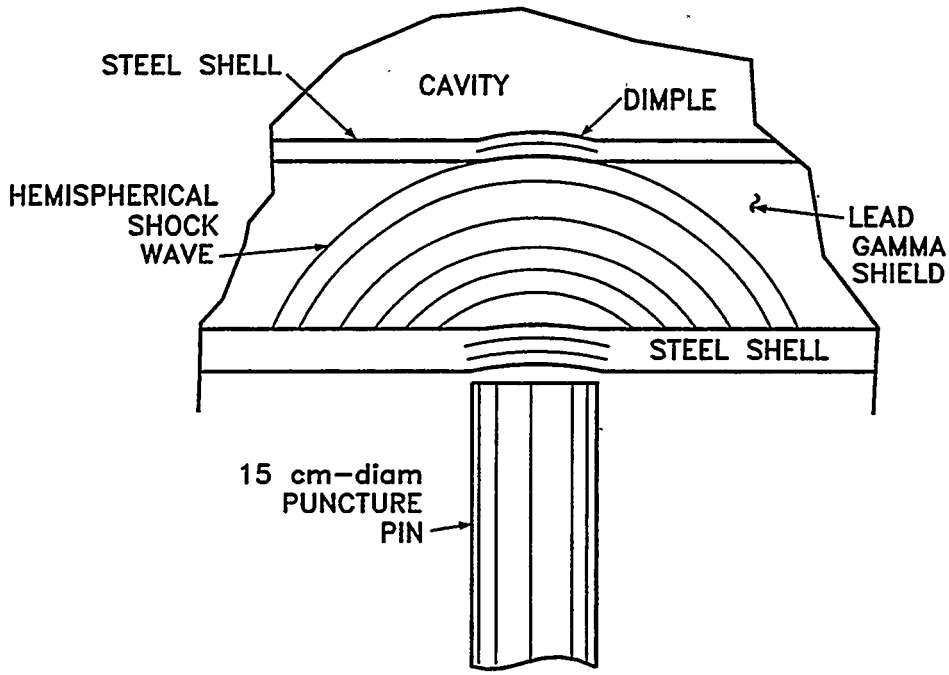

Fig. 5.1. Pin-puncture event.

For hot-rolled carbon steel and stainless steel outer shells, the minimum outer shell thickness required to withstand the punching action of a steel piston is given by Nelms equation:

$$t = (W/S)^{0.71} , \tag{5.1}$$

where

t = shell thickness, in.,
W = cask weight, lb, and
S = ultimate tensile strength of the outer shell, psi.

Note that this equation gives acceptable results when the diameter of the package is greater than about 75 cm and when the impact location is not close to a stiffening structure such as a heat transfer fin; for packages having a diameter below 75 cm, one should apply a factor of 1.3 to the resulting outer-shell thickness.

Nelm's equation does not apply to steel-depleted-uranium (DU) steel multiwall constructions because DU has properties similar to those of mild steel. The impact force of a puncture pin is thus transmitted to the inner structural wall in a concentrated form. A relatively thin layer of lead in a steel-DU-lead-steel cask provides puncture shock isolation because of the shock-absorbing effect of the lead and because the deformation of the outer wall must exceed the thickness of the lead annulus before the outer wall directly contacts and loads the DU layer. The minimum thickness of lead required to provide puncture shock isolation is package-design-dependent, but an analysis[5] of typical rail casks showed that a layer of 0.125 cm or more provided isolation. It should be recalled that the 15-cm-diam pin is constant regardless of the size of the cask and that a thin layer of lead would therefore be effective regardless of cask size. The manufacturing

process required to provide a lead layer with consistent thickness is difficult and can require vacuum pouring.

Single-wall monolithic casks must withstand pin puncture with only local deformation. Fracture toughness requirements for monilithic casks may be evaluated per NUREGs for shells less than 4 in. (0.1 m) thick[1] and greater than 4 in. (0.1 m) thick.[2]

5.2.2 Bending

Cask shells must resist bending tensile loads without yielding. Compressive bending loads can lead to buckling instability that causes failure of a cylindrical shell before yield is reached. Analyses must show that the containment (inner) shell of a cask does not buckle. ASME Buckling Code Case N-0284 may be used to demonstrate that buckling does not occur. Classical crippling or buckling analyses may be used as an alternative to N-0284, but factors of safety appropriate to the specific cask and calculational methodology must be established. The factors of safety specified in N-0284 are applicable only if the methodology established in the N-0284 is used; these factors of safety may not be used with classical crippling or buckling analyses. Factors of safety are applied directly to the load to account for ovality and variations in shell thickness and must be evaluated together with the buckling stress methodology to ensure conservatism. Hence, the factors of safety of N-0284 may not be excerpted and applied to classical crippling or buckling analyses.

Cylindrical shells are desirable for casks because the stresses on them are axisymmetric. Casks with square or rectangular cross sections have been licensed in the past, but stresses are higher in their corner areas. Square shells with rounded corners reduce these stresses somewhat, and the rounded corners avoid discontinuities in the stress distribution which might be difficult to calculate precisely.

5.2.3 End Forgings

Forgings used to form the ends of casks may contain penetrations for filling or venting and could cause local stress concentrations and weak areas. Robust sections are desirable so that the effect of penetrations is minimized. Closure end forgings cannot take credit for any support that might be provided by bolted closure lid(s) and must be treated as unsupported rings.

The joint between shells and end forging (castings are not favored because of potentially nonuniform material properties) should not be a butt-weld. A portion of the shell material must be part of the end forging, as required by the ASME code, so that the weld joint is between cylindrical sections. Full-penetration welds are required to join sections of the containment boundary.

Weld efficiency factors may not be applied to containment welds. Radiography is required for all containment welds per the ASME code, and derating of weld zone stress allowables through the use of efficiency factors is, therefore, unnecessary.

Transition zones, thicker shell material near the end forgings, can reduce the effect of discontinuity stresses, as illustrated in Fig. 5.2. The taper (slope) of transition zones should be 1:3 (1-cm decrease in a 3-cm length of shell) or shallower as directed by the ASME Boiler & Pressure Vessel Code (B&PV), Sect. III, Division 1, NB-3339.6, and NB-4232.1.

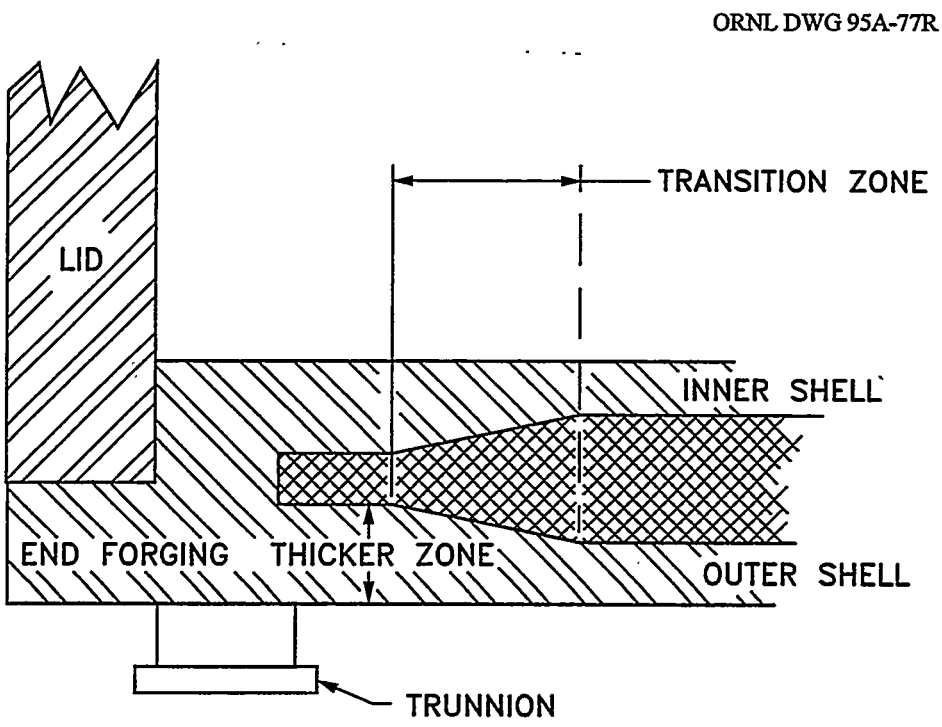

Fig. 5.2. Cask-shell forging joint with transition zones.

5.2.4 Fabrication Stresses

The fabrication processes may result in residual stresses which can affect the package and cause premature component failure. Fabrication operations which involve bending of components always result in residual stresses. Such residual stresses may be reduced or removed through stress-relieving operations if the component is not subject to distortion from heating. Cask shells are often fabricated by the rolling of flat sheets into cylinders or hemicylinders, which are later joined by welding. Weld locations must be specified in the SARP and welds in the containment shell must be radiographed. Rolled shells may be stress-relieved prior to final machining to eliminate the fabrication stresses.

The pouring of lead shielding into the region between two structural shells of a multiwall cask body also introduces fabrication stresses. Uneven heating of the inner and outer shells is often unavoidable, and differential thermal expansion results. Stresses may be avoided or reduced if the inner and outer shells are not structurally joined during the pouring operation, but this is not always possible. The lead cooldown is also a source of fabrication stresses because the lead will contract more than the inner cylinder and apply a compressive load. This effect must be evaluated to demonstrate that the inner cylinder buckling-stress limits, at the elevated temperatures present during the lead pour, are not exceeded.

When a poured lead annular cylinder has cooled, a gap is often left between the outer structural wall and the lead annulus. Careful pouring procedures can minimize this gap formation by "topping off" the molten lead as it cools and shrinks. Directional cooling, allowing the downward end of a cask to cool first, ensures that the surface of a solidifying layer of lead is covered by molten lead so that the molten lead will flow into the gap as it forms. The bulk temperature of

the cask should remain at or above the maximum predicted operating temperature to ensure that expansion of the lead does not cause excessive pressures to develop when the cask is heated in service.

5.2.5 Welding

Welding is discussed in Chapter 13, "Package Fabrication and Acceptance Testing."

5.2.6 Effects of Drop-Accident Scenarios upon Lead Shielding

The shock caused by an impact of a 9-m free fall of the cask onto a solid, unyielding surface can cause the lead shielding of multiwall casks to flow. This phenomenom is usually called lead slump; it causes the lead to press inward and outward simultaneously on the cask structural walls, similar in effect to a hydrodynamic pressure. The forces produced can be calculated by applying the deceleration force generated by the cask impact limiters to the lead medium in order to compute the equivalent hydrostatic pressure. The inward directed force applies a buckling load to the cask inner cylinder. The outward load can cause bulging of the outer cylinder.[6] The elastic limit should not be exceeded for the inner shell, but moderate plastic deformation may be acceptable for the outer shell.

The determination of the material properties of lead are problematic since lead is not an ASME code material. However, there are ASTM specifications for several types of lead, including chemical lead and corroding lead. Both of these types of lead are high purity, and their static and dynamic material properties are well documented over a range of temperatures. The dynamic properties of lead are not as well established, but room-temperature dynamic properties may be scaled to higher and lower temperatures by the ratio of the static properties at the appropriate temperatures.

Lead may be bonded to steel shells during a lead pour using a lead solder which is applied to the shell surfaces before the pour. The effect of lead bonding is to reduce the amount of lead slump which occurs in a drop scenario, and the buckling stresses are accordingly reduced. Lead bonding is usually chosen by the cask designer to improve heat conduction through the cask wall, and its structural benefit is a secondary consideration. The dynamic properties of bonded lead must also be established and documented.

5.2.7 Effects of Fire-Accident Scenarios upon Lead Shielding

The pressures that develop during a fire-accident scenario that causes lead melting can exceed the strength of the shells and result in rupture of the cask wall. The outer neutron shielding of spent fuel casks acts as a thermal jacket to protect the lead, and additional thermal insulator may be added to protect lead shielding which lies close to an exposed surface.

Lead melting, which might occur if a cask were accidentally involved in a long-duration fire, could cause relocation of lead, which, in turn, could cause the shielding capability to degrade in specific regions of the cask body. Such degradation could lead to localized increase in the gamma-radiation dose rate external to the outer shell of the package. Lead melting may also cause excessive pressure to be developed in the lead zone, a phenomenon which has occurred in some

tests and which could lead to the failure of the cask-shell/end-forging joints. Such possibilities must be carefully evaluated when a lead-shielded cask is designed.

5.3 CLOSURE LID

A cask may have single or double lids for its containment closure. Both ends of the cask may have lids, if desired, to aid in fuel-handling operations, such as horizontal fuel loading and unloading. If possible, the closure lid should be recessed within the cask end forging to avoid side loads on the closure bolts, which would result in shear in bolts. The closure lid should be stepped to minimize thickness of ring of end forging material. If the lid cannot be recessed, the outer diameter of the lid should be less than the outer diameter of the end forging to mitigate side loading. The ring of material that forms a recess should not yield under hypothetical accident loads because any deformation will close the gap to the closure lid and could cause difficulty in the removal of the lid after an accident. Local plasticity caused by a pin puncture impact is often unavoidable and is usually permissable if it does not affect the functionality of the lid.

In an accident situation, closure lid bolts must be analyzed for uneven loading that could be caused by the weight of the lid plus the forces generated from the impact of the contents of the package against the underside of the lid, if such an impact is possible. A "pinned lid" analysis should be performed, during which one bolt takes a hight load while the lid tries to "pivot" around the bolt located opposite it on the bolt circle. Closure lid bolts should be fabricated from a steel with thermal expansion properties similar to those of the lid in order to minimize any differential thermal expansion and thermal stress in a fire-accident scenario. The lid-bolt material should be ductile such that the yield strength does not exceed 80% of the ultimate strength. Helicoils are often used to provide higher strength materials for the threads and to provide a mechanism to facilitate replacement of damaged threads in the cask body.

The closure lid should generally be free of protrusions which could cause load concentrations in an impact on the closure lid end of a cask. Load concentrations occur when an impact occurs directly on a protrusion for a small cask (without impact limiters) or when the impact limiter crush stroke of a larger cask is inadequate to protect a protrusion. Lifting attachment fittings (such as D-rings) should be recessed, or fittings (such as eye bolts) may be bolted into lid bolt hole locations during cask handling.

5.3.1 Closure Seals

Seals for the closure lid may be metallic or elastomeric. Teflon™ seals are not elastic, but they provide a seal that is more tolerant of minor surface imperfections and scratches. The closure-bolt load must preload the seal to prevent leakage. Bolt stretching during end-drop scenarios can cause the seal to lift from the seal surface, thus allowing a transient release of cavity gas ("burping"). This transient loss of sealing may be acceptable if the maximum possible radionuclide loss is within regulatory limits and if the seal will always reseal following the transient. Metallic seals are more prone to such transients because the metallic seal deflection is less than an elastomeric and, hence, less bolt extension is needed to unseal. Performance of seals in cask drop scenarios is often demonstrated by testing rather than by calculation. Demonstration of an acceptable leak rate before and after drop testing indicates that the seal is performing adequately.

Closure seals are typically face-type seals in which an O-ring in compressed by the lid against a flat surface machined in the cask end forging. This type of seal provides a reliable seal when the closure lid bolts are torqued, to an easily predetermined value. Bolt-torque seal preload calculations must be confirmed by testing the seal during the factory acceptance tests. An alternative seal type is the piston-ring type, which consists of an O-ring which encircles a machined portion of the closure lid. The seal is formed by compressing the O-ring between the cask end forging and the outer perimeter of the closure lid. This type of seal is very sensitive to machining imperfections, scratching, and contamination and is not favored for very large closures. An advantage of this type of seal is that the diametral space requirements are less than for face-type seals because the O-ring rides in the gap between the lid and the cask end forging bore.

Seals are discussed in more detail in Chapter 8.

5.4 VALVE COVERS

Valve ports allow penetrations to drain and vent the cask. Valves have been used as containment boundary in the past if they were qualified at hypothetical accident fire temperatures; they may be used in postaccident recovery in order to test for radionuclide release from the cask contents. Valves must be protected whether they serve as part of the containment boundary or not; they may be covered by port covers which provide the boundary. Quick-disconnect valves may not be used as part of a containment boundary, but they are convenient designs which require the use of port covers to provide the containment boundary. Valve covers should be recessed to prevent side loads, which would cause shear in the cover bolts.

A valve cover must withstand direct pin-puncture forces. Credit for impact-limiter protection of a valve cover can be taken only if the limiters can be shown to remain attached after 9-m and pin-puncture impacts, and worst-case damage (deformation) of the limiters by these impacts is included in the analysis. Testing of the impact limiters may be used to qualify the use of the impact limiters in puncture scenarios.

The worst-case angle for a pin-puncture event on a valve port cover is sometimes difficult to establish. Recessed ports with a smaller diameter than the puncture pin cannot receive the maximum load of the pin in a perpendicular impact orientation, as shown in Fig. 5.3. A small deviation of the cask body from the horizontal can cause the puncture pin to impact a recessed valve port cover, and essentially the maximum load of the puncture pin is applied to the port cover. The impact load in this scenario is concentrated at the center of the cover plate. This usually represents a worst case.

Valve port covers may consist of one or two plates. Double covers are used to separate the containment sealing function from the pin puncture protection function, as shown in Fig. 5.4. Single plates are more convenient for operation and require a smaller outer diameter than the outer plate of a double cover system. Valve port cover seals may be face-type seals or bore-type seals.

Containment valves or valve port covers which provide containment must be qualified at the extreme temperatures predicted in a fire-accident scenario. Seals that are temperature-sensitive should be tested at the low end of the temperature range (–40°C) as well as at the high end.

Valve port covers may be fabricated from high-yield-strength material to improve resistance to pin puncture events. Excessive deflection or deformation of a cover plate could compromise the

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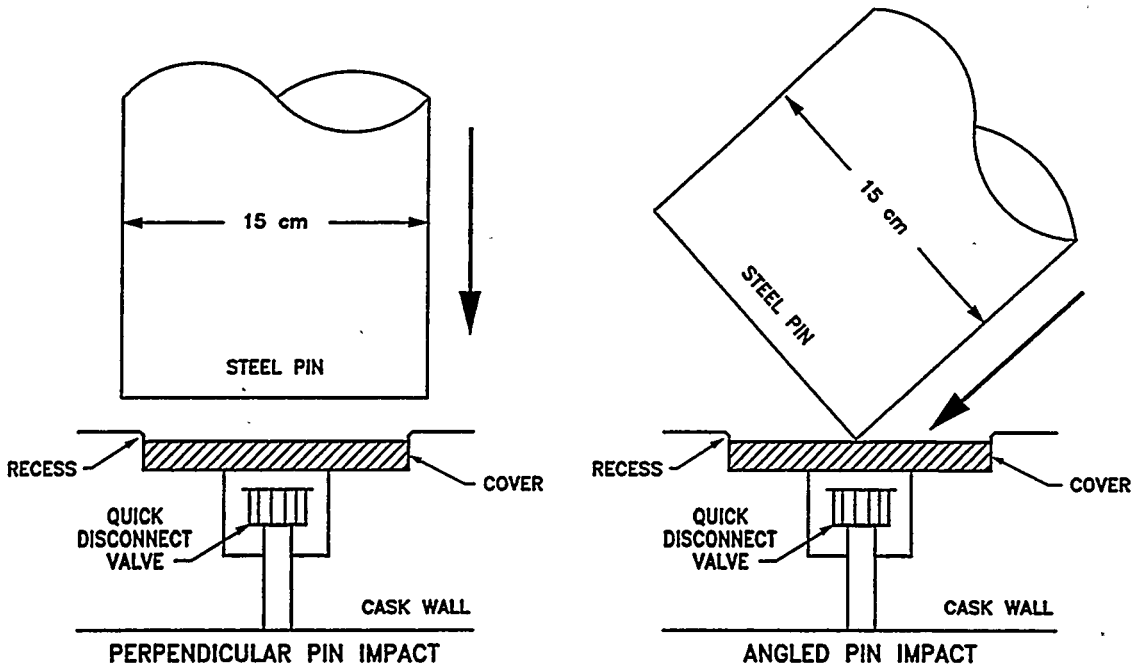

Fig. 5.3. Valve port cover pin impact orientations.

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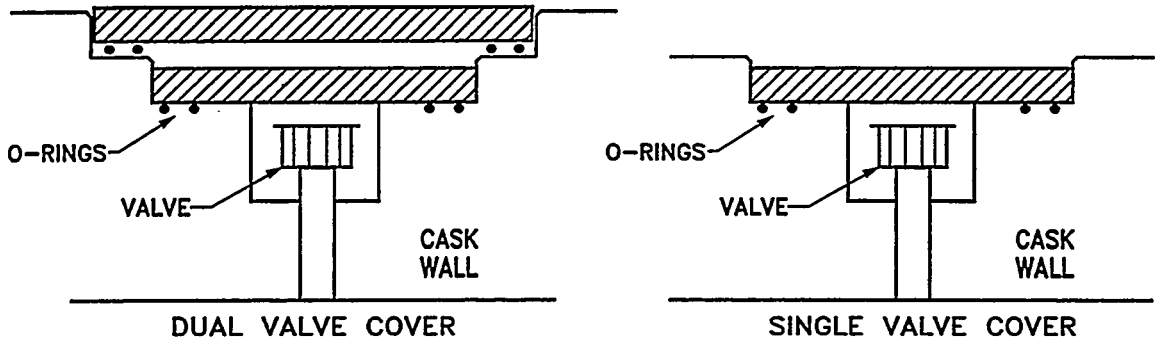

Fig. 5.4. Single- and double-plate valve cover designs.

sealing capability; therefore, the yield stress of the cover plate material should not be exceeded. The use of dissimilar metals for the port cover plate and the cask body can create a pin-puncture effect in which the harder cover plate material applies a local load and bearing stress to the cask body material of the port area, thus exceeding the allowable bearing stress limit.

5.5 BASKET

The contents of a shipping cask may require structural support to protect the radioactive material and maintain the integrity of the contents envelope, to minimize the spread of radioactive material and avoid excessive radionuclide contamination of the cask interior cavity. Fissile material casks may also contain support structures, or baskets, which contribute to the ensurance of subcriticality in all conditions of transport. If the structure of the basket is important to criticality safety, then the structure must not exceed yield in any substantial portion. Yield is chosen as a limit so that all structural materials remain in the elastic regime, thus ensuring that the structure does not change geometry by deforming. Such deformation would typically affect the criticality safety of the package in an adverse fashion. The yield stress limit ensures that the geometry analyzed for criticality safety is maintained. The basket structure must not buckle if it is important to criticality safety. Further, components contained within the cask (which are not part of the basket) must also satisfy the structural requirements of the basket if failure of such components could interfere with the ability of the basket to perform its function. A component is "connected" if failure of that component or its attachment could compromise the functionality of other components of the cask which are important to safety.

The basket may yield if it is not important to safety, although yielding should not be so extensive that the contents of the cask cannot be removed after an accident. Localized plasticity around the outer edge of a basket is permitted if the outer edge does not establish an important geometry. (Limited plasticity almost always occurs when a cylindrical basket is placed in a cylindrical cavity because the initial line contact causes very high stresses until the basket edge dents enough to lower the contact stresses.)

A basket for fissile materials (e.g., spent fuel) must simultaneously satisfy the requirements of structural strength, neutron absorption for criticality safety, and heat transport to maintain fuel and basket temperatures within allowable limits. The optimization of these parameters to provide the maximum payload is usually an iterative procedure.

The basket must be fabricated from ASME code materials (or an acceptable alternative code) if it is important to safety. Materials which are not documented to a code should not contribute to the structural strength of a basket that is important to safety because material properties and allowable stress limits have not been established. Extensive documentation of the material properties of a number of metal heats is required before code approval is granted. The evaluation of such data has historically been the domain of organizations such as ASME and the American Society for Testing and Materials (ASTM). Regulatory agencies may choose to accept code materials at their discretion to simplify the certification review process or may evaluate data for noncode materials (e.g., crushable material for impact limiters) in the absence of code-approved materials.

The use of a structural material that also absorbs neutrons has been investigated, especially with the development of a boron-stainless steel alloy which has been granted ASTM approval.[7] Structural alloys that absorb neutrons are desirable for fissile-material casks, and several stainless

steel alloys with boron contents up to 2% have been granted ASTM approval. Aluminum alloys containing boron have been developed and could also be considered for code approval. Ferritic steels with a significant boron content have not been developed because the ferritic metallurgical structure is more sensitive to boron addition than the structure of stainless steels.

A wide variety of construction methods have been used for cask baskets in existing cask designs because of the different types of fuel and materials to be shipped. Baskets constructed from tubes are one of the simplest types. A sheet of structural material is bent into two L-shaped angles, which are welded at the corners. The material selector should consider that a postweld heat treatment is usually not possible because of thermal distortion of the tube. Tubes have also been fabricated by bending sheet metal into two C-shaped channels, which are welded together. This places the weld material of the joint at a location that is more likely to interfere mechanically with a fuel assembly or waste canister inserted into the tube. If a neutron absorber is required for criticality control, then neutron absorber sheets may be welded to the exterior of the tubes, as shown in Fig. 5.5. A benefit of borated structural steel would be the potential for dispensing with attached neutron-absorber sheets, resulting in a more compact, lighter basket. Basket designs that rely upon tight tolerances of square fuel tubes to remove free space and preclude component movement have also been used for storage of spent fuel.

Spent fuel baskets constructed of square tubes with boron neutron absorber sheets may not provide sufficient neutronic isolation to adjacent fuel cells because boron is a thermal neutron absorber and fast neutrons can pass unhindered between adjacent cells. Alternative neutron absorbers are silver-indium-cadmium (Ag-In-Cd) and exotic materials such as gadolinium (Gd) and hafnium (Hf), but the cost of these materials may be prohibitive. One method of improving the neutron-capture capability of a boron neutron-absorber material is to insert gaps between two sheets of neutron-absorber material. One approach to providing a gap is to insert structural spacers between

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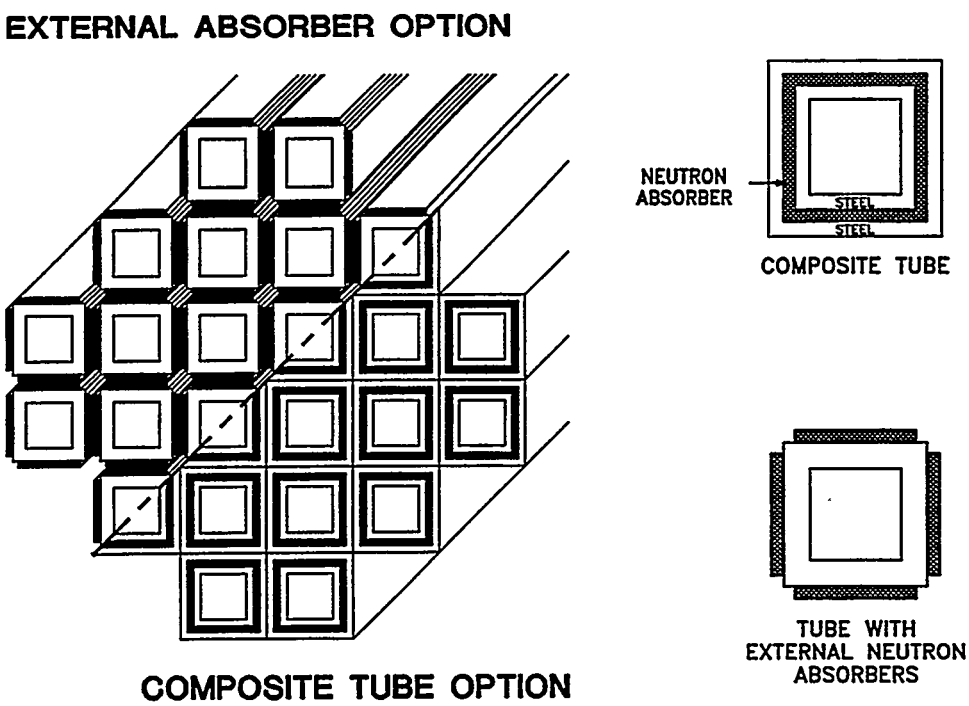

Fig. 5.5. A spent fuel basket fabricated from tubes.

adjacent tubes. Another approach, shown in Fig. 5.6, uses disks with square holes to support the tubes. When the cask basket is filled with water during loading and unloading or in a hypothetical accident scenario, the gap between neutron-absorber sheets contains water, which moderates neutrons. The moderated neutrons are slowed to energies which may be captured by ^{10}B, thus causing the absorption of greater numbers of neutrons and improving the neutronic isolation of adjacent fuel cells. The water gap, sandwiched between two neutron absorber sheets, is often called a "flux trap" because the neutron flux is reduced in the water gap. Regardless of the method of formation of the flux-trap gap, the structure must be capable of maintaining this gap in any drop impact scenario because the fuel assembly and neutron-absorber spacing are important to criticality safety.

The basket geometry must be stable if it is made up of a structure such as support disks, which could conceivably tilt or move during an impact. Baskets which contain structural spacers that create water gaps to act as "flux traps" for criticality safety must not be subject to relative motion of adjacent spacers. Welded or mechanical joints are often used to retain basket components in their proper position.

A third type of basket design, which also employs flux traps, is shown in Fig. 5.7 (ref. 8). This basket was developed as a conceptual design for a multipurpose canister (MPC) that could be used for storage, transport, and ultimate disposal in a geologic repository. The MPC basket uses bent steel angles to form a structural "backbone" for the fabricated tubes which serve to contain fuel assemblies. The flux trap in this basket is formed by the steel angles, which are welded together by spacer bars that create the requisite water gap. Neutron-absorber panels are placed on either side of the water gap to establish the flux trap.

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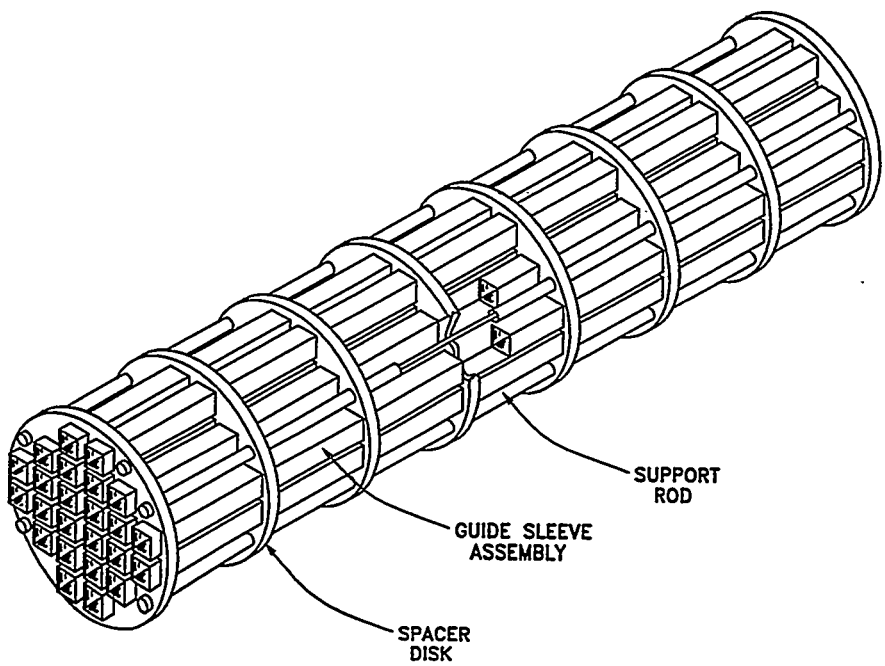

Fig. 5.6. A flux-trap spent fuel basket with support disks.

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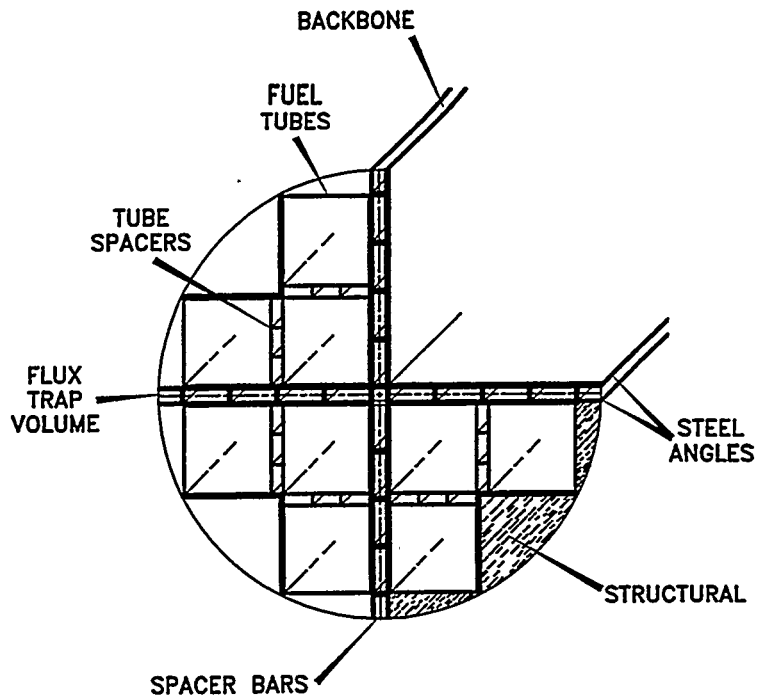

Fig. 5.7. A flux-trap basket with a structural "backbone."

The three basket concepts described above illustrate the relative complexity of the analysis of various structural approachs which can be used to construct spent fuel baskets. Basket designs which have clearly established load paths that are amenable to conventional static analysis techniques are more easily analyzed and tested, and the certification process is facilitated. A comparison of these basket concepts could be summarized as follows:

1. Tube style: full-length, independent tubes.
2. Support disk style: full-length tubes supported by intermittent spacers.
3. Backbone Style: full-length backbone with full-length fuel tubes.

The MPC basket has a full-length structural backbone, which is a continuous weldment, and the load path is well-defined. The structural analysis of such a design is straightforward, and confirmatory testing would be expected to show that the structure is satisfactory.

The support disk basket design introduces a concern regarding the geometric stability of the support disks in the cask axial direction, which must be addressed in the design and structural analysis so that confirmatory tests will be successful. Thus, the structure of this basket design can be more difficult to certify.

It is difficult to assemble the tube-style basket design so that each tube is structurally welded to its adjacent, neighboring tube. If some of the corners of the tube are welded to adjacent tubes while some are not (access to all corner joints is not possible), then a complex load path results, and the structural analyses and confirmatory testing are complicated. If the tubes are not structurally welded, then the analysis is not a conventional static structural analysis because the tubes can move relative to each other in a side-impact scenario. Confirmatory testing of such a

nonwelded basket design becomes even more important, and certification efforts are more difficult than for the other basket design.

Cask baskets must be evaluated for fabrication stresses. The square tubes used in the tube-style basket may be extruded or formed by joining sections formed by bending plates. Residual bending stresses in corners formed by bending should be evaluated. Square tubes may also be formed by extruding round tubes and rolling each tube into a square on a mandrel. Residual stresses in the resulting square corners should also be evaluated. The outer periphery of a basket constructed from square tubes may be rounded by adding fabricated formers or blocks curved on one side. Aluminum or steel blocks are sometimes used to enhance heat transfer from the basket to the surrounding cask wall, but the attachment of such blocks to the basket structure could introduce fabrication stresses.

The ASME code generally considers thermal stresses to be secondary stresses, which are added to normal operation stresses, but may not be added to accident-scenario stresses. However, thermal gradients in spent-fuel-cask baskets can cause significant thermal expansion of the basket at maximum operating temperature. The expansion of the outer diameter of the basket is greater than the expansion of the inner diameter of the cask cavity, and the manufactured gap between the basket and cask body may close. The resulting stresses caused by this differential thermal expansion can be larger than impact stresses and, hence, must be evaluated. A basket may be designed so that it maintains a small gap to the cask cavity at maximum operating temperature to avoid stresses caused by the constraint of the cask body.

Nonuniform thermal gradients within the cask basket itself may cause significant thermal stresses. Even in fully symmetric cask baskets, thermal stresses are created if fuel or radioactive material in adjacent cells generates different heat loads. Thermal stresses must be included in calculations that sum the stresses caused by impact, pressure, and fabrication.

Axial expansion of the basket (and its contents) should also be considered to ensure that an interference with the cask closure lid at maximum temperature does not induce excessive stresses.

Basket structures may be fixed permanently in a cask or may be removable. Removable baskets should be held in place by a structural joint to prevent movement of the basket during end-drop scenarios to avoid loading the cask closure lid and adding to the stresses in the closure lid bolts. A structural joint may be obtained by bolting the basket in place, inserting a retaining ring, or adding structural spacers to occupy free space at the end of the basket. Baskets that are removable should be designed to minimize trapping and buildup of radioactive material ("crud"), and attention should be given to decontamination methods.

5.6 IMPACT LIMITERS

Impact limiter design for casks must be based on a balance between (1) the stresses produced in the cask and cask contents and (2) the available space for the impact limiters. Large, soft impact limiters produce lower deceleration loads on the cask and contents, thus improving structural margins, but such limiters may also be heavier than more compact limiters would be. Impact limiters may be fabricated from crushable materials (such as wood and honeycomb) or from materials with elasticity (such as foam). These materials are not code-documented, and a documentation of tests is needed to establish the material properties under all conditions of transport. Tests of crush or deformation properties are typically required at room temperature,

elevated temperature, and low temperature. If the material is not isotropic, then several angles must be tested.

The design of shock-absorbing structures for the absorption of the kinetic in energy drop is central to modern package design. Smaller Type A packages may not require special impact limiters because the drop energy can be absorbed in elastic deflection of package structures. This is not likely for larger packages. The impact target is an unyielding surface and may not absorb energy, so all of the drop energy goes into deforming the package structure.

The absorption of energy by plastic deformation or crushing of cask containment structures is not permitted. The plastic deformation or crushing of noncontainment external structures, such as Type B package neutron shields, would absorb energy in a cask-drop scenario; however, credit for such energy absorption may not be permitted if such shields contain noncode materials.

Impact limiters designed to absorb the 9-m drop energy must absorb all of the energy before the limiters lose their functionality because of complete crushing or failure of attachments. The impact energy must be absorbed before any structurally rigid portion of the cask (such as trunnions) contacts the unyielding target because such contact would generate very high localized stresses.

Crushable material must be contained within a ductile envelope so that the material remains within the limiter volume during the large deformations that can be caused by the impact. Brittle materials (such as aluminum and the bonding material for Kevlar™ joints) should not be used unless the functionality of the envelope can be demonstrated through testing.

Calculational methods used to predict limiter performance must be qualified through testing at a sufficient number of drop orientations to ensure that predictions at other important angles and orientations will be valid.

Properties of the crushable materials used in impact limiters must be validated through testing at a variety of angles, and for the temperature range to which the limiters would be exposed in normal operation.

All materials store energy during the deformation resulting from an impact, which can be returned to the package during a multiple impact. This stored energy is approximately 5 to 10% for wood and aluminum honeycomb materials, but may be as high as 50% for foam. Energy restoration may be measured in reduced-scale impact limiter tests. (Quarter- and eighth-scale testing of limiters has proved adequate in recent licensing.)

The 9-m drop onto an unyielding surface is generally the most demanding impact of the Hypothetical Accident scenario. The normal-operation scenarios also include a drop of from 0.3 to 1.2 m (depending on mass) onto an unyielding surface in the end- corner- and side-drop orientations. A small package may conceivably be dropped such a distance without defining the impact as an accident, but a 10-ton or heavier cask dropped 0.3 m would be a notable occurence. Large casks dropped less than 0.3 m (1 ft) vertically, or rotated through an arc with less than a 0.3-m vertical component, may be returned to service[9] after such impacts if no damage to the containment (no plastic deformation) occurs and only minor, repairable damage is sustained by the other cask components. The cask designer may thus consider the normal operation drop scenario as a means to establish sustainable impact damage limits which would not result in retiring a package from service.

The absorption of energy by an impact limiter is equal to the work done in crushing a volume. Generally, the energy absorption follows the equation:

$$E = \int F(x)dx \ , \tag{5.2}$$

where E is the total energy absorbed, $F(x)$ is the crush force at crush depth x, and $F(x) = P(\Theta)A(x)$, where $P(\theta)$ is the material crush strength as a force per unit area at the angle of crush determined by the impact orientation, and $A(x)$ is the area of the material being crushed at crush depth x.

5.6.1 End, Corner, and Side Drops

Impact limiters must be designed to absorb impact energy at all possible drop angles, but the designer initially considers the end-, corner-, and side-drop angles listed in 10 CFR Part 71. Impact limiters are usually axisymmetric with regard to the long axis of a cylindrical cask. Cask baskets are often symmetric as well, so that the drop orientation with regard to the cylinder axis does not matter. If the impact limiters or basket are affected by orientation, then impacts at various orientations must also be considered.

The analysis of an end drop is the easiest of the drop angles to evaluate because the crush area is constant throughout the impact. Materials such as aluminum honeycomb shear cleanly at the interface between material which is "backed up" by the incompressible cask and material which is not. Materials such as wood do not shear if the wood grain is at an angle to the shear interface, and material which is not backed up must also be compressed as the impact progresses. This effect adds to the force required to crush the limiter and must be included in calculations of load-deflection curves. The magnitude of the effect is dependent upon the geometry of the impact limiter, and the use of test data gained from crushing small samples may not yield accurate results.

The side-drop analysis is also relatively straightforward because the drop angle is fixed during the impact. For cylindrical cask designs, two impact limiters are crushed in a side drop, but symmetry usually permits the analysis to be performed for one impact limiter that absorbs one-half of the impact energy. The crush geometry is simple for cylindrical impact limiters, and the area that crushes as a function of crush depth is easily obtained from scale drawings. Calculations of impact-limiter performance in the side drop are thus easily performed, unless significant grain-orientation effects or shell-deformation energy must be included.

The corner-drop analysis is similar to the end drop in that only one impact limiter is crushed, but the crush area increases with increasing crush depth as for the side drop. The corner-drop angle is defined as the angle at which the center of gravity of the package is directly above the corner of the impact limiter as it initially contacts the unyielding surface. The specific geometry of an impact limiter causes the crushing area, sometimes called the footprint, to move such that its centroid is no longer located directly below the center of gravity of the cask. For large impact-limiter volumes, this effect can cause the cask to topple over after the corner-drop impact, but the energy which must be absorbed as the second end of the cask strikes the unyielding surface is small compared with the energy absorbed in a side drop. Thus, any secondary impact after a corner drop is bounded by the side drop and may not require a separate analysis or test.

5.6.2 Engineering Tests of Impact Limiters

Impact limiters are superficially simple structures, but may possess complex internal structure or other features which make the performance of the import limiter difficult to predict through calculations alone. Such instances are common, and engineering tests to develop the design of impact limiters may be necessary. The term "engineering test" is used in this chapter to refer to testing used to investigate the stress-strain performance of a prototype impact limiter. Impact limiters may be redesigned several times based upon the results of such tests, so the information gained from early tests may not be relevant to the final design and hence those tests would not be described in the SARP. The test results obtained for the final design are useful in substantiating the performance of the impact limiters and would typically be described in the SARP. These tests may be performed quasistatically and are in addition to dynamic drop testing performed with a scale cask model.

The design procedure for impact limiters may be simplified by designing the limiters to absorb the energy of end and side drops first because these drop angles are easily analyzed and tested. Once the performance of the limiters for the end and side drops has been demonstrated to be adequate, analysis and testing at the corner drop can be pursued with confidence that the methodologies being used are valid.

Quasistatic tests of impact limiters are performed by crushing test articles in a tensile test machine, which is principally designed to test specimens in a tensile mode, but may also be used in the compressive mode. The compressive loading is performed at a slow rate, hence the use of the term *quasistatic*. The measured load-deflection curves for quasistatic tests at end, side, and corner orientations provide the loading conditions needed for finite-element computer analysis of the cask and basket. The load required for each small increment of crush is read directly from the tensile test machine instrumentation, and the deflection is read from an extensiometer attached to the test head as shown in EPRI NP-7389 (ref. 10). These tests may be used to demonstrate that a computer code is properly predicting the load-deflection curve for a given impact limiter design so that testing at angles between the corner and side drop is not necessary.

Tests performed quasistatically must be adjusted to dynamic conditions by correcting the measured load-deflection curve for the dynamic-to-static load ratio. This ratio is measured by crushing small test samples both dynamically and statically (10-cm cubes or cylinders of material are sometimes tested). Each data point on the quasistatic load-deflection curve is then multiplied by the ratio of the dynamic crush force to the static crush force. (Note that the dynamic-to-static crush force ratio may change as a function of angle.) The resulting load-deflection curves are the basis for finite-element analysis of a cask under drop conditions.

Quasistatic tests also measure the elastic energy storage "rebound" parameter of an impact limiter design. Impact limiters absorb energy through permanent, plastic deformation of the limiter volume; however, a portion of the total absorbed energy must be used to produce the elastic compression of the absorbing medium and the impact limiter outer shell. This elastic component of the absorbed energy is restored to the cask after the impact is complete, and the cask end is stopped. The rebound energy may then add to the severity of a secondary impact in a "slapdown" scenario. Rebound energy is a function of the materials and geometry of the impact limiter and is very difficult to calculate accurately.

5.6.3 Oblique Drop

Oblique drop angles are all angles between the corner- and side-drop angles. It is not necessary to evaluate every angle between the corner (typically 15 to 25°) and side-drop; current practice is to evaluate the drops in 15° increments. Because oblique drops do not impact the target with the center-of-gravity directly over the point of impact, the impact limiter on one end of the package will strike the ground first and start to crush; the cask will then start to rotate to a nearly horizontal position, at which time the second impact limiter will contact the target. This secondary impact will be similar to an impact of the package in a side drop—except that the energy which must be absorbed may be different and the position of the package will not be quite horizontal.

One of the most difficult analyses for cylindrical packages is evaluating the effect of a slapdown following an oblique drop of the package on an unyielding impact pad. In this scenario, the package is dropped at a shallow angle, and the lowest edge of the package contacts the impact pad and is crushed as the first end of the cask is stopped. The center-of-gravity of the package continues its descent, and this action causes the opposite end of the package to rotate, adding to its downward velocity. In addition, the rebound energy stored in the first impact limiter may be restored to the first end of the cask, adding to the rotational energy. In some cases, the impact velocity of the second end of the package can be significantly higher than the impact velocity of the first end.

To provide some insight into the methods of analysis of this phenomenon, the analysis of the Nuclear Assurance Corporation (NAC) Spent Fuel Legal Weight Truck (LWT) Cask is presented. This work can be found in Docket No. 9225 in the Public Document Room of the NRC.[11]

The NAC LWT cask was analyzed, and a scale model was tested for the secondary impact resulting from a 9-m oblique drop (hypothetical accident condition) onto a flat, unyielding, horizontal surface.

The NAC LWT cask in its slapdown geometry is shown schematically in Fig. 5.8. This analysis assumes that the impact limiter is crushed by a 0.3-m fall before the 9-m accident begins, for a total of 9.3-m. At the time that this analysis was performed, it was uncertain if the normal operational 0.3-m fall requirement and the 9-m fall hypothetical accident requirement were cumulative or independent; it was later determined that these requirements are independent, and not sequential. This section addresses the primary impact loads on the cask structure and the adequacy of the impact limiter to absorb the remaining energy during the secondary impact on the unyielding surface for an oblique cask orientation based on test results.

The total amount of energy (E_T) to be dissipated after the cask has fallen is calculated as:

$$E_T = E_1 + E_R + E_P + E_S \quad , \tag{5.3}$$

where

E_1 = energy absorbed during the first impact,
E_R = remaining energy after the first impact,
E_P = rotational potential energy, and
E_S = elastically stored energy.

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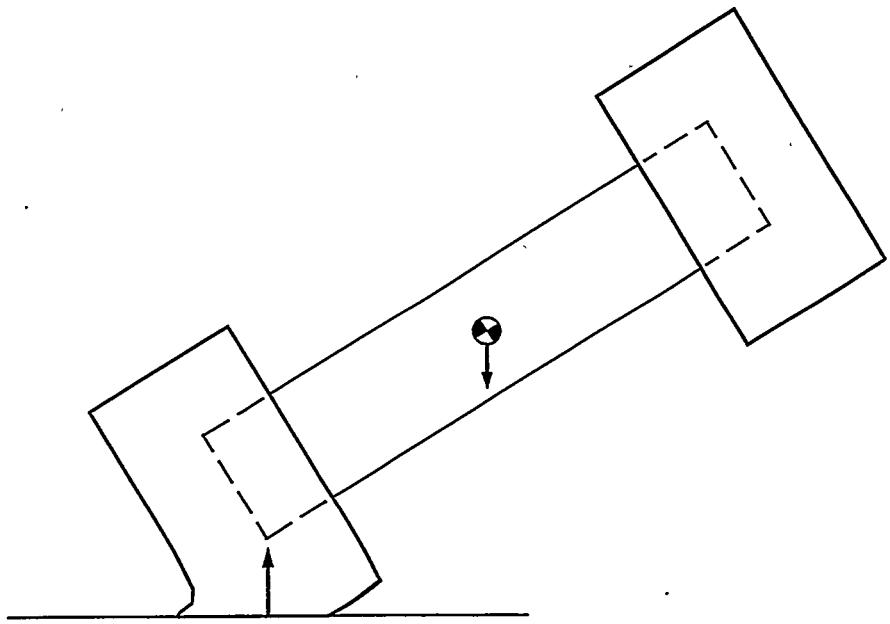

Fig. 5.8. Cask in a slapdown geometry.

The sum of the last three terms equals the energy absorbed by the second impact limiter (secondary impact). Terms E_1, E_R, and E_P can be calculated; however, testing of the aluminum honeycomb in an impact-limiter configuration was necessary to determine E_S.

A quarter-scale model of the NAC LWT cask (Fig. 5.9) has been tested, and the secondary impact phenomenon has been reviewed. It was concluded that the rebound energy (5 to 10% of the primary impact energy) may be restored to the cask in such a manner as to cause the first impacted end of the cask to lift from the ground several inches during the secondary impact (slapdown). Thus, the restored rebound energy must be dissipated by the second impact limiter.

Table 5.1 shows that at a drop angle of 75°, the energy dissipated in the secondary impact (E_2) is greater than is the energy dissipated in a secondary side impact; consequently, that the cask "slaps down." This slapdown occurs because the residual drop energy not absorbed in the first impact (energy transformed to rotational energy), impact-limiter rebound energy (elastic rebound), and potential energy converted to kinetic energy as the cask rotates to the second impact combine to exceed the energy absorbed by one limiter in a pure side-drop orientation. Results of the scale impact limiter tests show that the impact limiters do have sufficient energy dissipation margin to absorb the 75° oblique secondary impact energy.

A simplified example of the effects of slapdown is to compare the equivalent drop energies of several scenarios. When a cask is dropped in the horizontal orientation, the potential energy of a 9-m height is converted into kinetic energy which must be absorbed equally by the two impact limiters; therefore, each impact limiter must absorb the energy equivalent of a 4.5-m drop. The design process for impact limiters must ensure that each impact limiter can absorb this 4.5-m drop energy equivalent. When a cask is dropped at an oblique angle, the equivalent energy of a 9-m drop is distributed unequally to the two impact limiters; the first limiter to impact the ground

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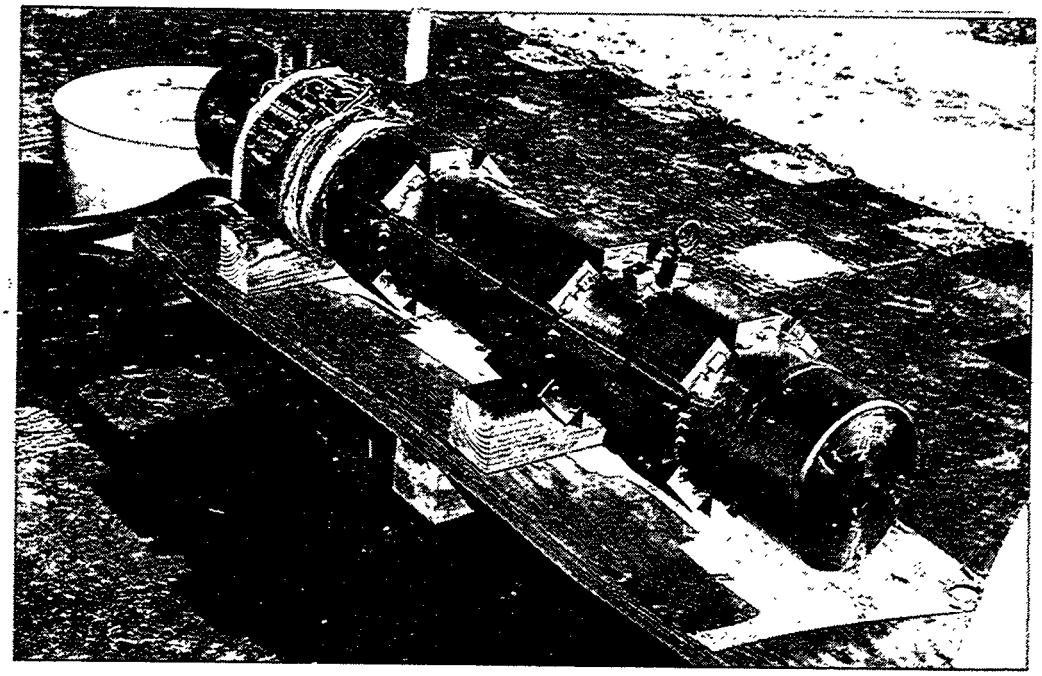

Fig. 5.9. The quarter-scale model of the NAC-LWT cask. Courtesy of NAC International, Norcross, Ga.

absorbs a greater percentage of the total so that the second impact limiter absorbs something less than a 15-ft energy equivalent. The second impact limiter strikes the ground in a near-horizontal position as the cask rotates from the initial oblique angle, and because the remaining energy for the second impact is less than a 4.5-m fall equivalent, the impact limiter energy absorption capacity would be adequate. However, the second impact has additional energy from the rotation of the center of mass of the cask from its elevated position (during the first impact) plus the rebound energy stored during the first impact. These additional sources result in more than a 4.5-m fall energy equivalent being delivered to the second impact limiter. Hence, impact limiters must be designed with more than a 4.5-m fall equivalent energy absorption capability to accomodate the slapdown.

Note that the value of E_1 in Eq. (5.3) can be calculated by summing the products that are determined by multiplying the incremental volume of impact limiter that is crushed by the force required to crush that volume. The summing process can be relatively straightforward if the package and its limiter impacts a surface end on and the size of the impacting "footprint" changes very little with time during the event. However, in a corner drop, or if a package is tested in a slapdown orientation, both the size of the footprint and the crush force change significantly while the impact limiter is crushing. Because of this, NAC developed a program called RBCUBED to determine the change in the footprint as the crushing of the limiter progresses. The RBCUBED program is a proprietary code, but a public domain version of this code called FADE (Force And Deflection Evaluation) is documented in EPRI NP-7389 (ref. 10). Within the computer program the value of E_1 is obtained by solving the equations of motion for the package and applying information on the force required to crush the impact limiter, which had been obtained in controlled crush tests on scale models of the limiter. The limiter force is equal to the crush strength of the aluminum honeycomb multiplied by the backed-up crush area. The program calculates this area as a function of impact angle and crush depth using a system of solid

Table 5.1. Determination of maximum energy remaining for secondary impact—full-scale impact limiter
[Units expressed in joules (in.-lb)]

Energy	Drop angle (degrees)						
	0	15	30	45	60	75	90
E_1 Energy absorbed by first limiter	2.18×10^6 (1.93×10^7)	2.13×10^6 (1.88×10^7)	1.97×10^6 (1.74×10^7)	1.72×10^6 (1.52×10^7)	1.39×10^6 (1.23×10^7)	1.17×10^6 (1.03×10^7)	1.09×10^6 (9.67×10^6)
E_R Energy remaining after first impact	0.00 (0.00)	6.10×10^4 (5.40×10^5)	2.19×10^5 (1.94×10^6)	4.68×10^5 (4.14×10^6)	7.96×10^5 (7.04×10^6)	1.02×10^6 (9.02×10^6)	1.09×10^6 (9.67×10^6)
E_P Potential energy of cask after first impact	6.89×10^5[a] (6.04×10^6)	5.69×10^5 (4.97×10^6)	5.04×10^5 (4.40×10^6)	3.90×10^5 (3.45×10^6)	2.50×10^5 (2.21×10^6)	8.68×10^5 (7.68×10^5)	0.00 (0.00)
E_S Energy stored in first limiter	2.11×10^5 (1.87×10^6) (9.7%)	1.40×10^5 (1.24×10^6) (6.6%)[b,c]	1.18×10^5 (1.04×10^6) (6.0%)[b,c]	9.79×10^4 (8.66×10^5) (5.7%)[b,c]	7.65×10^4 (6.77×10^5) (5.5%)[b,c]	6.30×10^4 (5.56×10^5) (5.4%)[b,c]	5.80×10^4 (5.13×10^5) (5.3)[b]
E_2 Secondary impact; Total of ER + EP + ES	8.94×10^5 (7.91×10^6)	7.63×10^5 (6.75×10^6)	8.34×10^5 (7.38×10^6)	9.56×10^5 (8.46×10^6)	1.12×10^6 (9.93×10^6)	1.17×10^6 (1.03×10^7)	1.15×10^6 (1.02×10^7)
E_{max}, side-drop maximum energy absorption capability	1.68×10^6 (1.49×10^7)	1.68×10^6 (1.49×10^7)	1.68×10^6 (1.49×10^7)	1.68×10^6 (1.49×10^7)	1.68×10^6 (1.49×10^7)	1.68×10^6 (1.49×10^7)	1.68×10^6 (1.49×10^7)
Energy absorption margin	88.32%	120.71%	101.79%	76.20%	50.10%	44.04%	46.33%

[a]Assumes tipover of cask onto the second impact limiter. Cask standing on end is unstable.
[b]Percent of initial impact energy.
[c]Interpolated values.

geometry subroutines developed by Oak Ridge National Laboratory as part of the MORSE shielding code. These crush area calculations were verified by manual calculation of crush area using graphical drafting techniques for several impact angles and crush depths.

The accuracy of the area calculation may be seen by an inspection of the scale limiter results presented in Figs. 5.10 through 5.12, which show that the calculated force (the crush area times the crush strength) accurately tracks the measured force. (Note that the figures were originally published using English units and are reproduced here.)

The impact limiter force is perpendicular to the unyielding surface and is applied to the cask body, as shown in Fig. 5.13. The weight of the cask continues to accelerate the cask downward as the impact limiter crushes, thus producing a crush force (F_d), decelerating the cask. The difference between these forces is the net force (F_{net}).

The net force applied to the cask produces a force and deceleration parallel to the cask's long axis and a force and angular acceleration perpendicular to the cask axis, as shown in Fig. 5.13. The parallel force component acts on the cask center of mass to slow the cask, but the perpendicular component transforms translational kinetic energy from the drop into rotational energy that must be absorbed in the secondary impact.

The calculations to determine the energy to be dissipated for each drop angle shown in Table 5.1 include the potential energy resulting from the cask tipping over (for $0° \leq \theta \leq 15°$) and elastically stored rebound energy from the first impact.

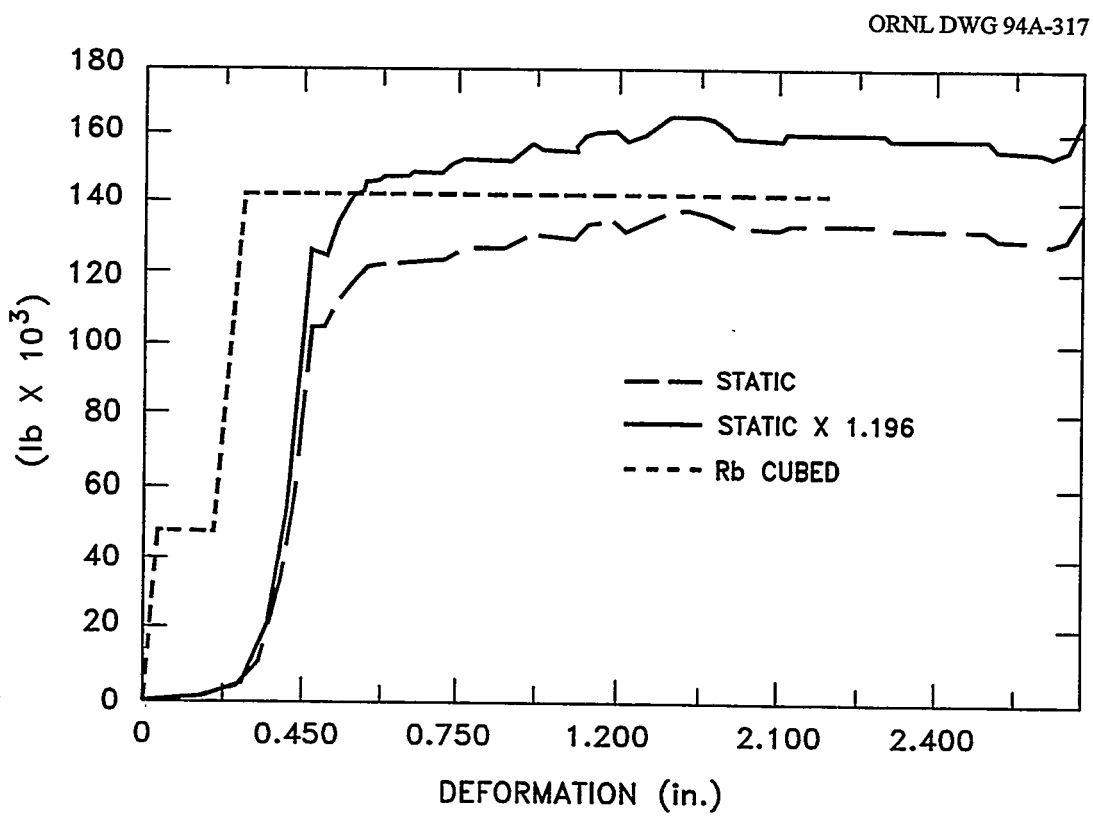

Fig. 5.10. The force-deflection curve of drop-tested limiter—0° impact. Used by permission of the Electric Power Research Institute, Palo Alto, Calif.

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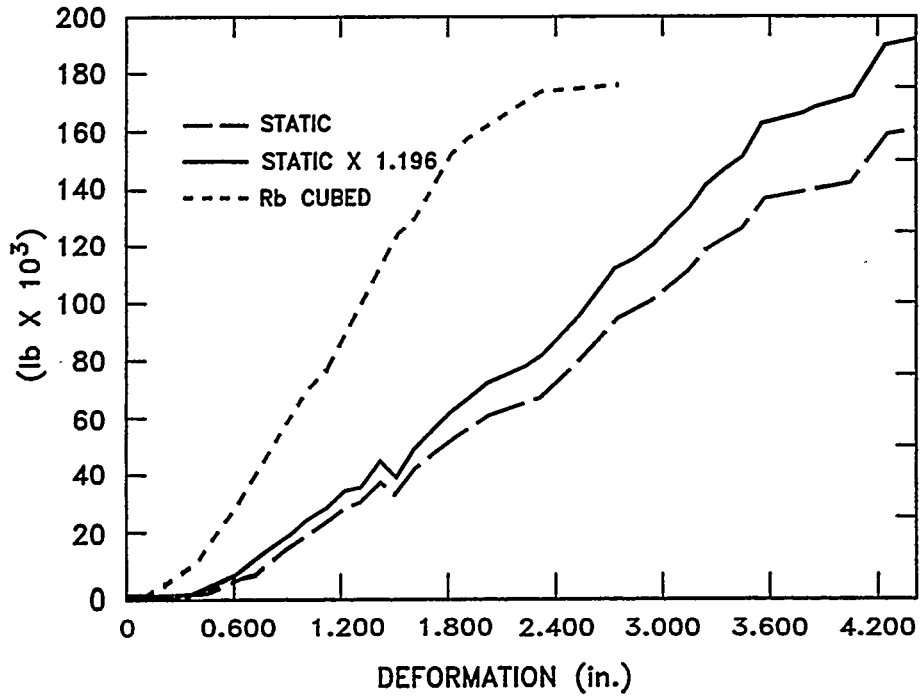

Fig. 5.11. The force-deflection curve of drop-tested limiter–60° impact. Used by permission of the Electric Power Research Institute, Palo Alto, Calif.

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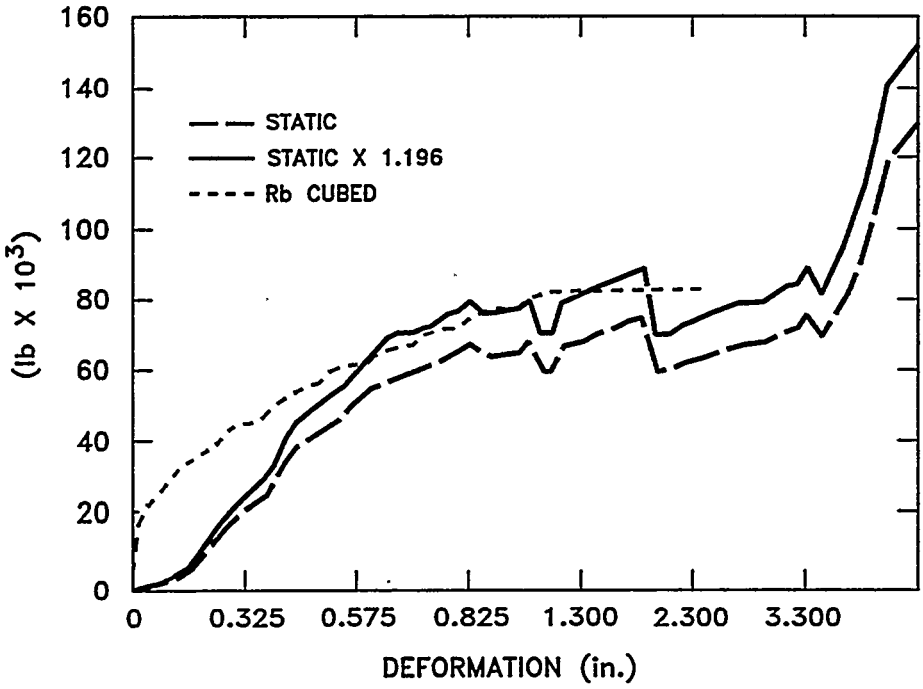

Fig. 5.12. The force-deflection curve of drop-tested limiter—90° impact. Used by permission of the Electric Power Research Institute, Palo Alto, Calif.

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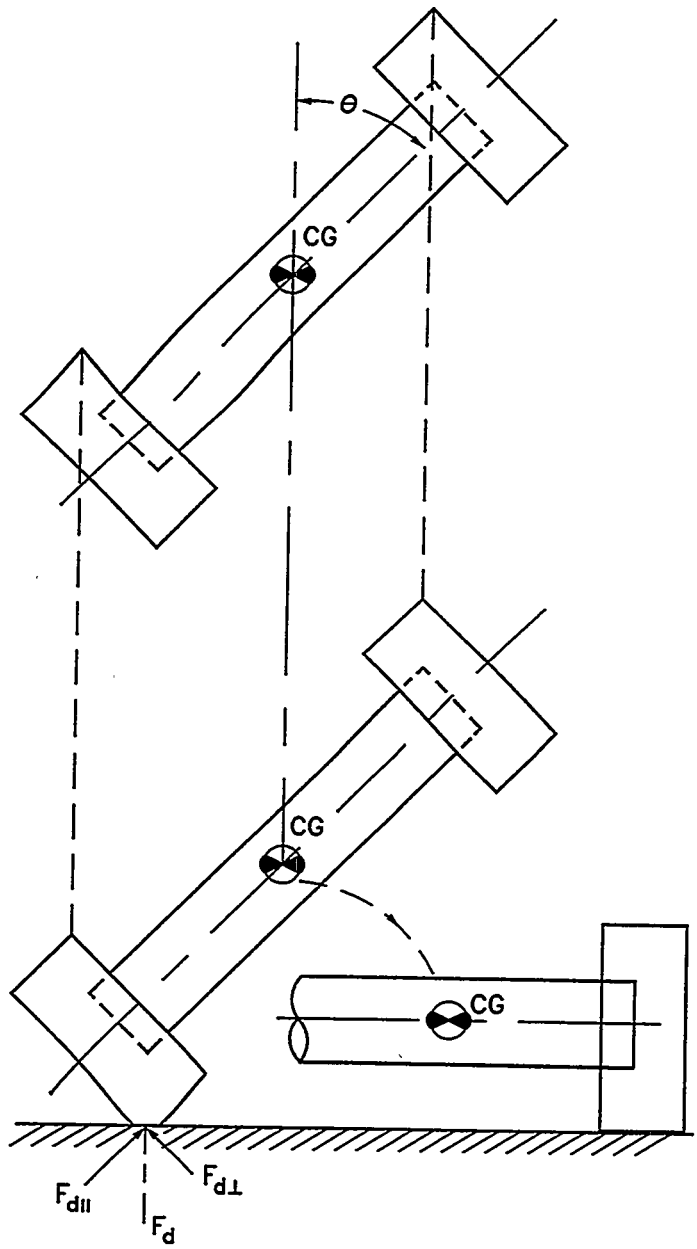

F_d = DECELERATION FORCE

$F_{d\parallel}$ = PARALLEL COMPONENT OF THE DECELERATION FORCE

$F_{d\perp}$ = PERPENDICULAR COMPONENT OF THE DECELERATION FORCE

Fig. 5.13. Forces applied to the cask during an oblique drop.

5.6.4 Energy Calculation

The cask and impact limiters are considered to be a mass (m). When the mass is released from rest at a height (H), it is accelerated uniformly by gravity (g). Because the mass is not acted upon by off-center forces while free falling, it will remain in the same attitude that it had when it was released. The change in potential energy equals the change in kinetic energy. The vertical velocity of the mass at the time of contact with the unyielding surface is calculated as:

$$V_i = \sqrt{2Hg} \ , \tag{5.4}$$

where

V_i = initial vertical velocity of cask at time of impact (t = 0 s), (m/s),
g = the gravitational constant (m/s^2), and
H = the drop height (m).

The cask may impact at any angle (0° to 90°). In the case of impact angles from 0° (end drop) through approximately 15° (corner drop), the cask could be expected to remain upright after the total energy of the first impact is absorbed; as a result, the center of gravity does not have a moment arm, which would provide a rotational moment about the cask base, causing the cask to tip over.

However, the calculated energies shown in Table 5.1 do include the energy of the tipover for conservatism. Ignoring elastically stored energy, which is absorbed by one impact limiter, the total kinetic energy is equal to the potential energy, that is:

$$E_T = mgH, \ 0° \leq \theta \leq 15° \ , \tag{5.5}$$

where

E_T = total energy to be absorbed (J),
m = mass of the cask and limiters (kg),
g = acceleration due to gravity (9.80 m/s^2), and
H = height of drop (9 m).

For oblique drops within the angular range from corner drop to side drop, the total energy absorbed is greater than that for the 0 to 15° range, if one ignores cask tip over and elastically stored energy. Cask impact angles greater than 75° are considered to be side drops because the first impact limiter stops the cask as the second begins to absorb energy.

Oblique drops (15° $\leq \theta \leq$ 75°) have four constituent energy-dissipation components, which account for all the potential energy given to a cask before it is dropped. Oblique drops are two-step phenomena. First, the cask vertically translates toward the unyielding surface. Then, after impact, the lower impact limiter decelerates the lower end of the cask. The amount of energy dissipated while stopping the vertical translation of the lower end of the cask is E_1. Next, the cask pivots on the crushed impact limiter until the cask is nearly horizontal, and the second impact limiter begins to crush. The remaining three constituent energy components (remaining energy

from the free drop, E_R, which is transformed to rotational energy; potential energy from pivoting to the horizontal, E_P; and elastically stored energy, E_S) are all absorbed in the second limiter in the side-drop orientation.

Thus, in oblique drops, the center of gravity will fall a distance greater than the drop height, H. The additional distance the center of gravity must fall is $L/2$ (cos θ) (Fig. 5.19), where L is the length of the cask body. Therefore, the total kinetic energy that must be absorbed to bring the cask to rest for angles greater than 15° is

$$E_T = mg\,[H + (L/2)\cos\theta],\ 15° \le \theta° \le 90° . \tag{5.6}$$

The total energy to be absorbed in a single impact limiter is greatest for the 0° cask impact angle because the first impact limiter stops the cask without the participation of the second impact limiter. For angles greater than 15°, the first impact limiter stops the lower end of the cask, and then the second impact limiter absorbs the other energy components ($E_R + E_P + E_S$). The total energy absorption is equal to the sum of E_1 and E_2 for each drop angle.

5.6.5 Rotational Velocity Change

For cask angles greater than 15°, the center of gravity of the cask is unsupported; this is considered a slapdown impact in which one end of the package hits first and the opposite end then slaps down on the impact surface. The net crush force is applied at one end of the cask, resulting in a torque (T) and then causing rotation of the cask about its center of gravity. As the cask is translating vertically while decelerating, it is also rotating around the impact limiter that is crushing. The cask is also attempting to rotate as a rigid body around its center of gravity because the torque is applied to the decelerating end of the cask by the perpendicular component ($F_{d\perp}$) of the crush force. The applied torque is

$$T = F_{d\perp}\,(L/2) = F_d \sin\theta\,(L/2) , \tag{5.7}$$

where

T = torque [m-newton(N)],
L = cask length (m), and
F_d = deceleration force (N).

The impulse equation for an applied torque (averaged over the time increment) is

$$T\Delta t = I\Delta\omega , \tag{5.8}$$

where

Δt = increment of time (s),
I = moment of inertia of a cylinder (cask) about its center of gravity, $mL^2/12$, and
$\Delta\omega$ = change in angular velocity (rad/s).

The angular velocity is equal to the rotational velocity change (Δv_t) divided by the radius; that is,

$$\Delta\omega = (\Delta V_t)/(L/2) \ . \tag{5.9}$$

Substituting for $\Delta\omega$ in the impulse equation,

$$T\Delta t = I(\Delta V_t)/(L/2) \ . \tag{5.10}$$

Substituting and solving for ΔV_t ,

$$\Delta V_t = (F_d \sin\theta \ \Delta t \ L^2)/(4I) \tag{5.11}$$

Substituting the formula for I_{cofm} (the moment of inertia about one end) for a cylinder yields

$$\Delta V_t = (F_d \sin\theta \ \Delta t \ 3g)/(W_T) \ . \tag{5.12}$$

(Note that the same result for ΔV_t can be obtained by regarding the cask as rotating about its center of mass because the equations of motion are independent of the frame of reference.)

This change in transverse velocity is subtracted from the transverse component of the initial velocity to determine the transverse velocity of the cask at the beginning of the next deformation step (δ'). When the sum of the transverse velocity changes equals the initial velocity, the impacting end of the cask has been stopped along an axis perpendicular to the longitudinal axis of the cask.

Some mention must be made of the behavior of the interface between the first impact limiter to impact and the impact surface itself. The cask applies a force to the impact limiter, which is resisted by the impact surface, which causes the crushing of the impact limiter. At the same time, the outer shell of the impact limiter is pressed against the impact surface. The pivoting of the cask as it rotates about the contact point is made possible by the friction between the unyielding surface and the impact limiter shell. If the impact surface were frictionless, the impact limiter would slip out from under the falling cask, and the cask would rotate about its center of mass, so that very little energy would be absorbed in crushing the impact limiter. This in turn would cause the second impact limiter to absorb a greater portion of the kinetic energy. Tests have shown that this effect does not occur, although some scuffing of the impact limiter skin and deformation of the impact limiter in the direction parallel to the impact surface may occur.

5.6.6 Load Application

The net force produced by elastic deformation and crushing of the impact limiters is applied to the cask body in a direction perpendicular to the unyielding impact surface. The total force may be represented as a collection of force vectors applied to the cask body, which induces stresses in the cask. The modeling of these forces affects the resulting stresses, and accuracy in the load application is thus very important.

The load produced by deformation of the impact limiters in a side-drop orientation is shown in Fig. 5.14. The force vectors are normal to the unyielding impact surface regardless of the impact limiter design. The total force is distributed over an effective arc (2 Θ), which approaches 180°. The actual arc may be determined from scale testing. One method is to paint the inner surface of

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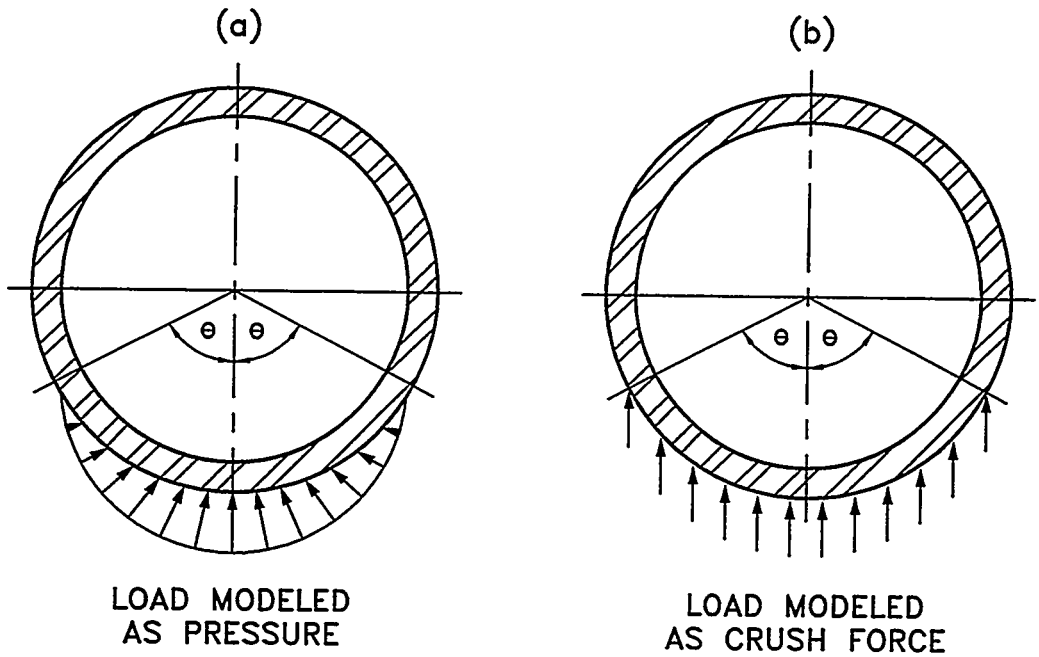

Fig. 5.14. Impact limiter load distribution.

the impact limiter so that the crush forces scrape the paint at a point where force is applied. If the crush material properties are isotropic, then the force vectors are constant over the entire arc, as shown in Fig. 5.14(b). Such a load distribution may not be convenient to model with some finite-element computer codes, but accurate modeling will result in accurate stress calculations.

A modeling alternative is also shown in Fig. 5.14, in which the actual uniform loading is represented as a radially applied pressure. In this example, the crush force is assumed to be greater at the area of initial contact with the unyielding surface. A sinusoidal distribution is used, and the vector magnitudes are determined by integrating the vertical force components over the surface area of the load arc. The integral must equal the maximum force produced by the impact limiter at any time during the impact event. This modeling approach overestimates cask stresses because the radial pressures apply a compressive horizontal load to the cask body, which is not part of the applied load. The cask stresses are also overestimated because the stresses in the region of highest pressure are enhanced. It is prudent to check the modeled force distribution to ensure that no portion of the load arc is subjected to a force greater than the crush force times the area involved. The limiter cannot generate a force vector greater than this value.

The load-deflection curve produced by any impact limiter is determined by the crushable material properties and the limiter geometry. For side- and angle-drop orientations, the impact limiter begins to crush at the point of contact between the limiter and the unyielding surface. Material that is pressed against the cask body by the impact is "backed-up" by the cask body. Crushable material that is not backed up tends to displace away from the unyielding surface instead of crushing. Crushing progresses from the outside inwards until the area of crushing equals the maximum projection of the cask body area, and the maximum deceleration force is produced. Crushing may then begin at the interface between the cask body and the interior surface of the

impact limiter. The actual progress of the crush plane at that time depends on the details of impact limiter construction and geometry and should be determined through testing. Figure 5.15 illustrates the crush plane propagation.

The crushing in an end-drop orientation may progress differently, with the cask punching into the limiter material. This occurs because the area that would be crushed at the interface between the unyielding surface and the limiter exceeds the area of crush at the interface between the cask body and the limiter.

When material is crushed, a force-per-unit-area-crushed is produced. A residue remains after the crushing because the material becomes essentially incompressible when the void area within a volume has become collapsed. The residue thickness is sometimes called the stack height, and may be 30 to 50% of the original thickness. Work is done only as the crush plane traverses the compressible portion of the material, and limiter thickness occupied by the residue does not contribute to energy absorption. This effect reduces the effective crush stroke of the impact limiter by the stack height percentage.

Impact limiters are usually contained within a shell, which protects the crushable material from the detrimental effects of moisture and minor impacts. This shell material can contribute to the containment of the crushable material in an impact if the shell is made from a ductile material, such as stainless steel. Some crushable materials, such as aluminum honeycomb and foam, tend to hold together during an impact. Others, especially wood, tend to fracture and disperse with relatively little energy absorption unless they are contained. Joints between sections of aluminum honeycomb or foam may separate during a dynamic impact because the shear strength of the joint adhesive is less than the strength of the surrounding material. The rapid progression of a dynamic

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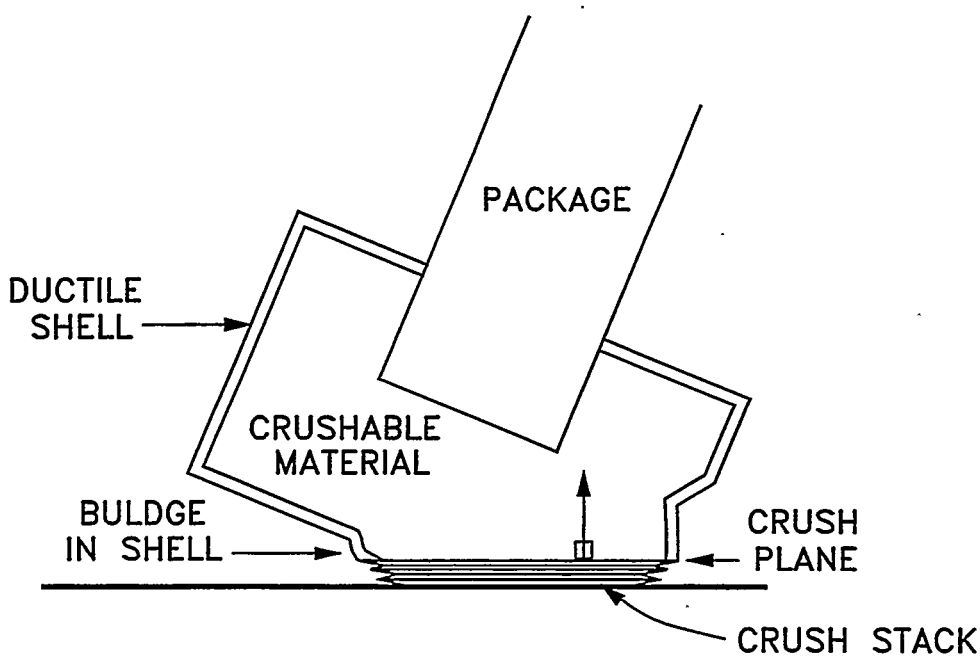

Fig. 5.15. Impact limiter crush plane.

impact may prevent significant joint separation from occuring because inertia tends to restrain the joined volumes in place. Brittle materials, such as aluminum, materials used in Kevlar™ joints, offer high strength-to-weight ratios but cannot contribute to the containment of crushable materials. The performance of the shell, together with the crushable material selected for the limiter, is best determined through testing.

5.6.7 Impact Limiter Attachments

Impact limiters must be firmly attached to the cask during the 9-m impact to fulfill their function. Attachments must withstand impact loads if they retain the limiter on the cask body during an impact. Limited plastic deformation of impact limiter-attachments is common in cask-drop scenarios because of the exposed position of many attachment designs, as illustrated in Fig. 5.16. Testing is generally required to show that any such deformation does not degrade the functionality of the attachment and that the impact limiter remains attached during an impact. Attachment designs that provide a protected location for the attachment are illustrated in Fig. 5.17.

Impact limiters, which are designed so that overturning loads (which would tend to remove the limiter during an impact) do not occur, do not require analysis of the limiter attachments under accident loads, although they must be evaluated under normal operation loads. This situation occurs because the impact limiter would be forced onto the cask body at all angles of impact, as illustrated in Fig. 5.18. Accident load performance must be demonstrated with scale-model tests for such limiters. An impact-limiter configuration that results in overturning moments is a cantilevered impact limiter, as illustrated in Fig. 5.19.

The behavior of limiters and attachments during the transition from normal operation to a drop-accident scenario may not be evaluated because this transitional scenario is not (and perhaps cannot be) defined. Nevertheless, the attachments and limiter should not be vulnerable to excessive damage caused by the failure of the cask support structure at the onset of an accident.

5.7 LIFTING AND ROTATION TRUNNIONS

Cask lift trunnions and tiedowns must be sized to NUREG-0612 (6 *g* on yield and 10 *g* on ultimate) as well as 10 CFR Part 71 (3 g on yield) if fuel is shipped. Lifting trunnions must withstand 3-*g* loads to satisfy 10 CFR Part 71, but the higher 6 g (yield, 10 *g* on ultimate) of NUREG-0612 may govern if the cask contains spent fuel or if a radioactive materials cask is operated over or near plant equipment that is important to safety. The tiedowns and cask support structure must be capable of withstanding the *g*-loads of normal transport without yielding. Transport by rail requires design *g*-load factors of 10 for longitudinal, 4 for horizontal and 5 for vertical. Transport by truck requires *g*-load factors of 2 (all axes) if the truck trailer has an air-ride suspension, and 4 if metal springs are used.[11]

Trunnions may be external (typically round protrusions, as shown in Fig. 5.20) or internal (pockets). Internal trunnions offer the advantage of a reduced outer diameter for impact considerations. The crushing of an impact limiter in a side-drop orientation may be restricted to prevent impact of an external trunnion. Internal trunnions have the disadvantage that the pins must be strongly supported to resist the cantilever bending loads applied during lifting or supporting the cask. Lifting yokes and support structures for internal trunnions tend to be heavy as a result.

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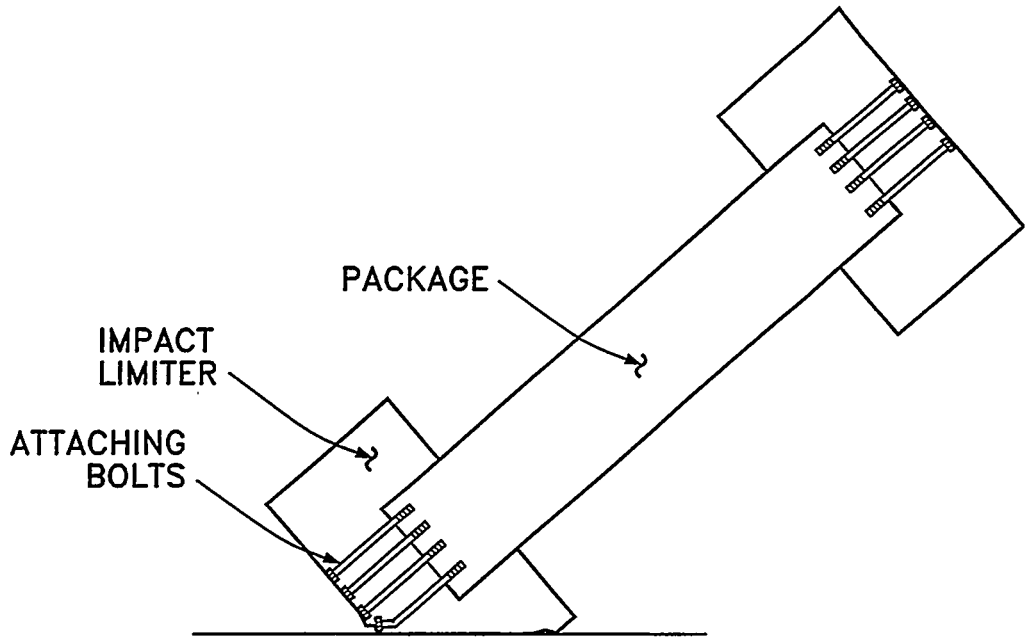

Fig. 5.16. Exposed impact-limiter attachments.

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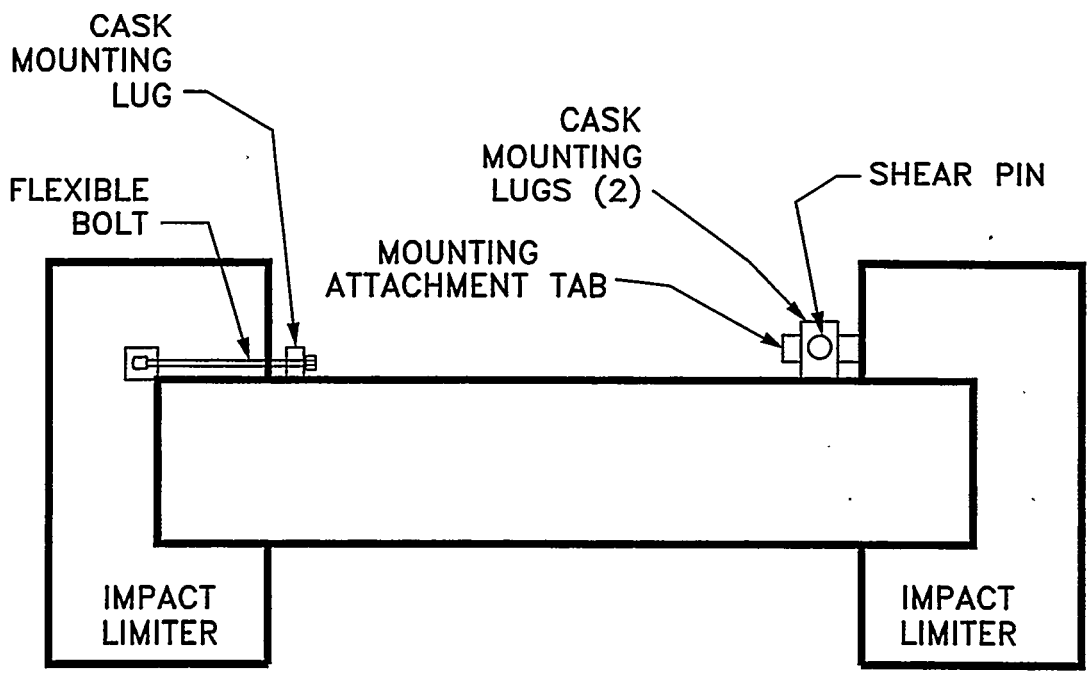

Fig. 5.17. Protected impact-limiter attachments.

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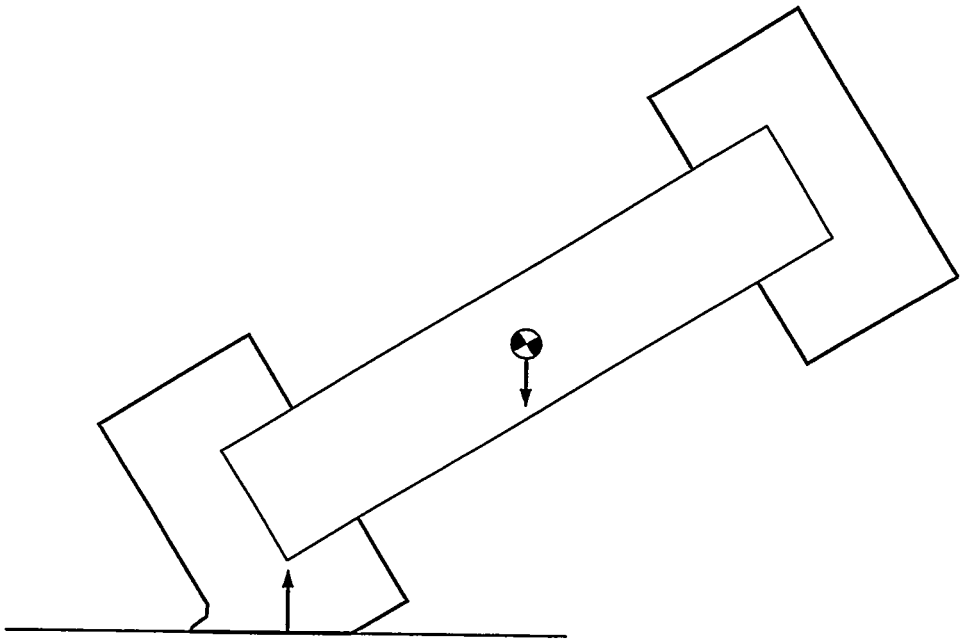

Fig. 5.18. Impact-limiter moments.

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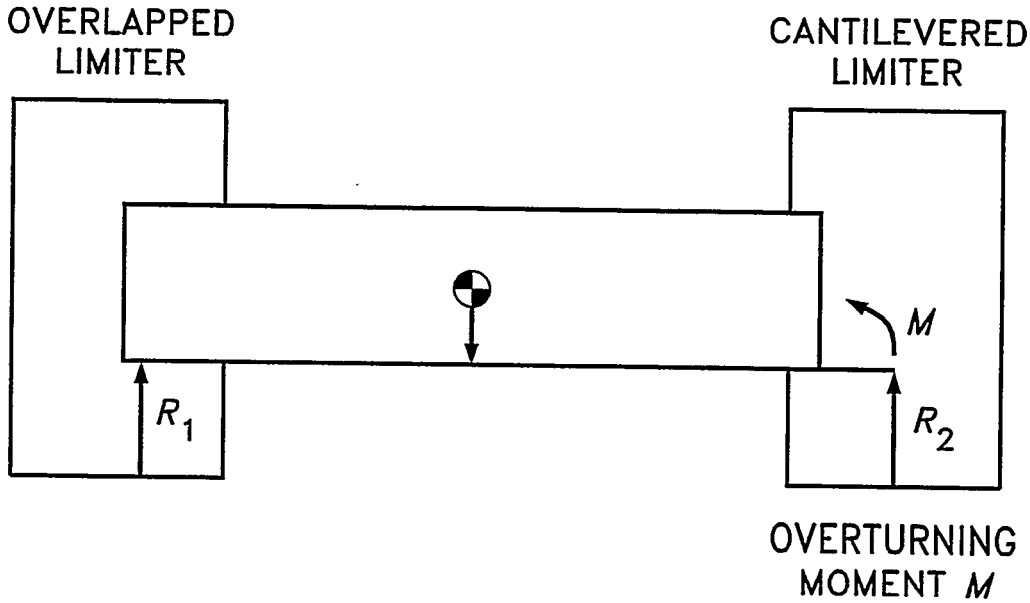

Fig. 5.19. Impact-limiter free-body diagram.

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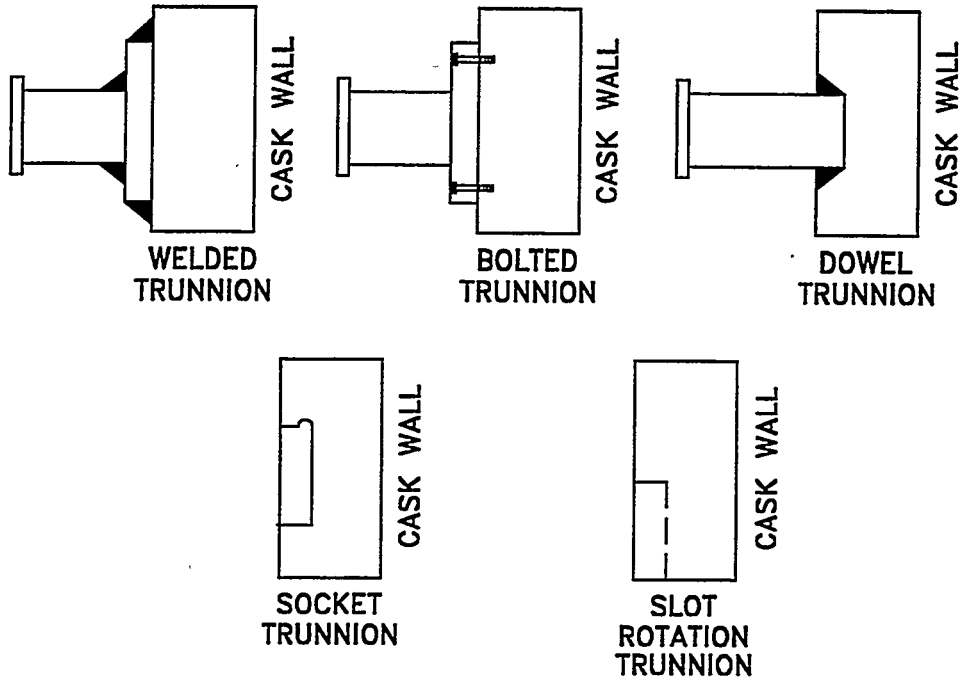

Fig. 5.20. Types of trunnion designs.

Trunnions that are bolted in the cask body must be analyzed for uneven loading of the bolts. The effect of the removal of material to provide bolt holes on trunnion mounting plates must be included in the analysis of the mounting plate. Bolted trunnions are favored when it is desirable to remove the trunnions for replacement or refurbishment.

Welded trunnions are generally lighter in weight than are bolted trunnions, but the former are not removable. If wear is a significant concern, then a bronze sleeve may be provided in the lifting-yoke apparatus to provide a removable wear surface.

External trunnions may be solid or hollow. Large-diameter trunnions do not benefit significantly from the central core material of the trunnion (the moment of inertia of the cylindrical beam is dominated by the exterior fibers) and are often hollow to reduce weight. The hollow may be filled by shielding material to replace shielding that was removed to facilitate trunnion installation (i.e., to alleviate interference between the trunnion and a neutron or gamma shield).

Large casks are usually handled in the vertical position, but they are shipped in the horizontal orientation to provide acceptable package heights for bridge overpasses and tunnels. The rotation of the cask from the vertical to horizontal position may be achieved via rotation trunnions near the cask bottom-end or by a cup-like structure.[13] Rotation trunnions may be internal (socket or slot type) or external. Interference between the rotation trunnions and the bottom-end impact limiter often presents a difficult design problem.

5.8 CASK SUPPORT STRUCTURE AND TIEDOWNS

The cask support structure must maintain the cask in the desired position during all normal operation transport conditions. The cask support structure is permitted to fail during hypothetical accident conditions. The cask support structure and tiedowns are part of the package evaluated under 10 CFR Part 71. Tiedowns, which bind the cask to the cask supports, must not yield under normal operating conditions or before the cask support structure yields under accident conditions. This nonyielding of the tiedowns ensures that the cask does not separate from the cask support structure until accident loads are developed. Cask support structures must not yield as a result of structural loads operated in normal operations. Spatial interference with impact limiters is a common design problem.

The interface between cask support structures and the transport vehicle must be evaluated to ensure that the vehicle structure to which the supports are attached is adequate for normal operation loads. Although truck trailers are not expressly regulated by 10 CFR Part 71, drawings of approved trailer designs must be included in the cask SARP. Consideration must also be given to the ride height of the package because the center of gravity of the package may cause overturning or failure of the trailer structure.[9] Guidelines have been formulated to assist in the design and operation of trailers for radioactive material packages.[14]

A design for the cask support structure that uses a concept of *breakaway welds* has been used to ensure that failure of the cask supports (with the possible separation of the cask from the transport vehicle) occurs at predetermined locations and stresses.[15]

5.9 TRANSPORT VEHICLE

The transport vehicle must be considered in the design of the package although it is not regulated by 10 CFR Part 71. One important consideration is the center of gravity of the mounted package. A center-of-gravity height of less than 80 in. is appropriate for truck casks. Railcars have a maximum center-of-gravity height specification provided by the American Association of Railroads (AAR).

Truck trailers with air ride suspension give a softer ride to the cask package; consequently, 2 *g* (simultaneously on three axes) is a sufficient design parameter for the cask support structure.[12] Spring suspension trailers produce a harder ride for the package; therefore, the cask support structure must be designed to withstand 4 *g*. However, because the air-ride suspension trailer does not prevent the package from leaning as the trailer negotiates a turn, the center-of-gravity height is critical. The center of gravity may be lowered by the use of a drop-frame trailer, but there is a weight trade-off because the gooseneck (the connection between the trailer frame and kingpin of the trailer) of such a trailer is more massive and heavier than it is for a straight-frame trailer. There can be a bonus for using a heavier trailer because its frame must be sufficiently strong to resist twisting moments present during the turning of the vehicle.

5.10 NORMAL OPERATION VS HYPOTHETICAL ACCIDENT

The normal operation and hypothetical accident scenarios of 10 CFR Part 71 and the equivalent DOE Orders define the conditions which a transport package must meet and, in many cases, provide acceptance criteria. In other cases, acceptance criteria are ill-defined. For example, a

package must be subcritical under all conditions, but the margin of safety which must be met is not defined in 10 CFR Part 71 or the *Regulatory Guides*. Similar examples exist for structural aspects of transport packages. The most important of these are the stress limits of various package components under impact conditions. Suggested limits are discussed above, and guidelines may be found in *Regulatory Guides 7.6* and *7.8*.

The reader might note that the scenarios for normal operation and for a hypothetical accident do not overlap and that the transition from normal operation to an accident scenario is not defined. This discontinuity is not an oversight, but the result of the variability of accident initiators. The difficulty in defining the package state at the beginning of an accident cannot be avoided, and the stringency of the 9-m drop and 1-m pin-puncture scenarios is a means to bound possible accident scenarios. The adequacy of these scenarios to bound real-world accidents was evaluated in the modal study.[16]

The transition of a cask from normal operation to an accident impact scenario generally will involve the separation of the cask from the transport vehicle and cask support structure. Packages and transport vehicles that are free from protrusions or structures which could degrade the performance of a cask and its impact limiters are desirable.

5.10.1 Shipping Cask Analyses Systems (SCANS)[17] Code

The SCANS computer code is a finite-element, cask-analysis code developed by Lawrence Livermore National Laboratory for the NRC to calculate the stresses caused by a 9-m drop onto an unyielding surface. SCANS is designed to model spent fuel shipping casks with lead gamma shielding, although judicious use of modeling inputs allows a variety of cask designs to be modeled with some approximations. The SCANS code is a one-dimensional model for cylindrical axisymmetric casks that have no significant asymmetry about the cask long axis. Parameters in the SCANS code were adjusted so that the calculated stress results for a rail cask more accurately matched the results from a detailed nonlinear dynamic analyses of structures in three dimensions (DYNA3D)[18] analysis. Users of the code should familiarize themselves with the modeling approximations which were necessary to permit implementation of SCANS on a personal computer. The code was developed to assist the NRC in the review of transportation casks, and it is intentionally conservative. SCANS should not be used by a cask designer to perform analyses for the SARP but it is prudent to determine whether stresses calculated by SCANS will be within the allowable stress limits prior to submittal of the SARP. Cask analyses performed with multi-dimensional finite-element codes usually predict lower stresses than SCANS, and 9-m drop test results may be used to confirm that these lower stresses are valid. In the absence of drop test results, it would be difficult to clearly confirm that calculated stresses in a SARP, that were lower than SCANS results, were in fact the correct values. Because SCANS requires that the stress-strain curve (also called the load-deflection curve) for the impact limiters be provided by the user, this is another area where test results may be used to support analytical results.

SCANS also includes a calculation of the capability of the cask design to withstand pin puncture. As with the stress calculations, this calculation is intentionally conservative and should not be considered to be the final answer. The methodology used by SCANS is similar to the methodology used in NELM's equation [Eq. (5.1)], but the methodology is sufficiently different to cause large differences in the thickness of steel needed to withstand the puncture event. Drop tests onto a steel pin are the preferred approach to demonstrate the adequacy of a given cask design.

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6. Materials

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6.1 INTRODUCTION

The materials selected for the fabrication of a packaging designed to transport nuclear materials are often chosen for their ability to shield and contain; they can also be used to maintain the spatial distribution and subcriticality within the package should fissile material be carried. The nuclear material must be contained within the package in normal and accident load conditions and environments as specified by the regulatory bodies. The shielding function and criticality prevention of the packaging are treated elsewhere; however, this chapter discusses materials of construction for the packaging to ensure structural containment of the hazardous material. Such materials of construction, supported with material certifications and structural analysis, must be shown by design and testing to be adequate for the containment of the radioactive material.

Energy-absorbing materials, which are used as impact load limiters in nuclear shipping packages, are described in Appendix A of this *Handbook*. These materials are very different from those described herein, and they have no nuclear material containment function.

For the packaging and transport of radioactive materials, the U.S. Nuclear Regulatory Commission (NRC) has issued 10 CFR Part 71 (ref. 1). 10 CFR 71.33(a)(5), which requires that information regarding materials of construction be furnished in sufficient detail to provide a basis for evaluating a packaging. Furthermore, 71.37(b) requires, in part, that the applicant identify any established codes and standards proposed for use in package design. These regulatory requirements compel the applicant to provide material property data that are thorough and of high quality. If possible, materials that are described by standard specifications should be used. This ensures that the material properties, which serve as a basis for the various safety analyses, will be reliable. The applicant must adequately characterize any specified materials that are not described by authoritative standards. Not only should the material be characterized with respect to the values of its mechanical, thermal, and physical properties, but also the means by which quality is ensured and the effect of fabrication processes should be addressed.

Material manufacture, certification, and fabrication are presented as standardized procedures by a substantial number of international organizations, which are identified in Sect. 6.4. Although following these procedures will take advantage of a large amount of previous experience gained and recorded in working with these materials, the responsibility for the choices of cask materials and the related fabrication processes remain with the design organization.

6.2 MECHANICAL PROPERTIES OF MATERIALS

6.2.1 Material Characterization

The mechanical properties of materials that are of primary interest to designers of radioactive material shipping packages include stiffness, ductility, strength, toughness, resistance to fatigue, corrosion, wear, and creep; and expansion under thermal loads. Materials should be chosen, or design limits on their use imposed, based on quantified requirements for these aforementioned characteristics. Before discussing the influence of these properties on package design in more detail, it would be useful to review some fundamental aspects of materials, particularly metals, that contribute to the way these materials behave.

Metals, as they are used in typical structural applications, are crystalline. That is, their atomic configuration is characterized by an orderly three-dimensional array in a space-lattice. Most metals may be classified as either body-centered cubic (BCC), face-centered cubic (FCC), or hexagonal close-packed (HCP). These crystal forms, together with a list of some of the common metals that display that form at room temperature, are illustrated in Fig. 6.1. At elevated temperatures, these crystal structures may change. The most familiar example is the transformation of alpha-iron (ferrite, BCC) to gamma-iron (austenite, FCC) at 912°C. These phase changes with temperature are used to advantage when the microstructure of alloys is tailored to achieve particular mechanical properties. The lattices are also a factor to be considered during fabrication processes such as welding when, if not controlled, they may degrade the mechanical properties. Material properties strongly influenced by the crystal structure include the elastic modulus, coefficient of thermal expansion, thermal conductivity, and melting temperature.

For material properties such as yield and ultimate strength, fracture toughness, and ductility, the differences in crystal structure are less important than the presence of lattice imperfections. These lattice imperfections are described as small regions within which the regular three-dimensional array of atoms breaks down. These imperfections may be classified as point, line, or surface defects. There are three point defects: substitutional impurity atoms, interstitial atoms, and vacancies. Substitutional impurity atoms are atoms of another species that substitute for host atoms on lattice sites. Interstitial atoms are atoms of the same or different species that occupy a site between lattice atoms. A vacancy is a lattice site from which an atom is missing. In each case, the interatomic forces strongly distort the geometry of the lattice in the vicinity of the imperfection.

Surface defects are associated with polycrystalline solids, and they strongly affect the mechanical properties of these solids. During the formation of a solid from its liquid state, solidification begins with the formation of small regions of the solid phase, called *nuclei*, throughout the liquid. These nuclei grow through the accumulation of atoms at lattice sites. The orientation of these growing crystal nuclei is random, and when solidification is completed, these crystallites or grains remain randomly oriented. The interface between crystallites is called the grain boundary, which is a highly disordered region that has properties different from those of the interior metal grains or matrix. The grain boundaries typically have a thickness of about one or two atomic diameters. Because of the imperfect atomic arrangement, impurity atoms tend to segregate to the grain boundaries.

A slowly cooled material will have a small number of nucleation sites and large grains. A rapidly cooled material will have a large number of nucleation sites and small grains. The slower cooling

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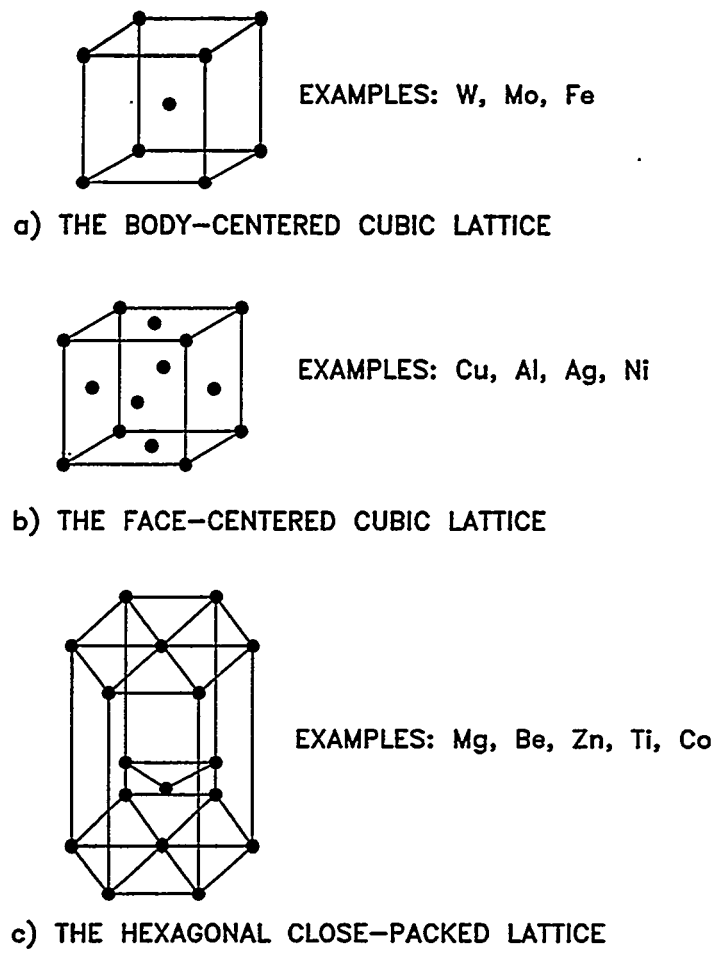

Fig. 6.1. The three common atomic packing arrays for solid metals.

rate favors the coarser microstructure because a more gradual temperature change results in the formation of a microstructure that is more nearly in its equilibrium, highly ordered state. Grain size is an important factor in determining the mechanical properties of materials. For example, fine-grained materials generally display greater hardness and greater flow stress than course-grained materials. Grain boundaries can play an important role in determining the failure properties of a metal. Under some conditions, the grain boundaries can be quite strong, and the material will fail with cracks passing through the crystals (transgranular fracture). Under other conditions, the grain boundaries can be weak, and failure will occur with cracks following the grain boundaries (intergranular fracture).

In an ideal crystal, all deformations under load are elastic and will reverse when the load is removed. The deformations are nearly linearly elastic for small displacements of atoms from their equilibrium positions and markedly nonlinearly elastic for large displacements. Failure would occur at theoretical strengths dictated by pulling apart atomic bonds, at levels far higher than those exhibited by real materials, in which the ultimate strength has values which are only 10^{-2} to 10^{-3} times that expected from the breakup of atomic bonds. This deviation from ideal behavior

is caused in part by the presence of line imperfections, called *dislocations,* of which there are three types: edge, screw, and mixed. An edge dislocation, viewed end-on as shown in Fig. 6.2, corresponds to the edge of an atomic plane that terminates within the crystal rather than passing all the way through it. Under stress, the dislocation moves easily because its cross section is displaced, in shear, one lattice space at a time. This displacement is not reversible when the stress is removed. Energy is dissipated, and this type of displacement is the basic means of inelastic or plastic deformation. Observable plastic deformation represents the accumulated displacements of large numbers of dislocations on parallel glide planes (much as a deck of cards might appear under shear).

On single crystals, the planes along which shear strains are observed are not necessarily the planes parallel to the direction of the shear stress. Rather, they are the planes that have the highest atomic density, where slip is most likely to occur. Slip is initiated when the component of the shear stress parallel to the slip plane reaches a critical value. In polycrystalline materials, which are composed of a conglomeration of crystallites randomly oriented with respect to each other, the random distribution of slip in each crystallite cancels any tendency for plastic strain to occur in a particular lattice direction so that the observed plastic deformation is in the direction of the applied shear stress.

As a metal is deformed plastically, dislocations will interact with grain boundaries, impurity atoms, interstitial atoms, and other dislocations. These interactions, in turn, generate additional dislocations and vacancies. As the dislocation density increases (dislocations per unit volume), these interactions tend to impede the motion of the dislocations, and greater shear stress is required to produce further slip. This process is the primary cause of strain hardening in metals.

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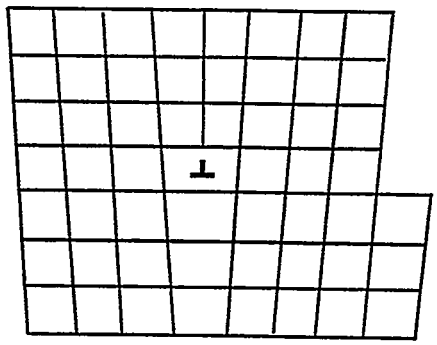
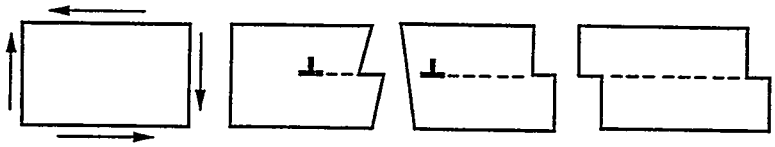


a) END–ON VIEW OF EDGE DISLOCATION



b) PLASTIC DEFORMATION PRODUCED BY DISLOCATION DISPLACEMENTS

Fig. 6.2. Schematic of an edge dislocation.

Particularly in BCC materials the motion of dislocations along slip planes is affected by temperature. The critical shear stress to move dislocations along slip planes increases by as much as a factor of 8 between room temperature and –196°C. This phenomenon produces the transition temperature for brittle and ductile failures observed in low-alloy steels. The stress concentration of the notch increases the transition temperature at which the ductile-failure behavior dominates.

6.2.2 Properties of Materials

The properties of materials are external measures of how the materials behave in the presence of applied loads and environmental conditions as internal or external pressure, dynamic and vibration loads such as free drops onto various structures, interactions with fluids, hot and cold temperatures, and so forth. The objective is to use these properties to describe fully how the packaging materials will react to any combination of applied loads and environmental conditions.

The properties of materials described in the following sections are the continuum properties of the materials. These properties are based on the idealization that the material behavior may be considered isotropic, homogeneous, and continuous over any length-scale of interest. This is a useful and generally accurate assumption for the polycrystalline materials used in the size and quantities typical of a nuclear material shipping container.

6.2.2.1 Modulus of Elasticity and Poisson's Ratio

When the applied loads to a structure are such that the resulting stresses and strains are small, the material behavior is said to be elastic and fully reversible. In polycrystalline metals, this elastic behavior is typically linear, and the constant of proportionality between the tensile stress σ and strain ε is the elastic, or Young's modulus, as shown in the following equation:

$$\frac{\sigma}{\varepsilon} = E \ . \tag{6.1}$$

The elastic modulus is important in describing the stiffness of a structure, and varies widely with different materials. The modulus of iron is 200 GPa (29×10^6 psi), while that of aluminum is 69 GPa. The stiffness of plastics and organics as would be used in energy absorbers in a nuclear material shipping package is less—in the range of 0.14 to 10 GPa. The modulus of elasticity may be measured by the load-vs-elongation characteristics of tensile and compression specimens, the speed of travel of mechanical waves in the material, or the vibration modes of a beam.

The elongation of a specimen in uniaxial tension, as for measuring the elastic modulus as described previously, may be accompanied by contractions in the perpendicular transverse directions. Poisson's ratio ν is the ratio of induced transverse strains to axial strains, as defined by the relationship

$$\nu = -\frac{\varepsilon_{transverse}}{\varepsilon_{axial}} \ . \tag{6.2}$$

For elastic deformation, the Poisson's ratio of iron is 0.28, while for aluminum it is 0.34. For plastics, the value of Poisson's ratio is typically 0.4. For plastic deformation in polycrystalline metals, where the material volume is nearly constant, Poisson's ratio is 0.5. The elastic modulus and Poisson's ratio may be used to derive the shear and bulk moduli of a metal.

6.2.2.2 Ductility

Ductility is the ability of a material to deform plastically prior to fracture. The plastic deformation is associated with dislocation motion. Experimental measures of ductility include elongation or reduction in area of a tensile test specimen prior to fracture. The reason for the importance of ductility is the unavoidable occurrence of stress concentrations associated with holes, notches, fillets and other discontinuities. In a ductile material, small plastic zones form at these stress concentrations. Therefore, a cask material should possess sufficient ductility to preclude, by a large margin of safety, fracture initiation at these stress concentrators. There are also instances in which local plastic deformations may occur as a result of assembly procedures, such as forced-alignment. The material should have sufficient ductility so as not to require special precautions during assembly, fabrication, and shipping. The question of what constitutes sufficient ductility is not easily answered. There is some consensus that above 20% elongation, a material is considered ductile, while below 5%, it would be characterized as brittle. For real manufacturing decisions, judgment, enhanced by more detailed analysis of local strains, is required.

For a material with sufficient ductility, the excursion of local strains into the yield region is of little concern because upon unloading, the stress and strain relax elastically. Because the material has undergone some prior plastic deformation, there will be some residual compressive stress at zero load. At reload, there will, at first, be a relief of the residual compressive stress. As the load continues to be applied, the stress will increase elastically up to its former level, and as long as the load is not increased beyond its former level, it will continue to cycle elastically, limited only by the high-cycle fatigue limit of the material. This process has come to be known as "elastic shakedown." Figure 6.3 graphically depicts elastic shakedown for a simple material behavior in which there is no strain hardening and where the yield strength in tension and subsequent compression is approximately equal. Elastic-cyclic material behavior will continue to govern the deformation of the material until the applied load is large enough to cause the residual compressive stress to exceed the yield strength in compression. For loads beyond this point, the material locally will cycle plastically with application of load and will require analysis of cyclic plastic strains to evaluate the time to fracture of the component.

Methods for the measurement of ductility include tests of a "go" or "no-go" nature as well as tests that quantify a measure of ductility, such as the reduction of area at failure. The American Society for Testing and Materials (ASTM) test method E190 is an example of this first type of test;[2] ASTM E8 is an example of the second type.[3] The designer must be aware that quantification of ductility in an absolute sense is problematic because the measure of ductility often depends on the specimen size and triaxiality of the stress state in the deforming region.

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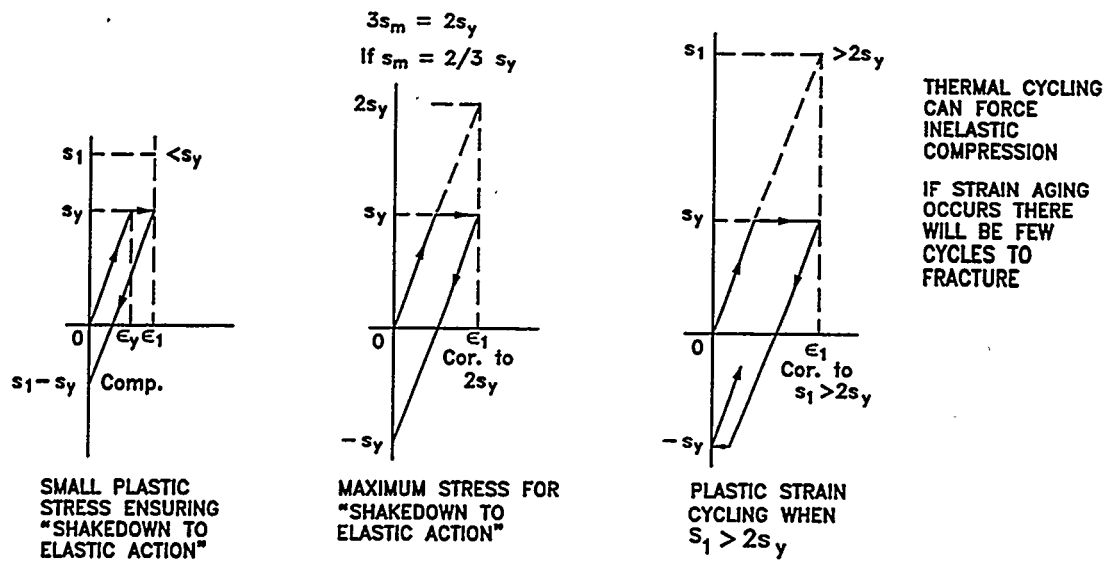

Fig. 6.3. Strain history beyond yield.

6.2.2.3 Strength

The properties of metals relating to ductility and strength are functions of strain, strain rate, and temperature. For many design situations, loads are of a magnitude and duration such that material properties obtained by standard slow- and moderate-speed pull tests at room temperature[3] are adequate. In this standardized test, yield strength, tensile strength, and elongation and reduction of the area at failure are typically reported. Requirements to provide complete stress-strain diagrams derived from these tests are not included.

Several test methods for measuring material strength and ductility at higher strain rates exist, but they are more difficult to standardize because of the complexity of the experimental apparatus and data-taking requirements. Even though standardized tests do not exist, there is a large amount of information and data in the engineering literature regarding testing under high strain rate and high-temperature conditions. Information on material behavior at high loading rates is presented in refs. 4–6.

Standard practices do exist for measuring strength and ductility at elevated temperatures (as an example, see ref. 7), and there also is a substantial amount of information regarding the use of the implementation of these data for the design and maintenance of structures.[8–10] Although the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC) and the related ASTM specifications remain the major source for material allowables regarding tensile strength. Other large data bases exist which contain material data for the mechanical properties for a wider range of steel and metal alloys.[11,12]

If an elastic-plastic stress analysis is used to determine the maximum deformation as a result of accident loading, it is generally necessary to develop a true-stress—true-strain material-response curve, which takes into account the decreasing cross-sectional area and increasing gauge length as the test specimen deforms under load. The engineering ultimate tensile strength is associated with the onset of necking of the test specimen. This is a condition of instability at the point at which the decrease in strength due to decreasing cross-sectional area begins to exceed the increase

in strength due to work hardening. This point can be defined on the true-stress—true-strain curve at the point at which the slope of the tangent is numerically equal to the true stress. This stress and the strain corresponding to this stress are important because they determine the limit of validity of the stress-strain data that can be used in the elastic-plastic analysis programs. For stresses and strains beyond this point, the changes in geometry with further plastic deformation will invalidate use of the original geometry of the modeled component.

An impact limiter functions differently from the way other package components do in that the impact limiter is expected to undergo large permanent deformations under some conditions. The important "strength" properties of an impact limiter are the force-displacement relations representing different impact directions. This information should be obtained by testing—either using compressive tests of representative specimens or tests of models or prototypes. An account should be made of the effect of loading rate and temperature (and possibly variables such as moisture content for wood, and other organic materials) on the response of the impact limiter.

An example of the use of the yield strength in the design of a structural member is seen in Fig. 6.4. Consider the limit load analysis of a cylindrical container under combined loading. This container or tube is assumed to have a radius R equal to 0.394 m (15.5 in.) and a thickness h of 0.0254 m (1 in.) and is subjected to internal pressure, bending moment, axial force, and torsion. The cylindrical container or thin-walled tube is assumed to be constructed from cold-rolled type 304 stainless steel with a yield strength σ_e of 560 MPa (81,200 psi). The stress state is approximated by a constant hoop stress σ_θ and shear stress $\tau_{z\theta}$ on the circumference. The bending moment and axial forces result in two regions of constant stress, uniform tension σ_{zt} over the part of the circumference $(\pi + 2\theta_0)$ and uniform compression σ_{zc} over $(\pi - 2\theta_0)$. The lower bound on the yield locus is defined as[13]

$$\frac{m^2}{\cos^2\theta_o} + \frac{3n_\theta^2}{4} + t^2 = 1 \ , \tag{6.3}$$

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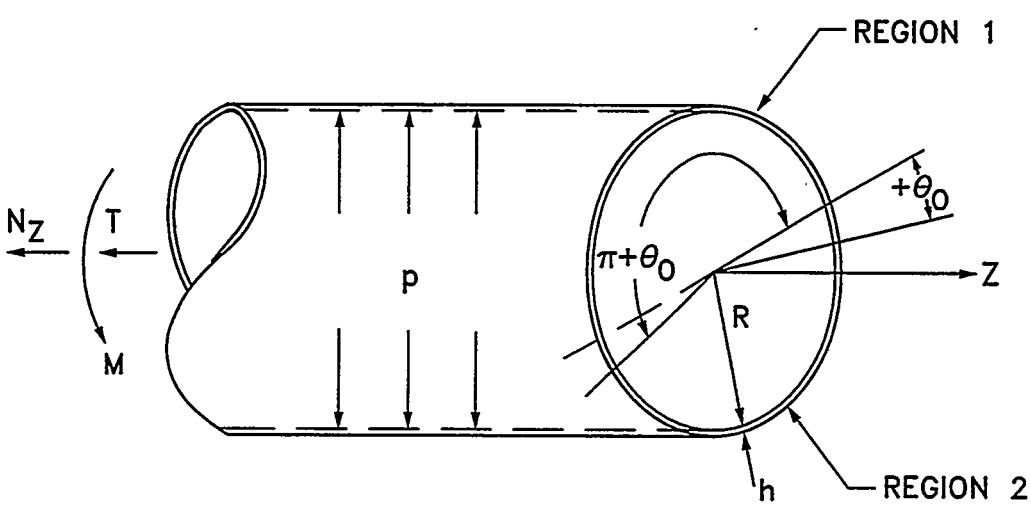

Fig. 6.4. Geometry and loading for a limit analysis of a thin-walled tube. *Source*: Fig. 1 from pp. 831–832 of *Journal of Applied Mechanics,* 1974, by permission of the American Society of Mechanical Engineers.

where θ_0, in terms of the axial and hoop forces and the applied torque, refers to the regions defined for the bending moments; that is,

$$\theta_o = \frac{(2n_z - n_\theta)\pi}{4\left[1 - \frac{3n_\theta^2}{4} - t^2\right]^{0.5}} . \tag{6.4}$$

Here m, n_z, n_θ and t are the dimensionless stress resultants for the bending moment:

$$m = \frac{(\sigma_{2t} - \sigma_{2c})\cos\theta_o}{2\sigma_e} , \tag{6.5}$$

the axial force:

$$n_z = \frac{\sigma_{2t}(\pi + 2\theta_o) + \sigma_{2c}(\pi - 2\theta_o)}{2\pi\sigma_e} , \tag{6.6}$$

the hoop force:

$$n_\theta = \frac{pR}{h\sigma_e} , \text{ and} \tag{6.7}$$

the torque:

$$t = \frac{\sqrt{3}\,\tau_{2\theta}}{\sigma_e} . \tag{6.8}$$

With a constant axial stress of 540 MPa (78,300 psi) in region 1, –308 MPa in region 2, and an applied shear stress of 180 MPa, the lower-bound estimate of the pressure which takes the stresses in the tube to the plastic limit load is 15 MPa.

The purpose of a limit load analysis is (1) to be able to compare actual loads to those load levels which represent the plastic limit of the material in a complex stress state and (2) thereby to establish a safety factor with respect to failure by gross yielding. A major limitation of this type of analysis is that it is limited to thin-walled structures in which the ratio of radius to thickness is greater than 10. For a thick-walled structure and a strain-hardening material, a numerical analysis (e.g., based on finite-element methods) would be required. The safety factor of the design use could be compared with various failure scenarios using a numerical analysis.

Table 1 (ref. 14) lists yield strengths and ultimate tensile strengths for a range of materials that have been used in pressure vessels and that might have use in packaging containers. Even though the use of such tables is an aid in material selection and design calculations, when used for the safety qualification of a structure, such tabulated data should be used only for comparison. Actual material properties, as measured from witness samples and as controlled through material process control (as an ASME/ASTM qualified material), must be used.

6.2.2.4 Fracture Toughness

All structures contain discontinuities or defects. These discontinuities may be inherent in the material as vacancies, dislocations, or grain boundaries. They may be present in the material because of fabrication processes, such as welding, rolling, or casting, which may produce slag inclusions, voids, or grain boundaries with excessive carbides. The defects may also be present because of the design and fabrication of the material into its final form, where the root of a screw thread or a very small radius cut may be viewed as a defect. When these defects are small and uniformly distributed, they determine the onset of plastic deformation in the material.

When the defect is on the order of the grain size or larger (on the order of 10 to 50 μm), and has a small root radius, it may be termed a crack. Because of stress concentration, plastic deformation tends to concentrate around this defect, and the net section-stress-level-to-failure may be less than the yield stress. This failure localization phenomenon is described as fracture, and it may be described as *brittle* or *ductile* based on the amount of deformation in the material directly adjacent to the defect or crack as it has grown.

In the stress analysis of structure with defects, a major contribution was made by considering the example of a large plate with an elliptical hole.[15] It was shown that the maximum stress σ_{max} is related to the average stress σ_a normal to the major axis of the ellipse by the relationship,

$$\sigma_{max} = \sigma_a \left(1 + \frac{2a}{b} \right), \tag{6.9}$$

where a is the half-major-axis and b is the half-minor-axis. The radius of curvature at the end of an ellipse is given by

$$\rho = \frac{b^2}{a}, \tag{6.10}$$

so that in terms of the half-major-axis and the root radius the maximum stress becomes

$$\sigma_{max} = \sigma_a \left(1 + 2\sqrt{\frac{a}{\rho}} \right). \tag{6.11}$$

Table 6.1. Specifications of materials and alloys for use in pressure vessels

Grade or type	Hardness (Rockwell)	Minimum ultimate tensile strength		Minimum yield strength		Remarks
		MPa	(ksi)	MPa	(ksi)	
			Low carbon steels			
ASTM A-30		380	55	210	30	
ASTM A-129		275	40	150	22	
ASTM A-201		380	55	210	30	
ASTM A-299		500	75	275	40	
ASTM A-414		310	45	165	24	
			Low alloy, low carbon steels			
ASTM A-202		500	75	310	45	
ASTM A-203		450	65	255	37	
ASTM A-225		485	70	275	40	
ASTM A-323		620	90	415	60	
ASTM A-357		415	60	210	30	
ASTM A-387		415	60	240	35	
USS "T-1"		800	115	620	90	
HY 80		~725	105	620	90	
HY 100		~860	125	760	110	
HY 180		~1360	200	1240	180	
			Alloy steels [all tempered at 370°C (700°F) or higher]			
4130	25–30 R_C		860–1000 (125–145)	710	103	
	32–36 R_C		1035–1170 (150–170)	910	132	7.5-mm-thick maximum
8630	25–30 R_C		860–1000 (125–145)	710	103	
	32–36 R_C		1035–1170 (150–170)	910	132	7.5-mm-thick maximum
4340	25–30 R_C		860–1000 (125–145)	710	103	
	32–36 R_C		1035–1170 (150–170)	910	132	
	39–43 R_C		1240–1380 (180–200)	1125	163	Beware of low
	43–46 R_C		1380–1515 (200–220)	1210	175	fracture toughness
			Titanium alloy (beware of brittle welds)			
TI-5 Al-2.5 Sn		800	(115)	620	90	RS-110C, A-110 AT
TI-6 Al-4 V		900	(130)	830	120	RS-128, C-120 AV
			Austenitic stainless steels (resistant to hydrogen embrittlement)			
	Annealed		655–690 (95–100)	310	45	Enhanced properties
21-6-9 WIM/ESR	96 R_B		770 (112)	470	68	result from warm
ASTM A-276)	43 R_C		1000 (145)	900	130	high energy rate
304	70–90 R_B		485–620 (70–90)	170–345	25-50	forging
	10–35 R_C		700–1240 (100–180)	345–1035	50–150	
316	70–85 R_B		500–620 (75–90)	210–415	30–60	
	10–30 R_C		700–1035 (100–150)	345–860	50–125	
321,347	79–90 R_B		500–655 (75–95)	210–380	30–55	
	10–35 R_C		700–1035 (100–150)	345–860	50–125	

When the root radius becomes small in comparison to the half-major-axis of the ellipse, the maximum stress at the end of the ellipse becomes very large. In the limit, as the root radius approaches zero, the magnitude of the stress becomes infinite, and the elliptical defect has become a crack.

Although in elasticity theory, the stress at every crack is infinite, the rate at which the stress goes to infinity, or the size of the region over which the stress is large, is distinguished by the use of the stress intensity factor K. This stress intensity factor should not be confused with the term stress intensity as used in the calculation of stresses according to the ASME B&PV Code. For the load-opening mode, during which the load on a large plate or structure is perpendicular to the major axis of the crack, the stress ahead of the crack tip is given by the relation

$$\sigma_{ij}^{*} = \frac{K_I}{\sqrt{2\pi r}} f_{ij}(\theta) \ , \tag{6.12}$$

where K_I is the mode I stress intensity factor, r and θ are the polar coordinates in the crack tip region with $r = 0$ at the crack tip, and f_{ij} are known functions of angular variation. Similar types of relationships apply for the inplane shear loading mode II and the antiplane shear loading mode III. Figure 6.5 depicts the three crack-loading modes.

The stress intensity factors are useful in predicting failure where material behavior is linear elastic, except in very small regions at the crack tip. When more general plastic deformation occurs prior to catastrophic crack growth, elastic-plastic characterizations of failure are necessary. For failure of more ductile materials, the related parameters, the J-integral or the crack tip opening displacement (CTOD) is used to characterize the severity of loading and crack geometry.[16]

The fracture toughness is the ability of the material under stress to resist the self-similar growth of a crack to a larger defect. The fracture toughness of a material typically depends on the temperature, rate of loading, and state of stress in the material near the crack tip. The fracture

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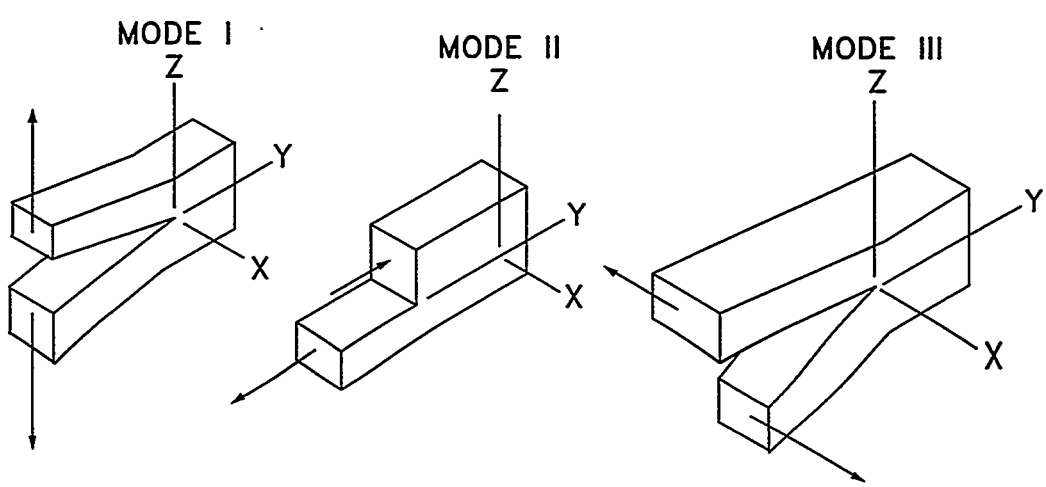

Fig. 6.5. The basic crack loading modes.

toughness of a material is generally larger with higher temperatures because of the greater mobility of dislocations. The fracture toughness of structural steels with a BCC (ferritic) microstructure is particularly sensitive to temperature. For dynamic loading rates, such a generalization is not possible because the dynamic fracture toughness may be greater or less than the static value. In regard to the state of stress, fracture toughness values are typically least when plastic deformation is most constrained as is the case when the stress state is hydrostatic or triaxial—that is, the stresses at the crack tip are nearly equal in the three orthogonal directions. The condition of maximum constraint for generalized two-dimensional analysis of stressed bodies is plane strain. The plane strain fracture toughness of a material K_{IC} is used in design as the minimum value for thick sectioned materials.[17]

For elastic-plastic behavior in more ductile materials, J_{IC} (ref. 18) and the critical CTOD (ref. 19) are used as the corresponding material crack growth parameters.

An alternative approach to specifying a minimum fracture toughness of a material has been developed for ferritic steels,[20] and has received approval by the U.S. NRC[21] for the control of fracture in shipping containers. In this approach, a design reference curve is used which correlates the dynamic critical stress intensity factor, K_{ID}, with the temperature relative to the nil ductility transition (NDT) temperature. The nondestructive testing (NDT) temperature is defined in ASTM E-208 (ref. 22). The K_{ID} vs NDT reference curve is used to characterize the toughness behavior as a function of temperature for selected structural steels. Then, a temperature difference between the lowest service temperature and the NDT temperature is required.

As an example of the application of fracture mechanics to the safe design of structure, consider the leak-before-break criterion. Here the preferred failure scenario is that the vessel will leak from a through-wall flaw and that the internal pressure of the vessel will be reduced before a crack can grow to a size such that the vessel will catastrophically fail, producing fragments that could become dangerous projectiles. Leaking is usually quite easy to detect, and the damage caused is easier to control in most design situations. Following the implementation of this criterion as presented in ref. 23, the leak-before-break criterion is

$$\frac{\pi(\sigma)^2}{\left[1-\frac{1}{2}\left(\frac{\sigma}{\sigma_{ys}}\right)^2\right]} \leq \frac{K_{IC}^2}{B}\left[1+1.4\left(\frac{K_{IC}^4}{B^2\sigma_{ys}^4}\right)\right]. \tag{6.13}$$

As an example, in an HY80 steel pressure vessel, the plate or vessel wall thickness B was equal to 0.095 m (3.74 in.), σ is the applied stress, σ_{ys} is the material yield strength of 552 MPa (80,000 psi), and K_{IC} was measured to be 131 MPa$\sqrt{\text{m}}$ (119.3 ksi$\sqrt{\text{in.}}$) (ref. 24). The applied stress is modified to be the quantity $(\sigma + p)$ in the numerator of the left-hand side of the equation to account for the increase in the stress intensity factor resulting from the presence of the internal pressure P. For the vessel pressure of 13.8 MPa there is a safety factor. If the safety factor is calculated on the pressure of 25.1 MPa, the point at which catastrophic failure and leak-before-break would have equal probabilities, the safety factor would take a value of 1.8.

6.2.2.5 Fatigue

The fatigue strength is the ability of the material when loaded under cyclic stresses to resist the growth of small defects to large defects. Fatigue is defined by ASTM as "the process of progressive localized permanent structural change occurring in a material subjected to conditions that produce fluctuating stresses and strains at some point or points and that may culminate in cracks or complete fracture after a sufficient number of fluctuations."[25]

The progression of small defects to large defects occurs as dislocations pile up to form slip bands, which move across grain boundaries to free surfaces to become extrusions, and the surface roughened by extrusions becomes the site for the formation of microscopic cracks which become macroscopic cracks. When the stress levels and defects are small, the propagation of defects is controlled by shear stresses, and surface imperfections play the major role. When the defects grow to the size of several grain boundaries, the stress normal to the defect plane starts to dominate. For components with small stresses and defects, a compressive stress at the surface tends to greatly increase fatigue life. For these components, when they are made from moderate strength steels, life prediction is typically accomplished through use of a stress-number-of-cycles-to-failure relationship, called an S-N curve.

An example of the data used to generate an S-N curve and its application to the prediction of failure in large-diameter bolts is found in ref. 26. For bolts in a range of sizes from 1 to 5 in. in diameter, a fatigue strength reduction factor of 4 represents the notch sensitivity resulting from the threaded connector's root radius. For the combination of mean (S_{mean}) and alternating (S_a) stress, a resultant equivalent stress, S_{eq}, was defined as:

$$S_{eq} = -\frac{7 S_a}{8 - \left[1 + \frac{S_{mean}}{S_u}\right]^3}, \tag{6.14}$$

where S_u is the ultimate tensile strength of the material. The stress-number-of-cycles-to-failure relationship is given as

$$S_{eq} = \frac{A E}{N_f^c} + S_e, \tag{6.15}$$

where S_e is 0.4, the ultimate tensile strength, and E is the elastic modulus of the material. A and C are measured material constants whose values are 0.444 and 0.611, respectively, for the quenched and tempered Cr-Mo/Mn 0.43 m (AISI 4340) steel of the bolts. A 82.5-mm (3.25-in.)-shank-diam bolt with a thread root radius of 0.43 m (0.017 in. and an ultimate tensile strength of 861.8 MPa psi) was cyclically loaded at 1.5 Hz. The test temperature was 21°C with a net section mean stress of 413.3 MPa (59,940 psi) and an alternating stress of 275.5 MPa (39,960 psi). There were 6,043 load cycles to failure. This number compares to a fatigue life prediction, using the previous equations of 2,580 cycles to failure using the standard notch sensitivity fatigue strength reduction factor of 4. Because four times the net section alternating stress was greater than the cyclic yield strength, there was no effect of the mean stress. This data scatter is quite typical, and

this lack of precision in life predictability is reflected in the safety factors of 2 on stress and 20 on life used in heavy-equipment design.

The test specification to develop basic material fatigue life curves is found in ref. 27. Design fatigue curves using a similar S-N approach are found in the ASME B&PV Code for nuclear power plant components in the Sect. III appendixes of the code.[28]

For components with higher stresses and larger defects, a fatigue crack growth criteria based on fracture mechanics is used. What constitutes a large or small stress or defect size is governed by the rules for each predictive criteria, and it depends on the material properties as well. Even though such fracture mechanics approaches to estimates of fatigue life are grounded in an extensive analytical and experimental developments found in the engineering literature, they are not as yet part of the regulatory requirements for nuclear material transportation, and so they have found little use in the design of transportation containers. The information provided by the fracture mechanics analysis is still valuable for the design engineer because it provides a basis for specification of allowable defect sizes in the procurement nondestructive inspection and qualification process.

As an example of fatigue-crack growth estimates based on fracture mechanics, consider a question regarding the replacement of a vessel used as an isostatic press. Part of this consideration is whether the steel pressure vessel in current use is safe for future use, and, if safe, how many pressure cycles should be allowed before nondestructive examination (looking for cracks in the vessel) is required.

The pressure vessel is forged steel with a Rockwell C hardness of 35 and an ultimate tensile strength of 1138 MPa (165,000 psi). The operating pressure is 138 MPa (20,000 psi). The vessel has an inside diameter R_i of 0.457 m (18 in.) and an outside diameter R_o of 0.749 m (29.5) in. A vessel of similar material and construction and use was withdrawn from service with about 12,000 cycles of use. A circumferential crack propagated from the root of threads machined in the part to a depth of 0.019 m (0.75 in.). The material was assumed to be similar to AISI 4340 steel based on a statement in the report documenting the failure of this pressure vessel.

Later it was determined from the original fabricator of the pressure vessel that the material was a steel similar to American Iron and Steel Institute (AISI) 4330V. The use of the typically lower fracture toughness properties for AISI 4340 steel is conservative and should introduce additional safety into the fracture recommendations.

The hydraulic oil used in the pressurizing operation should provide a benign environment for the steel. Leak-before-break is not an option as a failure criteria because of other safety considerations.

The lifetime prediction for this vessel is based on the assumption that the failure of the vessel would cause no injury and only moderate dollar losses because the vessel is operated from a remote location. Material properties found in the engineering literature will be used.

The material properties used in the analysis are from ref. 29. The fracture toughness values for AISI 4340 steels from Table 6.29.2 of ref. 29 suggest that a K_{Ic} of 88 MPa$\sqrt{m}$ (80 ksi$\sqrt{in.}$) is a best estimate for the lower bound fracture toughness for the material, with 115 MPa$\sqrt{m}$ (105 ksi$\sqrt{in.}$) being the higher bound. From Table 6.29.3.2 of ref. 29, the cyclic crack growth parameters are derived using a linear least-squares fit on a log-log plot of the data using the relationship

$$\frac{da}{dN} = A(\Delta K_I)^n \ . \tag{6.16}$$

In this relationship N is the number of applied load cycles, a is the crack length, ΔK is the change in the applied stress intensity factor from minimum load to maximum load, and A and n are the curve-fit parameters. For a ratio of the minimum to the maximum load equal to 0.50, $A = 3.82 \times 10^{-11}$ and $n = 2.50$; ΔK is measured in units of MPa $\sqrt{\text{m}}$ and da/dN is in meters per cycle.

The largest flaw size that could escape detection (as estimated by the nondestructive examination inspector), a_o, was taken to be 0.762 mm (0.030 in.) deep in the threaded section where the magnetic particle inspection method is used. In the cylindrical portion of the tube which is inspected using a hand-held ultrasonic transducer, the maximum flaw that could escape detection was taken to be 3.18 mm (0.125 in.). Based on a consideration of the maximum stresses, the critical flaw geometry was assumed to be long and shallow, with the plane of the crack perpendicular to the hoop stress. For the cylindrical geometry of the pressure vessel as shown, then according to ref. 30, the stress intensity factor is

$$K_I = 1.12\left[\frac{2p(R_0)^2\sqrt{\pi a}}{(R_0)^2 - (R_i)^2}\right] . \tag{6.17}$$

This relationship is valid for the prefactor equal to 1.12 for crack depths less than 14.5 mm (0.57 in.) and $R_{id}/R_{od} = 0.61$. In this example, the internal pressure p is equal to 13.8 MPa (20,000 psi). For $K_{Ic} = 110$ MPa$\sqrt{\text{m}}$ (100 ksi$\sqrt{\text{in.}}$), the critical crack depth is calculated from the above relationship to be $a_{cr} = 15.7$ mm (0.62 in.). For $K_{Ic} = 88$ MPa$\sqrt{\text{m}}$ (80 ksi$\sqrt{\text{in.}}$), then $a_{cr} =$ 10.2 mm (0.40 in.).

By integrating the above relationship, the number of cycles to failure can be shown to be

$$N_f = \left(\frac{2}{2-n}\right)\left(\frac{1}{A}\right)\left[\frac{1}{\left(\dfrac{2.24\,p\,(R_0)^2\sqrt{\pi}}{(R_0^2 - R_i^2)}\right)^n}\right]\left(\frac{2-n}{a_{cr}^2} - \frac{2-n}{a_0^2}\right) . \tag{6.18}$$

Taking the lower value for $K_{Ic} = 88$ MPa$\sqrt{\text{m}}$ (80 ksi$\sqrt{\text{in.}}$) and $a_o = 0.125$, the number of cycles to failure is $N_f = 5{,}000$. That is, after 5,000 cycles the crack depth will be at its critical value of 10.2 mm (0.40 in.). With planned operations that cycle the pressure in the vessel about 200 times per year, the current inspection schedule is once per year. When the risks of failure are reasonably small, continued use of the vessel under this inspection interval and environment is acceptable.

The foregoing calculations of fatigue life are based on the assumption of a constant ΔK, which makes the number of cycles to failure a reasonable hand calculation. Where more complex loading spectra are prescribed, the integration of the load cycles to predict failure is usually done on a computer.

6.2.2.6 Corrosion

Corrosion may be defined as the surface degradation of metals resulting from chemical reactions with other materials in their environment. Corrosive degradation may take several forms depending on the metal surface and the reactants. Forms of corrosion include:[31]

1. *Uniform attack.* Here there is a progressive uniform attack of the metal surface. This simplest form of corrosion has a straightforward design remedy. The thickness of the material required for pressure containment or shielding is increased based on the number of hours of service and the surface degradation rate. Knowledge of this corrosion allowable is required by the ASME B&PV Code for nuclear pressure boundary components.[32] For example, for carbon steel immersed in sea water a typical corrosion rate is 130 μm/year (ref. 33).

2. *Pitting.* Here the corrosive agent attacks certain regions of the surface preferentially. A hole or pit forms at the surface such that its width is comparable to its depth. Pitting may occur underneath another layer of corrosive product caused by uniform attack, so that failures by the corrosion may occur unexpectedly.

3. *Crevice attack.* This is another form of nonuniform corrosion attack. Here the corrosive attack becomes concentrated at geometric discontinuities. These discontinuities may be bolt holes, machined threads, the edge of a protective layer, etc. Depending on the geometry of the discontinuity, the corrosive agent may substantially change as the diffusion of species takes place in the environmental medium, such as an aqueous solution, and the corrosive specie must be replenished from the surface to the more inward active region of higher stress. In general, a crevice will also be a region of large stress because of the geometric discontinuity. Large stresses will also increase the energy available for release through corrosive action.

The most severe form of geometric discontinuity is a crack. The geometry of the crack tip region may accelerate or decelerate the diffusion of corrosive species to the crack tip region, but the large active and residual stresses present there accelerate rate growth. In the presence of a corrosive media, normally ductile materials may fail in a brittle fashion at low loads. Given the existence of defects, the stress intensity factors for defects may be calculated, and crack growth may be predicted on the basis of the applied stress intensity factor. The threshold stress intensity factor below which corrosive attack will not occur is termed K_{ISCC}. When the applied stress intensity factor is above K_{ISCC}, the crack growth rate may occur at a constant rate over a wide range of stress intensity factors.

As an example of this behavior, Fig. 6.6 shows the crack growth behavior of 304 stainless steel immersed in a 42% $MgCl_2$ solution.[34] K_{ISCC} here is seen to be about 11 MPa $\sqrt{m}$. K_{ISCC} for this material appears to remain fairly constant for a number of corrosive environments, although the plateau crack growth rate shows variation, as seen in ref. 28. For a postulated axial crack of uniform depth a of 0.003175 m in a piping section of 0.0127 m wall thickness b, inside diameter

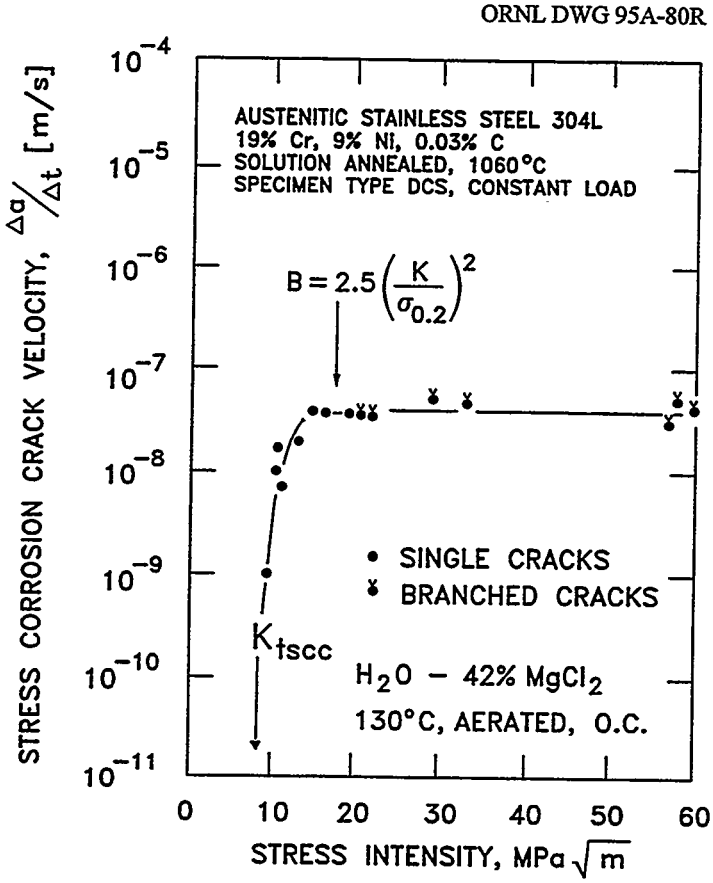

Fig. 6.6. Stress corrosion crack velocity for 304L stainless steel. *Source*: Fig. 2 of *Stress Corrosion Crack Growth in Austenitic Stainless Steel,* by M. O. Spiedel, with permission of NACE International.

R_i of 0.127 m, and an internal pressure p of 26.0 MPa, an applied crack driving force (J_a) may be calculated. As noted in ref. 16, the plastic portion of the applied crack driving force for this situation is given by:

$$J_a = \alpha\sigma_0\varepsilon_0 c\,(a/b)\,h_1(a/b, n, R_i/R_0)\,(p/p_0)^{n+1}\,. \tag{6.19}$$

The plastic portion of the crack driving force is 99.6% of the total crack driving force at this load level, with the elastic energy giving a negligible contribution to the total force. The material parameters α (a material constant), n (a strain-hardening exponent), σ_0 (yield strength) and ε_0 (yield strain) are taken from the power stress-strain law,

$$\frac{\varepsilon}{\varepsilon_0} = \alpha\left(\frac{\sigma}{\sigma_0}\right)^n\,. \tag{6.20}$$

Typical material constants for type 304 stainless steel with some cold working are $\alpha = 0.997$, $n = 7.3$, $\sigma_0 = 56$ MPa, and $\varepsilon_0 = 0.00228$. The dimensionless function h_1 [see Eq. (6.19)] is tabulated

in ref. 16 and describes the crack tip stress field, and depends on the ratio of the crack depth to the wall thickness, the strain-hardening exponent, and the ratio of the inside-to-outside diameters. The ratio of the applied internal pressure to the plastic limit pressure also enters the calculation for the crack driving force. The total applied crack driving force was calculated to be 659 (N· m)/m^2.

From tests of type 304 stainless steel piping products, an elastic plastic fracture toughness J_{IC} of 846,000 N · m/m^2 (4830 in.-lb/in.^{2}) was measured.[35] Since

$$J_a << J_{IC} \ , \tag{6.21}$$

no failure resulting from this crack driving force will occur. However, the applied stress intensity factor K_I may be calculated from the equation

$$J = \frac{K_I^2 (1 - \nu^2)}{E} \ , \tag{6.22}$$

given J as the applied crack driving force, where E is Young's modulus and ν is Poisson's ratio of the material. The applied crack driving force of 659 (N·m)/m^2 is therefore equivalent to an applied stress intensity factor of 11.8 MPa$\sqrt{\text{m}}$. Because the applied stress intensity factor is greater than K_{ISCC} corrosion crack growth, as a specialized form of crevice attack, will occur at a rate of 5×10^{-8} m/s. Because this crack growth rate is constant for a wide range of stress intensity factors, the time to failure by corrosion crack growth may be calculated as the remaining ligament divided by the crack growth rate. This calculation yields a time to failure of 1.9×10^5 s or 53 h. It is important to note in this example that the crack driving force is not a linear function of pressure because the energy for crack growth is largely stored in the plastically deformed material at the crack tip. If the applied pressure in the piping section were reduced 32% to 17.6 MPa for a safety factor of 3 with respect to the yield strength, the applied stress intensity factor would be reduced 80% to a value of 2.46 MPa$\sqrt{\text{m}}$. Because this is much less than K_{ISCC}, the corrosion crack rate becomes 4 orders of magnitude less—in the range of being immeasurably small.

4. *Galvanic coupling.* When two metals, or dissimilar electrodes, are connected in an electrically conductive environment, an electrolyte, preferential attack occurs on the anodic material. The anodic material is the active or electronegative of the pair. The cathodic member of the couple will be protected from galvanic corrosion by its connection to the anode.

When exposed to normal climatic conditions, steel will corrode even when it is not connected electrically to another metal. This process conforms with the principles of galvanic coupling (as stated previously because steel is not homogeneous, but rather it contains areas of slightly differing chemical composition. Figure 6.7 shows a number of common metals arranged in order of decreasing electrode potential. Steel is placed in the middle of the chart. Under normal circumstances, each metal in the scale will be anodic to those listed under it and will corrode when electrically connected. Conversely, each metal is cathodic to those listed above it and the steel component will be protected from corrosion when in physical contact with the more active metal.[36]

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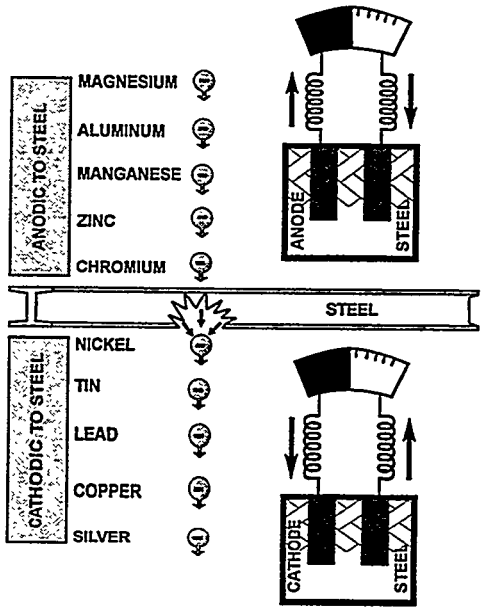

Fig. 6.7. Electrode potentials of elements. *Source:* V. R. Pludek, *Design and Corrosion Control*, Macmillan Press, Ltd., Basingstoke, England 1977.

5. *Hydrogen embrittlement.* Although embrittlement may occur in many forms,[37] hydrogen embrittlement is probably the most common form. Hydrogen embrittlement itself does not describe a single process but rather a group of related processes during which hydrogen from various sources acts to degrade the ductility and fracture toughness of a metal structure. A definition of hydrogen embrittlement from a military handbook for protective finishes follows:[38]

> When atomic hydrogen enters high-strength steel (and certain other high-strength alloys), it may cause any of several undesirable events to occur. If large quantities of hydrogen are introduced, there may be a general loss of ductility, or if the hydrogen accumulates in certain localized areas, internal bursts or blisters may be produced. Under certain circumstances, hydrogen may react with the alloy to produce catastrophic brittle failures at applied stresses far below the yield strength or the nominal design strength for the alloy. These phenomena are collectively referred to as hydrogen embrittlement. However, only the latter (i.e., the hydrogen induced brittle failure is of primary concern in the plating and coating of high strength steels). Because this phenomena often happens (SIC) in materials that exhibit no significant loss in ductility (as measured by a conventional tensile test), it is sometimes referred to as hydrogen-induced delayed brittle failure or hydrogen stress cracking.

6. *Corrosion fatigue.* Corrosion fatigue is failure caused by cyclic stresses in the presence of a corrosive environment. Just as static stresses increase the corrosion rates in the presence of crevices, cyclic stresses combine with structural discontinuities to cause corrosion fatigue. This results in metal failure at stress levels well below the endurance limit for noncorrosive conditions. The combined deleterious effects of corrosion and fatigue are greater than the sum of the damage each would cause individually.[39]

6.2.2.7 Galling

Galling (or adhesive wear) is a common, unavoidable form of material surface degradation. Although it is unavoidable, the resulting damage may be reduced to an acceptable level by understanding its causes and applying the appropriate remedies. Adhesive wear occurs when two bodies come into sliding contact and surface asperities contact each other. These asperities or surface ridges that form both sides come into intimate contact, and material from one surface may be transferred to another. Later, this material transferred may return to the original surface or be broken off as a fragment between the two surfaces. Large attractive forces may occur between the two surfaces, depending on the total area in actual contact.[40]

Adhesive wear may be greatly reduced by lubrication between the two surfaces. If it is impossible to provide lubrication or a sacrifical slip layer between surfaces, material selection for the opposing bodies can also greatly reduce the wear rate both in terms of the amount of material removed and the forces required to cause continued sliding. Tables in refs. 41 and 42 provide information regarding the choices of material couples to reduce adhesive wear.

6.2.2.8 Thermal Expansion

When heat is added to a material, there is a corresponding change in temperature and volume. The coefficient of thermal expansion α for a material (at constant pressure) is defined as

$$\alpha = \frac{1}{L}\left(\frac{\partial L}{\partial T}\right)_p \tag{6.23}$$

for a unidirectional case, where L represents the gauge length in question and T the temperature. For polycrystalline metal materials, the assumption of isotropy is usually made so that the coefficient of thermal expansion is the same in all coordinate directions.

All the structural materials that are part of a nuclear material packaging must be considered to experience at least the full prescribed environmental temperature variation of –40 to 50°C. The actual metal temperature may significantly exceed the ambient temperatures because of decay heat released by nuclear processes within the package.

Thermal stresses arise when the expansion or contraction of a structural member is constrained. This situation may occur in the structural materials of the container because of the thermal shock produced when rapid temperature changes are experienced, or when materials with different coefficients of thermal expansion are joined together.

Values of the coefficient of thermal expansion for a wide range of materials are found in ref. 43. Figure 6.8 shows a plot of percent thermal linear expansion as a function of temperature for well-annealed alloys of carbon steel. The ASME B&PV Code also has an extensive table of values for the coefficients of thermal expansion as functions of temperature for code-certified materials.[44]

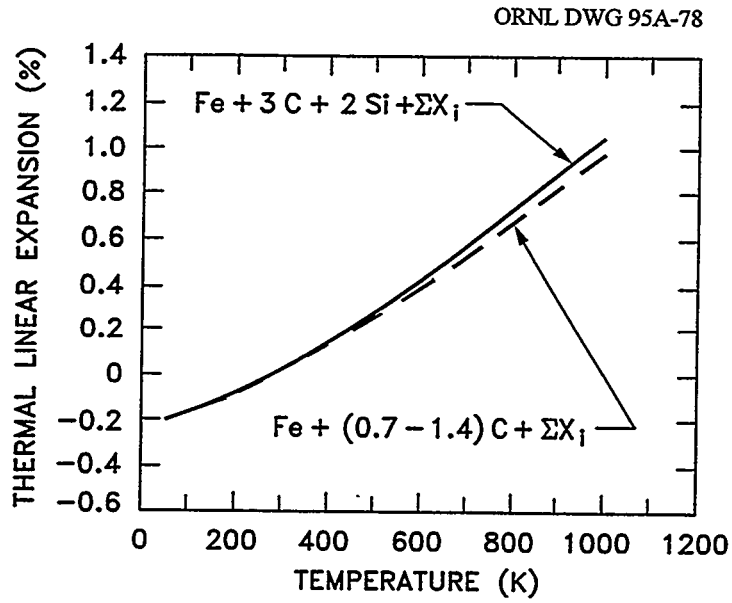

Fig. 6.8. Values of thermal expansion for selected steel alloys. *Source*: Fig. 257R from *Thermal Expansion: Metallic Element and Alloys,* by Y. S. Touloukian, R. K. Kirby, R. E. Taylor, and P. D. Desai, with permission of Plenum Publishing Corp.

As an example of thermal stresses that can occur in packages, consider the use of a ceramic material that might be used as part of a long-term storage container for nuclear waste.[45] Assume two alumina half-cylinders are fabricated to permit spent fuel to be inserted before they are sealed. Once spent fuel is inserted, the two cylindrical pieces are joined in the middle at the waist by an aluminum diffusion bond. The bond is developed by applying a combination of temperature and pressure locally. A region within 4 in. of the joint is heated to produce a bond temperature of 850°C, and an external pressure of 138 MPa is applied. The temperature gradient in the alumina and the resulting stresses can be calculated by using finite-element thermal and stress analysis computer codes. The relevant material properties for the alumina are the following:

Modulus of elasticity	379 GPa
Poisson's ratio	0.21
Thermal conductivity (400°C)	100 W/m K
Specific heat	879 J/kg K
Coefficient of thermal expansion	
(25°C)	3.4×10^{-6}
(500°C)	6.0×10^{-6}
(1000°C)	8.0×10^{-6}

The thermal stresses were found to be a maximum at the waist outside surface, with an combined effective stress of 76 MPa. With the application of the external pressure, the stresses in the principal coordinate directions all become compressive, and the region of maximum effective stress shifts to the inside diameter of the alumina-cylinder-half in the waist region. The magnitude of the effective stress is 460 MPa.

Both residual and thermal stresses that might be present in waste cylinders containing, for example, spent fuel elements may be modeled using these same procedures. These types of analyses provide some guidance for the specification of the fabrication procedures for these ceramic nuclear waste containers.

6.2.2.9 Creep

Plastic deformation is a process that is affected by the time, temperature, and stress level at which the change of shape occurs. Plasticity in polycrystalline materials may be classified by the atomistic processes which produce it: (1) collapse at the ideal strength flow when the ideal shear strength is exceeded; (2) low-temperature plasticity by dislocation glide, as was discussed in Sect. 6.2.2.2; (3) low-temperature plasticity by twinning; (4) power-law creep by dislocation glide or glide plus climb; and (5) diffusional flow.[46] Creep (e.g., time- and temperature-dependent plastic flow) includes the last two mechanisms, though in the analysis of real structures, the deformations may occur simultaneously and in combination.

The material behavior of creep is usually associated with plastic deformation phenomena which occurs at low strain rates over long periods of time. The ASME B&PV Code design stress for a material is based, in part, on the concept that the design stress should cause a creep strain rate of no more than 10^{-7}/h (ref. 47).

Creep in different metals and alloys is compared through the use of a dimensionless ratio of the absolute temperatures T/T_m, where T is the temperature of interest and T_m is the material melting temperature. Parabolic and secondary creep are the most important structurally and can occur within a temperature range of from 0.2 to 0.7 T_m (ref. 48). For metals with low melting temperatures (e.g., lead with a T_m of 600 K), creep phenomena should be considered at temperatures above 120 K (–153°C), which is much less than a typical ambient temperature. Commercial-pure aluminum has a melting temperature of 919 K, so that creep phenomena here may be significant at 183 K, depending on the stress level and time of temperature exposure. For steel alloys, for which an average T_m of 1810 K is used, the corresponding temperature at which creep might start is 362 K (89°C).

Section III of the ASME P&VC Code specifies a maximum metal temperature for all materials allowed for pressure-retaining components; these temperature values are found in Sect. II, Part D, Subpart 1, Tables 2A, 2B, and 4 (ref. 49). Above these maximum temperatures creep must be considered as a significant factor. Even though many of the ferrous materials have a temperature of 644 K (0.36 T_m) specified, a few of the materials have maximum allowed temperatures as low as 310 K (0.17 T_m); therefore, some caution is advised.

Each material should be evaluated for the possible significance of creep behavior based on its measured, pedigreed behavior and exposure in terms of the applied temperature and stress time history.

6.3 SIGNIFICANCE OF MATERIAL PROPERTY DATA

This section provides guidelines for the evaluation and presentation of the significance of material property data. The term *significance* is meant to imply that there should be a standard method for assessing the level of confidence for use in analysis and design for a particular material property.

It should also imply that for different situations, different values of the same material property may be properly used.

As an example, in the design of a pressure vessel, one may use a material with an ASME B&PV Code allowable stress level, or a minimum yield strength from an ASTM specification in order to define the maximum pressure capacity of the structure. However, if one is trying to realistically predict the expected failure modes in the vessel resulting from accident conditions and if a decision is required whether the vessel will fail by plastic yielding or crack growth, the average material yield strength would properly be used to define the onset of plastic deformation.

The level of confidence that can be assigned to any particular quantification of a material property value (whether average, minimum, or other) depends on the test procedure, the number of tests, the test recording system, and the distribution of data values.

Statistical methods have been developed to objectively measure and compare test results which quantify material property values. A complete description of the statistical treatment of a wide variety of mechanical test data is found in ref. 50.

As an example of the use of statistical methods to calculate design allowables, consider the "A" and "B" basis allowables of the preceding reference. Here the A basis is defined as that property value of which at least 99% of the population of values is expected to equal or exceed the A basis mechanical property allowable with a confidence of 95%. The B basis is defined as that property value of which at least 90% of the population of values is expected to equal or exceed the B basis mechanical property allowable with a confidence of 95%.

A code allowable, such as from an ASTM specification or the ASME B&PV Code, typically does not have a statistical significance attached to its value, but this significance may be derived if the criteria for the choice of the allowable and the material database used to justify it is available.

6.4 CODES AND STANDARDS

In order to ensure a high level of quality, 10 CFR 71.37(b) mandates that ". . .any established codes and standards proposed for use in package design, fabrication, testing, maintenance and use. . . ." be identified. Although materials of construction are not specifically mentioned here, their selection is an integral part of the design process, and, consequently, they are subject to the requirement that the codes and standards that determine their properties be identified. Furthermore, even though paragraph 71.37(b) seems to require *identification* of codes and standards, the usual practice has been to limit the choice of materials for package components to those *for which a code or standard exists.*

6.4.1 Adaptation of the ASME B&PV Code to Packaging System Components

The ASME B&PV Code has proved to be a conservative basis in many areas, including the design, fabrication, inspection, and acceptance of nuclear components. Selection of appropriate materials for package design with associated fabrication and acceptance guidelines is, therefore, based in part on the applicable ASME code section requirements.

The criteria for the various component safety groups can be associated with the requirements of the ASME code. An overview of the appropriate sections of the code applicable to materials and fabrication issues is given below:

- Sect. II, "Materials Specifications"
- Sect. III, Division 1, "Nuclear Power Plant Components"
  - Subsect. NCA, "General Requirements"
  - Subsect. NB, "Class 1 Components"
  - Subsect. ND, "Class 3 Components"
  - Subsect. NF, "Components Supports"
  - Subsect. NG, "Core Support Structure"
  - Mandatory Appendixes I to XXII
  - Nonmandatory Appendixes, A to R
- Sect. V, "Non-destructive Examination"
- Sect. VIII, Division 1, "Pressure Vessels"
  Division 2, "Alternative Rules"
- Sect. IX, "Qualification Standard for Welding and Brazing Procedures, Welders, Braziers, and Welding and Brazing Operators"
- Sect. XI, "Rules for In service Inspection of Nuclear Power Plant Components"

Although Sect. III of the code was developed for nuclear reactor components and Sect. VIII was developed for pressure vessels, portions of these sections may be used for design criteria for transportation packages. The other sections listed above are referred to by the governing articles contained in Sects. III and VIII.

The transportation package components which function to ensure the containment of the contents are recommended to conform to Sect. III, Subsect. NB of the code. All transportation package components which function to ensure that the package remains subcritical are to conform to Sect. III, Subsect. NG, which was originally prescribed for core support and internal structures of a nuclear reactor system. The transportation package components that function to ensure adequate shielding of the radiation emitted by the package contents and other safety-related functions are to conform to Sect. III, Subsect. NF, or Sect. VIII, Division 1, of the code.

Paragraph C.1 of NRC *Regulatory Guide 7.6* (ref. 51) (adopted by the U.S. Department of Energy), specifies that values for material properties, design stress intensity (stress allowable), and design fatigue curves for Class 1 components should be in accordance with ASME B&PV Code rules and specifications. Specifications for materials are contained in Sect. II of the code—Part A for ferrous materials and Part B for nonferrous materials. The design stress intensities for these materials appear in the Sect. III appendixes of the code.

In the event that a material selected is not covered by an ASME specification, the NRC allows stress intensity limits to be defined in accordance with Article III-2000 of the ASME code appendixes. These procedures for Class 1 components mandate the safety margins that are to be applied to either yield or tensile strength minima specified in an authoritative material specification. The code of choice, if the material is not listed in the ASME code, is the collection of specifications published by ASTM.

In addition to providing material specifications that can be used for packaging designs, the ASME code establishes rules of safety governing fabrication, inspection, material testing, and acceptance during construction of pressure vessels and related support and internal structures. These safety

rules may, in some circumstances, be applicable to the design and fabrication of nuclear material transportation packages. Section III, Subsects. NB and NG, should be consulted for components which ensure containment or subcriticality, respectively. These sections of the code should provide a complete, conservative basis for the design, fabrication, inspection, and acceptance of these components. Safety-related components may be constructed with guidance from Sect. III, Subsect. NF, or Sect. VIII of the ASME code.

An addition to the ASME code is under consideration which would provide design rules and material requirements for nuclear material transportation packages.[52]

6.4.2 Other Codes, Standards, and Specifications

Even though adherence to the suggested codes is recommended, it is not required. There may be circumstances for which additional information is available from other sources. Generally, however, a package designer should reference an authoritative code, standard, or specification when reporting material properties and fabrication, testing, inspection, and acceptance procedures. A partial list of organizations that issue standards or specifications, and applicable handbooks, follows:

1. American Society of Mechanical Engineers, 345 East 47 Street, New York, NY 10017
2. American Society for Testing and Materials, 1916 Race Street, Philadelphia, PA 19103-1187
3. Society of Automotive Engineers of America (SAE), *SAE Handbook,* 400 Commonwealth Drive, Warrendale, PA 15096-0001
4. American Iron and Steel Institute, 1133 15th St. N.W., Suite 300, Washington, D.C. 20005
5. American National Standards Institute, 11 West 42nd Street, New York, NY 10036
6. American Welding Society, 2501 Northwest 7th Street, Miami, Florida 33125
7. *Military Handbook: Metallic Materials and Elements for Aerospace Vehicle Structures, MIL-HDBK-5 Coordination Activity,* WL/MLSE, Wright-Patterson AFB, OH 45433-6533
8. *Aerospace Structural Metals Handbook,* Mechanical Properties Data Center, Traverse City, MI 49684
9. British Standards Institution (BSI), Linford Wood, Milton Keynes, MK14 6LE, United Kingdom
10. International Atomic Energy Association (IAEA), WagramerStrasse 5, P.O. Box 100, A-1400 Vienna, Austria
11. International Standards Organization (ISO), 1 rue de Varembe, Case postale 56, CH-1211, Geneva 20, Switzerland
12. National Aeronautics and Space Administration (NASA), Structural Design Criteria Applicable to a Space Shuttle, NASA-SP-8057, National Technical Information Service, Springfield, VA 22151

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15. R. W. Hertzberg, p. 239 in *Deformation and Fracture Mechanics of Engineering Materials,* 2nd ed., John Wiley, New York, 1983, p. 239.

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7. Thermal Analysis

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7.1 INTRODUCTION

All packages that contain radioactive materials must be evaluated to determine their expected normal operating temperatures and their responses to the accident conditions specified in the regulations. The amount of heat produced, the amount or radioactive material carried, and the package design itself may affect the type of analysis of testing that must be carried out to convince the designer (and the regulators) that the package is safe in the transport environment. This chapter contains a discussion of (1) thermal design considerations of a Type B packaging for normal conditions of transport as defined in 10 CFR 71.71 and (2) the hypothetical accident sequence as defined in 10 CFR 71.73. The major issue in the thermal design of a packaging involves the conflict between passively removing the radioactive decay heat from its contents (with a small temperature gradient through the package) while passively protecting its contents from external heat sources. Although elevated temperatures in a package are not necessarily harmful to the public, such temperatures must not compromise other functional requirements of the package, such as containment, shielding, or criticality control.

The thermal regulatory requirements can be satisfied by subjecting the package design to analysis, by physically testing a prototype packaging, or by use of a combination of the two. However, testing is often preferred for packages that are built from components that are poorly characterized from a thermal standpoint or that may exhibit a phase change or chemical decomposition under the regulatory hypothetical accident conditions.

7.1.1 Design Considerations

This chapter presents methods for analyzing the requirements set forth in the regulations. These methods have been applied to the analyses of packagings in the past, follow good engineering practices, and should help the reader to avoid problems in the thermal analysis of any packaging.

A primary goal of the designer is to develop a package that will carry the radioactive material safely while meeting all applicable regulations; it is also important for the designer to recognize that a package should not fail if an accident of severity equal to or greater than the regulatory requirements occurs. A package should possess positive margins so that the licensing process can proceed smoothly. The designer should also consider the interactions among the structural, criticality safety, shielding, and containment functions of the package and the thermal characteristics of the package. To this end, a package designer should consider the effect that the

9-m (30-ft) drop and pin-puncture accident scenarios could have upon the thermal design of the package. Damage that would be considered to be minor from a structural standpoint could have major effects, however, upon the heat-transfer capabilities of the package.

7.1.2 Regulations

DOT has jurisdiction over the movement of radioactive materials in transport, but for Type B packages, DOT must also comply with the NRC regulations given in 10 CFR Part 71.

With the exception of specified maximum package-surface temperatures, the thermal performance of a package is not included in this discussion because elevated temperatures, in themselves, are not harmful to the health and safety of the public. However, all thermal limits are derived from requirements that are necessary to prevent the compromise of these functional objectives of containment, shielding and criticality control.

That the thermal capabilities of a package comply with the regulations is typicaly demonstrated via calculations, although the thermal properties of materials used in the package are usually gathered from physical tests. Many times this is the case because determining the thermal characteristic of a packaging by testing must be accomplished using a full-size prototype. The thermal testing of a scale model cannot be used to accurately predict the response of the full-size package because radiative, convective, and conductive heat transfer do not scale linearly in space and time.

7.1.3 Design Issues

As noted in the introduction, the major thermal design issue involves the conflict between passively removing the radioactive decay heat from a package while passively protecting the package from accidental external heat sources. Thus, a passive system (e.g., insulation) that might be used to protect the containment system from overheating in an accident involving a fire may exacerbate the difficulty of transferring the radionuclide decay heat from the containment vessel to the package surface (and ultimately to the environment) during the normal conditions of transport. An ideal passive thermal-protection system for a package is a thermodiode with a high-heat conductance from the containment vessel to the package outer surface and a high-heat resistance from the package outer surface to the containment vessel. An example of a thermodiode is a heat pipe with the wick or groove configured for unidirectional liquid-capillary pumping. However, heat pipes are vulnerable to damage even in the normal use of a package. Thus, a practical thermodiode is very difficult to achieve for a transportation package.

7.2 OVERVIEW OF THERMAL REGULATIONS

In this chapter, the emphasis is on the thermal design of a Type B packaging for conditions of transport, defined by 10 CFR Part 71. In addition to the normal conditions of transport, a packaging must be able to survive a hypothetical accident that involves a sequence of damaging events, during which the package undergoes a 9-m drop on an essentially unyielding surface, followed by a puncture test prior to a fire.

The portions of the 10 CFR Part 71 regulations that apply to the thermal design of a packaging are reproduced verbatim here. Where various parts of a paragraph apply to several design disciplines, only those parts of a paragraph that apply to thermal design have been included. Additionally, every effort has been made to duplicate the text of 10 CFR Part 71 to avoid introducing an interpretation of the regulations because that is reserved for the NRC.

In addition to the 10 CFR Part 71 requirements, the design must consider all other applicable regulations. Further information on this topic is given in Chapter 3 of this *Handbook.*

7.2.1 10 CFR Part 71

The performance criteria for the packages are given in 10 CFR 71.43, 71.71, and 71.73. General standards for all packages are established in 10 CFR 71.43. These standards include the temperature requirements that

- ". . . a package must be designed, constructed, and prepared for transport so that in still air at 38°C (100°F) and in the shade, no accessible surface of a package would have a temperature exceeding 50°C (122°F) in a nonexclusive use shipment, or 82°C (180°F) in an exclusive use shipment."

The conditions that represent normal conditions of transport are defined in 10 CFR 71.71. The conditions include the following temperature and heat-load conditions.

- "*Initial conditions.* With respect to the initial conditions for the tests in this section, the demonstration of compliance with the requirements of this part must be based on the ambient temperature preceding and following the tests remaining constant at that value between –29°C (–20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be considered to be the maximum normal operating pressure, unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the tests is more unfavorable."

- "*Conditions and tests — Heat.* An ambient temperature of 38°C (100°F) in still air, and insolation according to . . . [the information given in Sect. 7.5.2 herein]" [and] *Cold.* An ambient temperature of –40°C (–40°F) in still air and shade."

The conditions that represent the hypothetical accident are defined in 10 CFR 71.73. The test procedures, test conditions, and test include:

- "*Test Procedures.* Evaluation for hypothetical accident conditions is to be based on sequential application of the tests specified in this section, in the order indicated, to determine their cumulative effect on a package or array of packages. . ."

- "*Test conditions.* With respect to the initial conditions for the tests, except for the water immersion tests, to demonstrate compliance with the requirements of this part during testing, the ambient air temperature before and after the tests must remain constant at the value between –29°C (–20°F) and +38°C (100°F) that is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be the maximum normal operating pressure, unless a lower internal pressure consistent with the ambient temperature assumed to precede and follow the tests is more unfavorable."

- "*Tests.* Tests for hypothetical accident conditions must be conducted as follows:"

- "*Thermal.* Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800°C (1475°F) for a period of 30 min, or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800°C. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond any external surface of the specimen, and the specimen must be positioned 1 m (40 in) above the surface of the fuel source. For purposes of calculation, the surface absorptivity coefficient must be either that value which the package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater; and be convective coefficient must be that value which may be demonstrated to exist if the package were exposed to the fire specified. Artificial cooling may not be applied after cessation of external heat input, and any combustion of materials of construction, must be allowed to proceed until it terminates naturally. Although the current version of 10 CFR 71 is silent about the issue of insolation during the post-fire cooldown, the IAEA requires the application during cooldown. The NRC has, however, adopted "the view of the thermal experts who participated in developing the IAEA regulations. Those experts thought the effects of solar radiation may be neglected before and during the thermal test but that such effects should be considered in the subsequent evaluation package response."[1]

Routine determinations for each shipment are specified in 10 CFR 71.87. They include that, prior to each shipment of licensed material, the licensee shall determine that the accessible package surface temperatures of the package with its contents will not exceed the limits specified in 10 CFR 71.43 (g) at any time during transportation.

7.2.2 Regulatory Guides

The regulatory guides do not have the force and effect of regulations; however, they do indicate approaches to specific issues that have been considered by the NRC staff and found to be acceptable. Consequently, following the guidance in a regulatory guide should simplify the certification process.

7.2.2.1 Regulatory Guide 7.8 (ref. 2)

The load combinations that must be included in the evaluation of normal conditions of transport and the hypothetical accident to assure that the most unfavorable conditions have been considered are presented in Table 1 of *Regulatory Guide 7.8*. The table defines the temperature, heat load (including insolation and content decay heat) and internal pressure environment combinations that are to be used in the tests to determine that the package satisfies the criteria for normal condition of transport and a hypothetical accident.

7.2.2.2 Regulatory Guide 7.9 (ref. 3)

To aid in the preparation of a safety analysis report for packaging (SARP), the NRC has established *Regulatory Guide 7.9*, which gives a standard format and content of 10 CFR Part 71

application for approval of packaging for radioactive material. This regulatory guide (which is not a substitute for 10 CFR Part 71) is to assist in the preparation of an *application that thoroughly and completely demonstrates the adequacy of the package in meeting the regulations.*

The SARP preparation of the thermal evaluation of a transportation package is described in Sect. 3.0 of *Regulatory Guide 7.9*, which should identify, describe, discuss, and analyze the principal thermal engineering design of the packaging, components, and systems important to safety and to compliance with the performance of 10 CFR Part 71.

7.3 PACKAGE DESIGN ISSUES

The thermal loads on a package may produce effects that may damage the package such that it does not perform the function of protecting the public health and safety. The effects produced by the thermal loads and their impacts on the various package functions are described in Sects. 7.3.1–7.3.3.

7.3.1 Phenomena Produced by Thermal Load

A thermal load may cause pressure buildup in the containment vessel producing stresses. Thermal gradients in the package may also produce thermal stresses and other effects that may jeopardize the ability of the package to perform its function in transporting radioactive material in a way that protects public health and safety and the environment. These phenomena are discussed briefly in the following, along with various issues that the applicant must consider.

7.3.1.1 Pressure Buildup

In addition to the natural effect of temperature increases, many thermally driven phenomena can contribute to pressure buildup within the containment boundary of a package. These may include the effect of (1) the temperature increase on a fixed volume of a fluid; (2) a release of gas from encapsulated materials because of the failure of the encapsulation boundary (e.g., a tritium reservoir or spent fuel clad); (3) radiolysis of the contents or packaging components by the radionuclide decay products of the contents and (4) by thermal decomposition of the package components (including the contents and the seal materials). Several of these sources of pressure buildup are discussed in Sects. 7.3.1.1.1–7.3.1.1.5.

7.3.1.1.1 Encapsulation release. Radioactive materials are often contained in a sealed capsule and placed in the containment system of the packaging. Occasionally, gas pressures will build up inside the capsule. Under such conditions, the failure of an encapsulation system will increase the pressure in the containment system. The steady-state pressure following the release can be determined from

$$p = p_i \left[\frac{V_i}{V_i + V_e} \right] + p_e \left[\frac{V_e}{V_i + V_e} \right], \tag{7.1}$$

where V_i and V_e are the volumes of the containment vessel and the encapsulating system, respectively, and p_i and p_e are initial pressures in the containment system and encapsulating system respectively. Examples of encapsulation systems include spent fuel rods for pressurized-water

reactors (PWRs) and boiling-water reactors (BWRs) and special-form containers for plutonium and radioisotopes.

7.3.1.1.2 Radiolysis. The radiolytic release of gaseous material into the containment vessel can be caused by the energetic radionuclide decay products breaking the chemical bonds of compounds made up of at least one element that is a gas at existing temperatures and pressures in the containment vessel. If the gaseous product can migrate through the original matrix (made up of the initial compound) without chemically reacting with other materials in the material matrix, the partial pressure of the released gas will be added to the initial pressure in the containment vessel. The release rates will depend on the quantity, type, and energy of radiation from the source as well as the bonding energy of the elements making up the initial compound. The reaction rates will also depend on the physical dimensions of the material region as well as the number density of the compounds. In short, this radiolytic process can be analyzed using methods similar to other transport processes if the properties of the source material and the characteristics of the source term are known.

7.3.1.1.3 Radioactive decay. The pressure buildup in a package can result also from radioactive decay of the contents. For example, several processes that may cause pressure buildup in a containment system include

- Production of protium or helium from neutron interactions with the contents or the containment vessel material [e.g., (n, p) or (n, α) reactions.]

- Production of two helium atoms per decay of a tritium molecule which will cause an increase the pressure in the containment vessel via

$$T_2 \rightarrow 2\ He^3\ .$$

Another process that may increase the pressure in a containment system (should tritiated water be present) is

$$2T_2O \rightarrow 4He^3 + O_2\ .$$

Note that this process goes from a liquid phase to a gaseous phase with 2.5 atoms being produced per molecule of water.

These processes are relatively straightforward to evaluate. The initial neutron flux and energy spectrum and the appropriate cross section of the containment vessel material can be used to estimate the gas buildup in the containment vessel from the neutron capture. The production of helium from an (n, α) reaction in a solid material usually does not increase the pressure within the containment boundary as the migration rate of helium through many solids is quite slow. The pressure buildup by the gases produced from radioactive decay (such as of tritium as is demonstrated above) is also fairly easy to determine if the half-lives of the radioisotopes are known.

7.3.1.1.4 Thermal decomposition. The rate of thermal decomposition of materials usually increases with temperature such as might occur in an accidental fire environment. The Arrhenius rate equation[4] is

$$\dot{n}(T) = A^{\left[\frac{-E}{RT}\right]} , \tag{7.2}$$

where $\dot{n}(T)$ is the rate of decomposition at temperature T, A is a preexponential factor, E is an activation energy, and R is the universal gas constant. For a closed system, the pressure buildup at time τ is proportional to

$$p(\tau) \propto \int_0^{\tau} \dot{n}(T)dt = A\int_0^{\tau} \exp\left[\frac{-E}{RT}\right] dt , \tag{7.3}$$

where the temperature T is a function of time t.

Equation (7.3) can be integrated numerically if the temperature, as a function of time, has been determined for the hypothetical accident condition. Note that because the peak temperature in the containment system occurs after the prescribed fire is over and because the thermal decay time of the containment vessel (with contents) is usually long, the temperature history of the system may be required over a long time relative to the 30-min thermal fire condition.

7.3.1.1.5 Phase change. A phase change of a material during a temperature excursion such as one occurring during a hypothetical accident condition will increase the pressure in a containment vessel. This process may be reversible such that, providing no leakage occurs during the temperature excursion, the pressure will return to normal upon cooldown. The difficulty with the phase change is that it occurs when the containment vessel is at the highest temperature and, therefore, is the least capable of withstanding pressure increases because of its reduction of strength and modulus of elasticity with temperature.

7.3.1.2 Thermal Stresses

The thermal stresses in any material are a function of the temperature gradient through the material and the constraints on the ability of the material to expand or contract. For elastic materials with a slowly applied thermal load, the thermal stresses can be determined by using the solution to the heat-conduction equation as input into the equations of the theory of elasticity. For simple temperature distributions and simple geometries the solution to the thermal stress equation is relatively straightforward for plain stress or plain strain problems.[5–7]

Another form of thermal stress results from the difference in thermal expansion between two components that have an insufficient gap between them to accommodate the expansion differences without interference. This phenomenon can be deliberately used to join two different components (e.g., by shrink fitting) or may inadvertently happen because of inadequate design. Two examples of how these thermally induced stresses may occur in a transportation package design include:

- Failure to provide enough clearance between components to provide for the temperature increase of the loaded package (caused by the decay heat of the radioactive contents) above that of the empty packaging. Note this may occur under the steady-state, normal conditions of transport.

- Failure to provide enough clearance between the components to account for the cooldown of the package after a hypothetical accident condition when the package outer layer cools much

faster than the internal components, resulting in the cooler outer layer of the package "shrinking" temporarily over the hotter inner components.

Even though the thermal stresses are not treated the same way as pressure or impact stresses because they are secondary (or self-limiting) stresses, they still are used in setting the design criteria for the structural analysis of shipping casks containment vessels.[8] Hence, normal operation thermal stresses must be included when evaluating the effects of the 9-m (30-ft) drop and pin-puncture impacts.

7.3.1.3 Temperature Effects

The increase in pressure and the stresses produced by the thermal loads result from the increase of temperature in the package. The mechanical strengths of most materials tend to decrease with an increase in temperature. Thus, for example, an increase in temperature causes an increase in pressure, with the concomitant increase in the containment stress. However, the strength of the material decreases with the increase in temperature. Thus, the reduction in the design margin caused by the increased pressure is exacerbated by the reduction of the material strength.

7.3.2 Packaging-System-Component Issues

The packaging-system components include the components that perform certain packaging safety functions such as containment, shielding, and assurance of subcriticality. Other packaging-system components are those that aid in mitigating the effects of both normal and off-normal conditions on the safety function components. These components may include valves and sampling ports, passive cooling systems, impact limiters, and thermal insulation. Some of these packaging components are discussed in Sects. 7.3.2.1–7.3.2.5 with an emphasis on thermal design issues associated with each component.

7.3.2.1 Containment

The containment-system portion of a package may consist of a containment vessel for a Type B quantity of radioactive material with its associated gaskets and a closure. Double-containment may be required for nonmetallic forms of plutonium and other highly toxic radionuclides.

The most critical component of the containment system for a transportation package is often the sealing system between the head, through which contents are placed in the containment vessel, and the vessel itself. If the seals are elastomers, their ability to withstand temperatures in the several-hundred-degrees and above Celsius range is limited to rather short times due to thermal decomposition as discussed in 7.3.1.1.4. The expected temperatures during normal and hypothetical accident conditions should be compared with the maximum allowable seal-operating temperature for both short and long times.

The thermal design issues associated with the package containment include maintaining the temperature of the containment seals below the maximum allowable temperature for a specified lifetime gasket. If there is air or another gas in addition to the contents within the containment boundary at the time of closure, the thermal analyses must prove that the increase of gas pressure within the containment boundary resulting from the increase in the temperature of the gas will not

cause the seal or entire closure system to unseat or the containment to otherwise fail and allow a release of the gas to the environment.

7.3.2.2 Shielding

The primary thermal design issues associated with shielding include volumetric dimensional changes caused by thermal expansions or material phase changes. In addition, the shield material may melt or burn in a thermal accident such as a fire.

7.3.2.3 Criticality Control

The prime thermal design issues associated with maintaining the subcriticality of the fissile contents of a transportation package include ensuring that the temperature of the neutron poison components are below that at which the poison components will deform or disintegrate. Also, the thermally induced deformations in the package must not cause nuclear criticality of the fissile contents. Thus, the stresses associated with the thermal deformations must remain in the elastic region in order to limit the deformations and ensure geometric control.

7.3.2.4 Pressure Vents, Valves, and Sampling Ports

A package should be designed to accommodate the maximum pressure buildup in the containment vessel or the packaging. The use of vents to relieve a pressure buildup in the packaging external to the containment boundary is allowed to prevent excessive stress levels in the package components. The use of vents or pressure-relief devices (valves or rupture disks) causes a Type B package to be reclassified from Type B(U) to Type B(M) per the definition of Type B package contained in 10 CFR 71.4.

Valves and sampling ports are part of the packaging that are used to allow interrogation of the gas within the containment boundary in order to detect the buildup of undesirable radioactive and chemical products within the packaging. If failure of a valve should allow the escape of the radioactive contents of the package, the valve sampling port must be protected from unauthorized operation and must be provided with a sealed cover to retain leakage.

The prime thermal design issues associated with the need for a vent (external to the containment boundary) are associated with a change in the internal pressures in the packaging caused by temperature or altitude changes of the package. Additional considerations are the buildup of pressures in the package from the release of gases caused by changes of the physical state (i.e., evaporation) or the thermochemical decomposition of the packaging materials or components (such as combustion of any internal thermal insulation during a fire). The design of the valve and sampling port/enclosure systems must ensure that the temperatures of the components during the most severe thermal loading (usually a hypothetical thermal accident) do not exceed the maximum rated, short-time temperatures of the components and the system.

7.3.2.5 Thermal Protection

Insulation is a passive system that is often used to protect the containment, shielding, and criticality control systems from overheating in an accident involving a fire. However for normal conditions of transport, the insulation increases the temperature gradient from the payload across the packaging to the package surface. This increases the difficulty of the removal of the content decay heat to the package surface, thus resulting in higher temperatures of the package internals.

The enhancement of heat transfer from a package can be accomplished using a combination of materials and design concepts, such as insulation to protect the containment system from external heat sources, using conductors or surface enhancement techniques (e.g., fins) to remove the heat load generated by radioactive decay of the package contents, or using naturally circulating gases to passively remove the source heat from the package contents.

In some package designs, the insulation can serve also as an impact limiter. Likewise, fins on the outer surface of the package can function as an impact limiter for the package and as a method for enhancing the transfer of the source thermal energy from the package to the surrounding environment.

Another design consideration associated with the thermal insulation is that for many packages, the insulation is constructed from wood or wood products, bagasse or other cellulosic materials, or from other organic materials or foams. These materials may decompose at elevated temperatures with the release of combustible gases which may ignite in the presence of air, increasing the thermal input to the package. The use of materials for insulation (and impact energy absorption) which will decompose within the postulated maximum possible temperature and which might release additional thermal energy that can couple to the package is discouraged. The use of insulating and impact-absorbing materials that could produce pneumoconiosis, such as refractory ceramic fiberboard, is also discouraged.

An alternative to removing the heat from the contents to the surface of the package by conduction is to use a fluid to remove the heat by natural convection. The use of gases such as air or helium to remove the content decay energy by natural circulation may be appropriate if the fluid operates within a containment vessel that cannot be breached by maximum expected accident conditions.

The use of a gas that has special energy transport properties (such as helium) as a coolant external to the containment vessel is not recommended if the gas (as well as its special ability to remove the coolant decay heat) can be lost from the system during an accident, increasing the risk of loss of containment subsequent to the accident. The use of liquid coolants is discouraged unless the coolant vapor pressure is low over the maximum credible temperature produced in an accident.

In summary, thermal protection of a package from external thermal accidents in a way that will still allow the radionuclide decay heat from the contents to be transferred to the surrounding environment is not a trivial problem. The use of a passive system for thermal protection is required. The use of materials that will change phase at temperatures up to and including the hypothetical accident temperatures is not permitted. The use of materials that will decompose at temperatures within the normal to hypothetical accident temperature range is discouraged. Also discouraged is the use of coolants to transport the energy of the package contents from the package.

7.3.2.6 Impact Limiters

The primary function of the impact limiters is to reduce the effect of dynamic external loads acting on the package in an accident. However, for many packages, the impact limiter also functions as thermal insulation and will protect the package, including the containment system, from fire or other external heat loads acting on the surface of the package.

Because materials that exhibit good mechanical energy-absorption characteristics can have good thermal insulation properties, the prime thermal design issues for the impact limiter are the same as for the thermal insulation described in Sect. 7.3.2.5.

The accident sequence given in 10 CFR 71.73 places the 9-m (30-ft) drop and pin-puncture-impact events before the fire event, so impact-limiting materials that are treated as thermal insulators must remain in place during and after the impact. This may be difficult to demonstrate, but it is not precluded.

7.3.3 Effects of the Transportation Conveyance on a Package

The requirements for a package design are usually based on reasonable, but artificial, boundary conditions that can be encountered in storage, handling, and transporting of a single package. Criticality often is the only condition that is examined when considering a group of packages carrying fissile material in close proximity to one another, and that is considered when the transport index of a single package is calculated. However, a shipment of packages within a given transportation conveyance should be evaluated to determine that package-to-package interactions and package-to-conveyance interactions do not create mechanical or thermal loads in excess or that allowed on a single package during normal and hypothetical accident conditions.

The thermal design issues for a shipment of packages in a single conveyance occur only if the package contents are radionuclides. The radionuclide decay heat from a group of packages must be transferred through the walls to the exterior of the conveyance and to the outdoor environment or to an artificial environment maintained by a heating, air conditioning, or ventilation system. For example, when the thermal energy of the package contents is transported to the environment by passive processes, the increased thermal impedance from the outdoor environment to the containment vessels may cause the steady-state temperatures of the containment vessels to be higher than a single package transferring its heat to the immediately surrounding environment. In addition to the steady-state temperature, the effect of a thermal accident consisting of an engulfing fire about the conveyance must be determined for the shielding, criticality controls, and containment system for each of the packages. For the case in which the interior of the conveyance is environmentally controlled by a heating, air conditioning, or ventilation system, the effect of a failure of the environmental control system on the shielding, criticality control, and containment systems must be determined for both the normal transport condition and a hypothetical thermal accident condition.

7.4 MATERIAL PROPERTIES

The thermal performance of a transportation package is highly dependent on the thermal properties of the materials of construction and the operational environment. The typical package may use a wide variety of materials which are characterized to a greater or lesser extent. For example, some

materials of construction may consist of austenitic stainless steel or other metals that are very well characterized, with physical and thermomechanical properties available as a function of temperature. Any insulation or impact limiters that might be used in the packaging are generally constructed of material such as solid foams, wood, and/or other cellulosics that are often poorly characterized at ordinary temperatures and that may decompose at unspecified elevated temperatures. Single-component gases and atmospheric air are usually very well characterized.

The material properties used in the design of a package usually represent the mean characteristics of the materials. However, during the preparation of the safety analysis report at the conclusion of the design, the properties should ideally be the bounding values. The design process is directed at ensuring the performance of the package design under nominal conditions. The use of mean values of the properties will give the designer the most representative thermal response of the package. Bounding values of the properties should be employed to ensure the predicted temperatures of the critical elements, such as the O-rings, represent the upper bound of the actual temperatures during operation of the package.

The selection of the value of the material property to be used in the SARP will depend on the case considered. For example, for a package having an internal heat source, under normal conditions of transport the temperature gradient through the packaging (and consequently the cavity temperature) will be maximized using conservative (low) values for the material thermal conductivities. However, for a hypothetical accident, the temperature increase in the various package components will be maximized by using high values for the material thermal conductivities.

7.4.1 Source of Material Thermal Properties

The thermal physical properties of material may be listed in general or specialized references. The use of general thermophysics property data given in heat-transfer texts may be inappropriate for use in the design or analysis of transportation packages that are to contain Type B quantities of radioactive material unless it can be shown that those values accurately apply to the specific materials of construction of the package.

7.4.2 General Sources of Data

A very comprehensive compilation of thermophysical property data is maintained by the Thermophysical Properties Research Center (TPRC) of Purdue University. This compilation is published in 13 volumes and is edited by Y.S. Touloukian and C.Y. Yo.[9] Another source of general thermophysical properties is compiled by K. Raznjevic'.[10] This is a less comprehensive compilation than that of the TPRC, but it is more manageable in that it is available in a single volume. P. E. Liley[11] also gives general compilations and identifies the sources of thermophysical property data for both solids and fluids.

7.4.3 Source of Material Thermal Properties of Solids

The normal thermal conductivity and thermal diffusivity of many of the code-approved structural materials are given in Division II of the American Society of Mechanical Engineers (ASME) Pressure Vessel Code.[12] The *American Society for Metals (ASM) Handbook* also contains a collection of some of the thermophysical properties of metals.[13,14] The properties presented usually

include the thermal conductivity, density, and specific heat for a wide variety of materials. A one-volume abbreviated version of the ASM Handbook has been published by the ASM.[15]

The TPRC has compiled a six-volume set of thermophysical data for high-temperature materials.[16] The specific properties covered in each volume include:

- Density
- Melting Point
- Heat of Fusion
- Heat of Vaporization
- Heat of Sublimation
- Electrical Resistivity
- Specific Heat at Constant Pressure
- Thermal Conductivity
- Thermal Diffusivity
- Thermal Linear Expansion
- Thermal Radiative Properties: Absorptance, Emittance, Reflectance, and Transmittance
- Vapor Pressure

Generally, only materials with melting points about 540°C (1000°F) are included, except for materials within the categories of polymers, plastics, and composites. A material index for the entire work is included at the end of each volume.

The thermophysical properties of woods and other cellulosic materials used as insulation or impact limiters are not as readily obtained as are those for metals and other compounds. The reason for the lack of thermophysical property data for wood and similar materials is that these materials have a wide variation in density and moisture content, and, in some cases, are anisotropic materials. R. Summitt and A. Sliper have compiled some mechanical and thermophysical properties of wood, including expressions for thermal conductivity, specific heat and thermal expansion coefficients.[17] Other data concerning the thermophysical properties of wood are given in the literature.[18–21]

Other materials such as cane fiberboard, corkboard, and various types of other solid forms are used as insulation or impact-limiter materials in packages. The thermophysical properties of these materials also are a function of the density and composition of the material. F. F. Wangaard, C. Q. Livingston, K. O. Arnold, and R. H. Kenan[22] give an example of the range of values of cane fiberboard as a function of density. C. Q. Livingston[23] reports on the thermal conductivity of a given density corkboard as a function of temperature. J. C. Anderson[24] has reported on some experiments that involve the charring of Celotex™ and presented some thermal properties of the Celotex™ for use in packages. He also described some data on the pyrolytic products of cellulose.[25] Hensel and Gromada performed experiments on cane fiberboard under hypothetical accident conditions. Coupling the experimental data with a mathematical simulation of the experiment, they determined a set of thermophysical properties for the cane fiberboard that can be used to perform an analysis of the response of a package to the hypothetical accident condition. These properties include the temperature variation of the thermal conductivity, density and specifies heat of the uncharred fiberboard, as well as the density, specific heat and temperature variation of the thermal conductivity of the charred fiberboard.[26,27]

In addition to the thermal radiation properties given in the refs. 9–11 and ref. 16, E. R. G. Eckert[28] gives the thermal radiation properties of some solids. Likewise, R. Siegel and J. R. Howell[29] also present thermal radiation properties of various materials for various surface temperatures and for

incident solar radiation with a surface temperature of 21°C. M. F. Modest[30] also gives the emittances of various materials as functions of the average surface temperatures and the extraterrestrial solar absorptivity. Siegel and Howell, Modest, and Eckert all present the normal-total emissivity. The variation between tabulated values of the surface emittance of a given material may result from such characteristics as the roughness and corrosion product formation (oxidation) of the material surface.

To estimate the total hemispherical emissivity from the total normal data, Modest[30] makes the following observations:

- materials with high emissivity tend to have a hemispherical emissivity that is 3 to 5% larger than the normal emissivity and
- materials with low emissivity tend to have a hemispherical emissivity that may be up to 25% larger than the normal emissivity.

Note that the difference between the normal emissivity and the total emissivity may be as large (or larger) than the variation between tabulated values of the normal surface emissivity.

7.4.4 Sources of Material Thermal Properties of Fluids

A very useful compilation of thermophysical properties of fluids is given by N. B. Vargaftik.[31] The fluids covered range from pure substances in the form of monatomic and diatomic gases and their condensed state to elemental liquid metals. Also covered in this compilation are mixtures of gases as well as some aqueous solutions. P. E. Liley[32] also gives a useful compilation of thermophysical properties of various fluids. The compilation does not cover the range of temperatures and pressures of the fluids covered by Vargaftik. However, Liley does give thermophysical properties of some fluids not covered by Vargaftik.

7.4.5 Determination Of Thermal Properties

The thermal property data should be determined by American Society for Testing and Materials (ASTM) tests.[33] These tests include, but may not be limited to, the following examples:

Test No.	Test title
ASTM C 201	Thermal Conductivity of Refractories
ASTM C 351	Mean Specific Heat of Thermal Insulation
ASTM E 1225	Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique
ASTM E 1461	Thermal Diffusivity of Solids by the Flash Method
ASTM D 2766	Specific Heat of Liquids and Solids

Other ASTM tests are also used to determine the thermal properties of both general and specific materials.

7.5 SOURCE CHARACTERIZATION

Three sources of heat must be considered in the design of a packaging to transport radioactive materials: (1) heat released by the radioactive contents, (2) insolation, and (3) heat received from the fire during the hypothetical accident. The characterization of each of these is presented in the following paragraphs.

7.5.1 Heat from the Contents

Radioactive decay results in the creation of heat from the slowing down and the capture of the decay products. One of the initial steps in the characterization of the contents is the specification of the decay heat rate from the packaging contents. It is important to recognize that the regulatory interpretation of contents heat release is usually based upon the assumption that the radioactive decay heat is deposited at the location of the decay process. The deposition of the decay energy at a location along the path of the decay product from the point of birth to the point of annihilation of the radiation is considered less limiting than the assumption that the decay heat is deposited at the point of birth. Assuming that the decay energy might be spread over the path would result in a decrease in energy per unit volume with a reduction in the actual temperature, as well as the temperature gradients, in the packaging materials.

The assumption that the decay heat is deposited in the package contents is generally valid for short-range decay products such as alphas and betas, as well as soft gammas. Plutonium is an example of radioactive material where virtually all the decay energy is in short-range products. Spent fuel and high-level waste are examples in which a substantial amount of the decay energy is in long-range products (e.g., hard gammas and neutrons). These long-range products will deposit some of their energy in the package contents. The remaining energy will be deposited in the packaging (primarily in the shielding), with some escaping into the environment. The amount of long-range decay energy that escapes a package is small—the regulatory dose rate of 200 mrem/h is roughly equal to 0.2 W/m^2 of 1 MeV gamma rays.

The heat source should be chosen and the package should be evaluated based on the most conservative assumption (which generally means a source having the largest thermal output) for which the package would ever be used. If the source is a single isotope (e.g., ^{137}Cs), the heat source can be readily calculated by knowing the number of curies the package is designed to carry and the energ(ies) of the isotopes' emissions. Occasionally, the daughter product(s) of the isotope and the energies of its decay chain become important and must be considered. However, when the contents include spent fuel (assemblies or fuel rods), the computer program ORIGEN is typically used to compute both the heat released by the decay process and the rate of emission of neutron and gamma radiation. The following documents are available to provide guidance for the determination of the amount of decay heat released by spent fuel:

Regulatory Guide 3.54, Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation.[34]

American National Standards Institute, Inc.,/American Nuclear Society (ANSI/ANS)-5.1-1979, *Decay Heat Power in Light Water Reactors.*[35]

NUREG/CR-5625, ORNL/6698, Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data.[36] [O. W. Hermann, C. V. Parks, J. P. Renier, September 1994]

References 34 through 36 were developed to provide a bounding limit on the decay heat as a function of the burnup, specific power in the reactor, and decay time. Other parameters related to the fuel (e.g., initial enrichment) and reactor operating history were considered in the work. As a rule of thumb, the decay heat of a spent fuel assembly after X days out of the reactor will be affected by the reactor operating history in the last X days prior to discharge. Thus, after 3 to 5 years of decay time the decay heat is very insensitive to any parameters but the burnup, average specific power (or power density) (burnup divided by time in the reactor, in Mw/T), and decay time. Reference 36 provides a study that demonstrates the sensitivity of the reactor operating history on the decay heat.

7.5.2 Insolation

The solar heat received by a package can be approximated as a cosine function of time with a 12-h half-period. The values for insolation in the table in 10 CFR Part 71 represent the solar heat incident on a surface during this 12-h period. The values of the insolation flux required for thermal analyses are given in Table 7.1. These are average values specified by International Atomic Energy Agency (IAEA) Safety Series No. 6 for a 12-h time period. The peak value is based on the maximum value of a cosine distribution of the insolation with a 12-h half-period with an average flux based on the surface form, location, and orientation as shown in Fig. 7.1.

The choice of the insolation data from Table 7.1 to be used for the thermal analysis for normal conditions of transport is dependent on the geometry of the package outer surface. Most packagings should be designed based on the absorbed surface flux averaged over a 12-h period applied as a steady surface heat flux. However, for the case in which the packaging has a thin external shell covering a thermal insulation, the package surface temperature should be based on the peak insolation flux given in Table 7.1. The interior temperature field should be determined by the application of the 12-h averaged heat flux applied to the external surface. The use of the

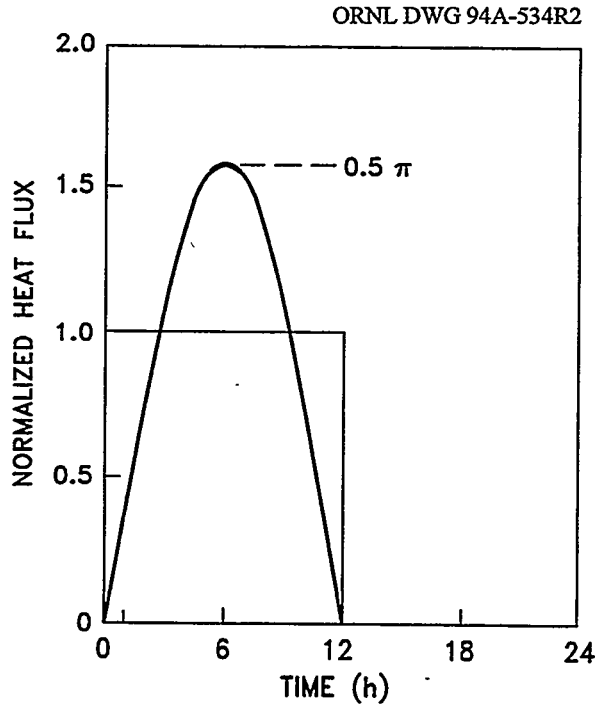

Fig. 7.1. Normalized insolation flux. The scale values are based on the 12-h average given in Table 7.1.

Table 7.1. Recommended values of insolation
(The given fluxes are the absorbed energy flux on the surface of the package)

Form and location of surface	12-h average (W/m^2)	Peak (W/m^2)
Flat surfaces transported horizontally		
Base	None	None
Other Surfaces	800	1260
Flat surfaces not transported horizontally	200	315
Curved surfaces	400	630

12-h averaged insolation applied as a steady boundary condition is the simplest condition for determining the temperature field within the packaging for normal conditions of transport. This steady condition, when applied with the other boundary conditions, will result in a conservative estimate of the internal temperature of the package contents; that is, an overestimate of the temperature. If a more nearly accurate calculation of the internal temperature field is desired, the insolation can be treated as a periodic steady-state heat flux with a period defined as 12 h on (with the 12-h averaged heat flux) followed by 12 h off (no heat flux). However, it is important to remember that the total surface heat flux includes the steady-state heat flux from the decay heat of the contents in addition to the insolation flux.

The insolation heat flux given in Table 7.1 is equal to the absorbed heat flux on the surface of the package. If the absorptivity of the solar spectrum by the packaging surface is assumed to be unity, the insolation heat flux given in Table 7.1 is equal to the absorbed heat flux on the surface of the package. Because the absorptivity is dependent on the surface condition (cleanliness, scratches, etc.) which may change with time and usage, this assumption is prudent and consistent with the NRC interpretation of its regulation 10 CFR 71.71(c)(1).

7.5.3 Thermal Input During Hypothetical Accident

The minimum thermal input for a hypothetical accident is defined by 10 CFR 71.73(c) (3). The heat input to the packaging is based on a 30-min exposure of a packaging surface having an absorbitivity of 0.8 to a radiation environment of 800°C (1475°F) with an emissivity of at least 0.9. This constitutes an initial radiation flux of 55 kW/m^2 to a package at 38°C (100°F). In addition to the radiation heat flux, there is also a convective heat flux applied to the package. The convective heat flux depends on the temperature of the package surface and the bulk-fluid temperature surrounding the package (generally air) as well as the amount of disturbance in the fluid. In a quiescent fluid, natural convection occurs caused by the buoyancy from the density gradients in the air near the package surface. In a fire, convection from the hot, moving fluid heats the package surface. The convection coefficient is a function of the velocity of the moving fluid. The convective heat flux also depends on the composition of the fluid.

The regulatory minimum heat flux to be applied to a package surface as a function of surface temperature in a hypothetical accident is shown in Fig. 7.2. For a regulatory thermal test in a furnace, radiation constitutes about 90% of the total heat flux of about 60 kW/m^2, with the remainder from natural convection. For a regulatory fire test, radiation constitutes between 75 and 90% of the total heat flux, with the remainder from convection of the heated air at various velocities normal to the axis of the horizontal cylinder. In both cases, the environment is assumed to be air at 1 atm.

7.5.4 Combinations of Heat Sources

For normal conditions of transport, the total heat source is made up from the decay heat of the contents and the insolation. This heat is rejected from the surface to the environment by thermal radiation and natural convection.

Under the hypothetical fire-accident condition, the total heat source (as seen by the packaging) is the sum of the decay heat from the radioactive contents contained within the packaging plus the thermal input from the externally imposed fire. Usually, the thermal input to the packaging from the fire far exceeds the energy input from the radioactive source contained by the package. Under these conditions the temperature of the external surface of the package can increase rapidly causing heat to flow toward the radioactive source contained in the cavity of the packaging. In some situations, the external surface of the package can become hotter than the surface of the contained radioactive source. This would cause the thermal energy from the radioactive source to remain with the package and the temperature of the package and source to increase. Following the accident and the removal of the external heat source, the package temperatures would decrease, and the thermal equilibrium would once again be established with the environment.

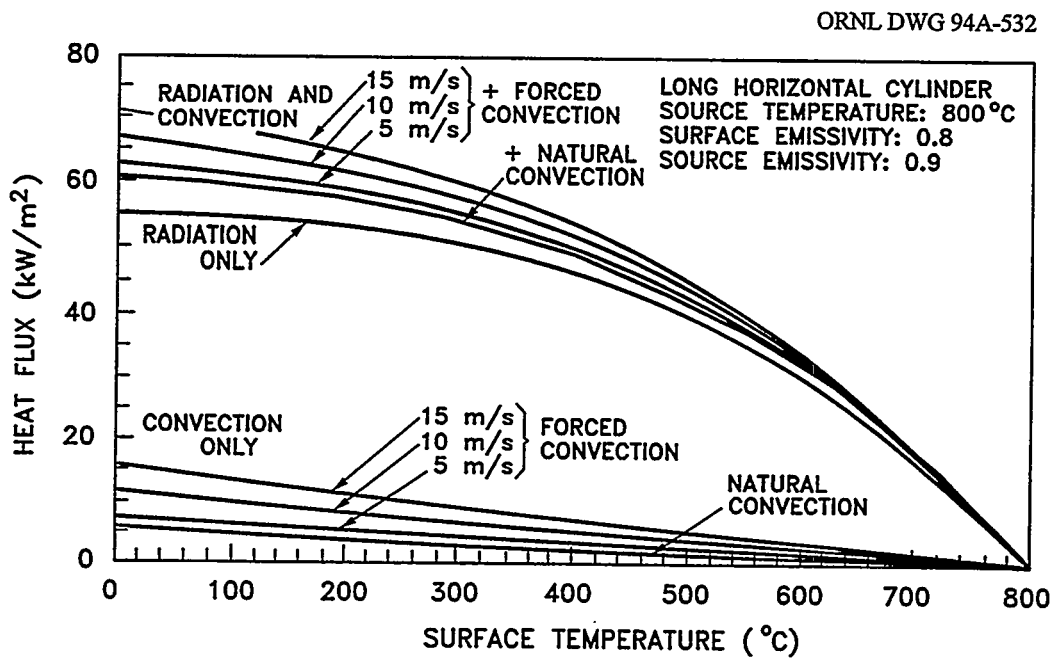

Fig. 7.2. The regulatory hypothetical accident thermal heat flux is dependent on the surface temperature and the type of convection.

7.6 MODES OF HEAT TRANSFER

The primary modes for transferring the radioactive decay heat from the contents of the package to the environment or for transferring the heat from a fire or other accident into the packaging are thermal conduction, convection, and radiation. The total energy transport is usually by a combination of all of these modes. Usually, the heat transfer within the packaging is by conduction; the heat transferred into and out of the package is by convection and radiation. Other mechanisms for transferring heat exist, such as phase change with mass transfer, but these will not be discussed in this section.

7.6.1 Steady-State Conduction

In most cases, it will be desirable to calculate the steady-state temperature distribution throughout a package assuming realistic linear and/or nonlinear boundary conditions (such as thermal radiation or natural convection). If the analysis is not based on simple shapes or the boundary conditions vary in space and time, the heat-transfer equations are usually solved numerically using finite-element or finite-difference computer codes. Mathematical idealizations are usually necessary to obtain analytical solutions to the generalized heat-conduction equation. These idealizations usually include the assumption that the materials are isotropic—that is, have properties that are independent of the direction of heat flow—and independent of temperature.

Boundary conditions, like the heat-conduction equation, are idealized to make the solution to the heat-conduction equation possible. These boundary conditions are usually made to be linear in temperature with isothermal or isoflux conditions along the boundaries. The boundary conditions are usually one of these types:

- constant temperature along the boundary;
- constant heat flux; or
- heat flux that is linear in temperature, such as a convective flux.

Note that the heat-flux condition includes the scenario in which no heat crosses the boundary (commonly called an "adiabatic" boundary). Also, the convection coefficient at a fluid-solid boundary is assumed to be invariant with the boundary temperature. The restriction to linear boundary conditions requires that if there is radiant energy transported to the boundary, the boundary condition must be linearized using, for example, a convection coefficient of a form that approximates radiation heat transfer, such as

$$h_r = 4\sigma\Im_{12}T_m^3 \ , \tag{7.4}$$

where $T_m = [(T_1 + T_2)/2]$ and σ and $\Im_{12}$ are defined as the Stefan-Boltzmann constant and the radiation interchange factor respectively. The temperatures T_1 and T_2 refer to the environment and the package.

Techniques for solving the idealized heat-conduction equation with various boundary conditions are given in many books on applied mathematics.[37–41] For an example in which the heat-conduction (diffusion) equation and boundary conditions are a Sturm-Liouville system, the characteristic functions are orthogonal. This orthogonal system allows many techniques to solve the heat-conduction equation with specified boundary conditions. Many of these techniques are

based on the application of various types of integral transforms.[42–44] The solutions to heat balance equations with the appropriate boundary and initial conditions often involve special functions with special characteristics.[45–47] The values of many of these functions for various arguments are tabulated in NBS-AMS 55 (ref. 47).

Many solutions to the simplified heat-conduction equation with simplified initial and boundary conditions are available. Perhaps the most widely available book of solutions is given in the classic on heat-condition by H. S. Carslaw and J. C. Jaeger.[48] A catalogue of even more solutions than given by Carslaw and Jaeger is the report written by J. H. VanSant.[49] An extended list of references is given by P. J. Schneider.[50]

7.6.2 Combined Modes

A real package will require the consideration of combined modes of heat transfer to remove the source heat or determine its response to a thermal accident. Examples of combined modes at various locations include:

- outer surface boundary condition: radiation and convection;
- finned outer surface boundary condition: radiation, convection, and conduction;
- thin gaps: radiation and conduction; and
- enclosures: radiation, conduction and convection.

Usually, convection is neglected (except at the outer surface boundary) because of the difficulty in its formulation. The neglect of convection in gaps and enclosures will result in an overestimation of the internal temperature under steady-state conditions, but an underestimate of the internal temperature under hypothetical accident conditions.

7.6.3 Finned Surfaces

Finned surfaces are frequently used to enhance the heat transfer from a package to the environment. The effect of the enhanced heat transfer is to reduce the package's surface temperature below that of an unfinned package.

The energy transport by a finned surface is a combination of:

- conduction along the fin,
- convection at the package surface and the fin surface, and
- radiation from the package surface and the fin surface to each other and the environment.

The surfaces of the fins are not isothermal, so a computer calculation may be necessary to determine the heat transferred from the package surface to the outside environment.

An approximate method for estimating the temperature of a finned surface is given by L. B. Shappert.[51] This method involves performing a heat balance between the decay heat and insolation with the heat convected to the ambient air and radiated to the surroundings. The effective emissivity of the finned surface for various geometries and surface emissivity is given by E. M. Sparrow and V. K. Johnson.[52]

External fins are vulnerable to damage during normal package-handling operations and may be damaged during the 9-m (30 ft) drop and pin-puncture impacts covered in 10 CFR 71.73. The effect of the degradation of the heat transfer provided by damaged fins should be considered. It is also important to note that the extended surface area that fins offer also provide areas that can become contaminated during operations, particularly the underwater loading and unloading of spent fuel into a large package. A finned surface is more difficult to decontaminate than an unfinned surface (see Sect. 12.3.4); therefore, this difficulty should be considered in the thermal design process.

7.7 COMPUTER PROGRAMS

The numerical simulation of heat flow through a conducting medium can usually simulate physical systems more realistically than can analytical methods. The general heat-conduction equation is

$$\rho(T)c_p(T)\frac{\partial T}{\partial t} = \nabla \cdot [k(T)\nabla T] + q'''(x,y,z,t) \quad , \tag{7.5}$$

where T is the temperature as a function of position and time, q''' is a volumetric heat source that can vary as a function of position and time, ρ and c_p are the density and specific heat of the material that can vary as a function of temperature, and k is a thermal conductivity that may be anisotropic and can vary as a function of temperature. The numerical simulation of heat conduction can also treat nonlinear boundary conditions (such as thermal radiation or natural convection) as well as geometric and boundaries not based on simple shapes.

Numerical methods were used to solve the heat-conduction equation for various initial and boundary conditions long before the advent of high-speed digital computers. These techniques, based on finite-difference methods, included the method of relaxation for solving one- and two-dimensional steady-state problems,[53–56] and graphical techniques, such as the Schmidt plot for solving one-dimensional transient problems.[53,54]

The importance of numerical analysis in the solution of heat-conduction problems is illustrated by the publication of a handbook devoted strictly to numerical heat transfer.[57] Most of the emphasis on numerical methods for solving the heat conduction equations are based on finite-difference and finite-element techniques.[58] The finite-difference method, the older method, provides a pointwise approximation to the heat-conduction equation. The newer finite-element method provides an average approximation to the heat conduction equation over the finite element. Both methods are described in the literature.[57,59,60] In addition, Monte Carlo techniques are also used to solve the heat conduction equation.

7.7.1 Finite-Difference Computer Programs

Finite-difference techniques are very useful for solving the heat-conduction equation. The finite-difference equations can be derived by either substituting a Taylor's series expansion of a temperature into the heat-conduction equation or by writing the equations for the conservation of energy across the faces of and wither a control volume (which is the method for the derivation of the heat conduction equation). The mesh is oriented along a coordinate system line. For a cartesian coordinate system, the elements are rectangles [two-dimensional (2-D)] or rectangular

bricks [three-dimensional (3-D)]. The mesh spacing is controlled by features within the conducting body, and the spacing propagates throughout the body. For nonconforming boundaries, which are bounding surfaces not described in terms of a fixed value of any one of the coordinates in the conduction region, the mesh cannot model the surface area and the volume of the body. Further discussion of the methods of solving a system of finite-difference equations is given in the literature.[61–65]

7.7.2 Finite-Element Computer Programs

A finite element is a subdivision of a continuum in which a partial differential equation applies. The problem unknowns (temperature, pressure, etc.) are associated with each node that forms a corner of the finite element. The finite-element method is a mathematical procedure for satisfying the partial differential equation (in an average sense) over the finite element. The finite-element methods require that an integral model of the partial differential equation be constructed. The methods available include the use of variational principles (especially for steady-state heat conduction) or weighted residuals for the general heat conduction problem.[66–69]

The use of the finite-element method for thermal analyses evolved from the use of the method for structural analysis of air frames. In fact, it has become very popular during recent years to use the same computer program for thermal analysis and structural analysis. In theory, this is excellent practice because it simplifies the problems of transporting information from one type analysis to the other. However, note that the mesh that results in the best solution for the thermal analysis may not be the mesh that results in the best solution for the structural analysis.

7.7.3 Transient Calculations

A number of algorithms are available for solving the transient problem of heat conduction. These algorithms include the explicit method (also known as the forward difference or Euler's method), in which the temperature distribution at any time can be easily calculated from the temperature distribution at an earlier time based on the time derivations at the earlier time. For the implicit method (also known as the backward difference method), the temperature distribution at a time is not completely determined by the temperature distribution at an earlier time, but it is based on the use of the time derivative at the new time. Other methods that combine the implicit and explicit methods include the Crank-Nicolson method, which uses the average of the time derivative at the old and the new times.[70,71] The explicit method is stable for time steps

$$\Delta t < \frac{(\Delta x)^2}{2\alpha}\left[1 + \left(\frac{\Delta x}{\Delta y}\right)^2 + \left(\frac{\Delta x}{\Delta z}\right)^2\right]^{-1} , \tag{7.6}$$

where Δx, Δy, and Δz are the distances between the spatial grid points and α is the thermal diffusivity. Thus, an increase in the dimensionality of the problem results in a reduction of the time steps for stability of the solution.

For a one-dimensional system with convection at the boundary,

$$\Delta t < \frac{(\Delta x)^2}{2\alpha\left(\frac{h\Delta x}{k} + 1\right)}, \tag{7.7}$$

where h is the convection coefficient, and k is the conductivity of the solid material.

For nonlinear cases, the time steps for stability may be further reduced. The Crank-Nicolson method, while subject to numerical oscillations, is conditionally stable for linear problems and has good stability for weakly nonlinear problems. The implicit method is unconditionally stable. Thus, for a given accuracy, longer time steps can be used for the Crank-Nicolson and implicit methods than for the explicit method. The relative accuracy of the Crank-Nicolson method and the implicit method is a function of the number of nodes and the size of the time step. Examples of the relative accuracy of the three methods are given in the literature.[72]

Some heat-transfer computer codes will allow an option of using either fixed- or variable-length time steps. The variable-length time steps are especially valuable for the situation in which a large heat flux affects an element and results in a large temperature increase within the element over the time step. The use of short time steps, which limit the temperature increase within the element during the time period to a fraction of the maximum temperature increase over the problem time, will result in the reduction of the calculated amplitude of the temperature oscillations over time. As the temperature approaches a final temperature, the time step can be increased without the development of the temperature oscillations. A typical initial time-step size may be 1% of the time constant of the smallest element with the system, where the time constant τ is defined as

$$\tau \equiv \frac{(\Delta x)^2}{\alpha}, \tag{7.8}$$

where Δx and α have been previously defined. The variable time step may be allowed to increase (1) to the time constant of the smallest element in the system or (2) to a time consistent with that of a specified maximum allowable temperature change in the problem.

7.7.4 Computer Codes

A number of codes that are used in the thermal analysis and design of packaging have been developed. These codes use either finite-difference or finite-element methods of solutions. In the past, these codes were designed primarily for use in large, mainframe computers. However, as the personal computers have become faster and more powerful, the codes have been adapted for use on these systems.

Examples of codes that can be used in the thermal analysis of packages for each numerical method are given in Table 7.2. The programs will run on either UNIX—based platforms or International Business Machines (IBM)-compatible personal computers. This list of programs does not include all thermal programs that are used for the analysis of a package, nor does the listing of a program constitute endorsement of the program.

Table 7.2. Computer programs used for the thermal analysis of transported packages

Program[a]	Contributor	Method[b]	Hardware
ANSYS	Swanson Analysis Systems, Inc	FE	UNIX, PC (≥ 386)
SINDA/G	Network Analysis Associates, Inc.	FD	UNIX, PC (≥ 386)
COYOTE II	Sandia National Laboratories	FE	UNIX
HEATING 7.2	Oak Ridge National Laboratory	FD	UNIX PC (≥ 386)
P/THERMAL	MacNeal-Schwendler Corp.	FE/FD	UNIX
TOPAZ	Lawrence Livermore National Laboratory	FE	UNIX, PC (≥ 486)

[a]This table provides a partial list of programs that can be used. The listing of a program does not constitute endorsement of the program.

[b]FE refers to finite element, and FD refers to finite difference.

The codes in Table 7.2 all treat linear and nonlinear steady-state and transient problems. They can use isotropic and anisotropic material with temperature-dependent thermal properties. The codes can treat problems with volumetric heat generation in various regions. Both steady-state and time-dependent boundary conditions that include temperature and heat flux as well as convection and radiation from the surface of a package to the surroundings can be applied to a problem. The codes can also treat initial conditions that are uniform or vary throughout the volume being analyzed. Several of the codes are able to treat some special situations. These situations may include thermal resistance across a gap, radiation across a narrow gaps, and radiation across an enclosure. Other special situations may include phase change and/or mass transfer within the package.

Each of the codes has both advantages and disadvantages. However, the disadvantages are usually not serious and can, in part, be mitigated by a skilled user who can

- develop correct model node network,
- specify correct initial and boundary conditions,
- specify appropriate material properties (including engineering units),
- select proper initial time steps, and
- interpret results.

The importance of the qualifications of the user of the code cannot be overemphasized.

7.7.5 Quality Assurance of Thermal Codes

The computer software used in the design analysis of a transportation packaging should be subject to the supplemental requirements for computer program testing that are given in Supplement 11S-2 of NQA-1-1989.[73] Additional guidance for software quality assurance (QA) is included in QA-2a-1990, Part 2.7.[74] For additional information concerning QA as applied to software, refer to Chapter 11, "Quality Assurance."

Testing shall be the primary method of software validation. The purpose is to ensure that the software produces correct results for the test cases. The testing consists of running test problems and comparing the results from alternative methods. These methods include a comparison with analysis without computer assistance:

- the results from another validated computer program,
- experiments and tests,
- standard problems with known solutions, or
- confirmed published data and correlation.

Solutions to many problems in heat conduction are found in the literature, including some graphic results.[48–50] A catalog of sample problems for transportation package designs is included in Appendix B of this handbook. Whatever method is used to validate a compute code for use in evaluating a transportation package, the test problem should be similar to the package that is to be analyzed.

7.8 NORMAL CONDITIONS OF TRANSPORT

A popular misconception is that hypothetical accident conditions are the most limiting aspect in the design of a package. In fact, normal conditions of transport can be more limiting because the acceptance criteria are more restrictive. For example the hypothetical accident requires consideration of a fire with a flame temperature of 800°C (1475°F), which is a very severe environment for any packaging. However, conditions when the ambient temperature is 38°C (100°F) may prove to be more restrictive, even if assuming the package is transported by an exclusive-use vehicle, because the accessible surface may not exceed 82°C (180°F).

7.8.1 Thermal Design by Analysis

The design of a package by analysis allows the designer to make estimates of the thermal safety margin that is built into the package. However, the design of complex packages by analysis may require sophisticated methods. These methods include the mathematical simulations of the heat flow using numerical methods and a digital computer or by using manual calculations based on "classical" mathematical techniques. Each of these methods will require the appropriate boundary conditions and initial conditions to accurately predict the temperature distribution within the package.

7.8.1.1 Surface Temperature

The "driving force" for the removal of heat from a package is the difference in the temperature of the package system and the surrounding environment. The surface temperature of a package is determined from the balance of the heat input with the heat rejected to the environmental ambient temperature. Heat from the contents combined with insolation defines the amount of heat that must be transferred from the surface of a package to the environment. Natural circulation convection and radiation heat-transfer modes are the only mechanisms that are available to remove heat from a package.

Artificial cooling systems are allowed, and they can be very useful for operational efficiency. The safety analyses, however, must be based upon the condition of an inoperable artificial cooling system to provide assurance that its failure does not compromise the ability of the packaging to perform its function. For example, a packaging that is designed to transport radioactive materials that release large amounts of heat, such as that generated by ^{238}Pu, will become very hot during transport. The inclusion of a cooling system (such as a fan to move air across the exterior of the packaging) would reduce the temperature of the packaging and the hazard associated with handling a very hot packaging. Under normal operating conditions, this may be an advantage and benefit package handlers during most shipments. However, the safety analysis must be based upon the assumption that the fan or cooling system is inoperable.

7.8.1.2 Tolerances

Each of the package components has tolerances on the dimensions, material properties, and content heat loads. Usually, the effect of the tolerances on the material properties and content heat loads on temperatures are easily calculated for simple, linear systems. For complex, nonlinear systems, the effect of tolerances on the material properties and the temperature can be determined by numerical studies using computer models. However, numerical studies that are used to examine the effect of dimensional tolerances on the temperature distribution may be difficult to perform. The construction of meshes for 2- and 3-D computer analysis is often laborious, and changes in the mesh geometry is required for each change in tolerance to be investigated. Thus, studies of the effect of tolerances on the temperatures in a package may most easily be performed by an analysis of an appropriate one-dimensional (1-D) simulation of the packaging or components.

7.8.1.3 Gaps

Gaps are built into a packaging to ensure ease of assembly and disassembly of the components of the system and to ensure clearance to reduce or to eliminate thermally induced loads between the components. Gaps may also occur if a component that increases in density with a decrease in temperature (e.g., a lead gamma shield) is cast into the packaging. Usually, for packages containing heat sources, the gaps between the inner and outer components will decrease in thickness because of the higher temperature in the interior of the packaging. But, if the packaging is subjected to an external heat source such as a fire for a short period of time, the gaps between the inner and outer components will increase in thickness because of the higher temperature in the outer regions of the packaging.

Heat transfer across narrow gaps is primarily by conduction across the gas and radiation through the gas in the gap. Thus, as the gap tends to close, the heat transferred by conduction increases, with a reduction in the temperature across the gap. In the limit, as the gap closes, the heat transfer is by contact conductance, and the temperature difference across the gap approaches zero. Heat transfer across wide gaps is primarily by convection at the surfaces of the gap and radiation across the gap.

7.8.1.4 Modeling Precision

The most nearly accurate representation of a packaging can be constructed by developing a 3-D mathematical model. Often, however, a 2-D model or even a 1-D model will adequately describe a packaging.

The number of mesh elements for approximately the same modeling precision scales roughly as the power of the dimensionality of the problem. The number of mesh points needed to describe one element increases likewise with the dimensionality of the problem. Thus, for a given model precision, the cost (including the mesh generation time as well as computer storage space and operating time) increases rapidly with the dimensionality of the problem.

To potentially reduce the cost of solving a given problem, a reduction of the dimensionality of the problem should be considered. For example, a 3-D problem can be reduced to a 2-D problem if the length of the packaging is much greater than the other dimensions of the packaging in a plane normal to the length. If the radionuclide content length is much greater than the other dimensions of the contents in a plane normal to the length and is concentric with the packaging, the packaging can be analyzed as a 1-D problem in the plane normal to the axis.

In the process of reducing the dimensionality of the model, care must be taken in the assumptions used in the dimensions and operating characteristics of the reduced dimensional model to ensure that the results of the analysis overpredict the temperatures in the packaging.

7.8.2 Thermal Testing

Component and complete package testing are both appropriate for identifying and defining the performance of a package during normal conditions of transport. Component testing involves making thermal measurements on a specific packaging component to determine the rate that heat is transferred from one component to another within the package. This type of testing is usually accompanied by a series of analytical evaluations of the test to determine the value of the material property or the effective heat-transfer coefficient associated with the component that is being tested. Once the appropriate material property or effective heat-transfer coefficient has been determined, then that information can be used with confidence in the analysis of the thermal performance of the entire package.

Complete package testing is the evaluation of the entire package under conditions that represent normal conditions of transport. Testing is not recommended as a design technique because the design-optimization process requires parametric variation of many design variables, and to examine each parameter empirically could become prohibitively expensive. However, testing to demonstrate the performance of the entire package following completion of the optimization by analysis is often justified to verify the results of the analytical optimization studies.

The use of tests to support a design automatically evaluates the materials used in the construction of a package. In addition, the use of a test will better determine the effect of gaps and the contact resistance between the package components than will the approximate (but conservative) methods discussed in Sect. 7.8.1.

The steady-state thermal tests used to simulate normal conditions of transport are, in concept, relatively simple. The tests should be performed on full-scale packages. Environmental chambers in which temperatures less than –40°C (–40°F) and greater than +38°C (100°F) can be created are available for testing. The radioactive decay energy of the contents of a package can be simulated using electric heaters.

The insolation incident (see Table 7.1) on the package surface in an ambient environment of +38°C (100°F) can be simulated using radiant heat sources. If the radiation heat source is other

than sunlight, differences between the photon energy spectrum of the source and solar radiation should be taken into account to ensure that the radiation energy flux absorbed by the package surface is the same as for sunlight. This incident radiant heat flux from the source will need to be adjusted to compensate for the difference in absorptivity resulting from the differences in the photon-energy spectrum.

Thus, the ratio of the required radiation heat flux, ϕ_{rad}, incident on the package surfaces to the solar heat flux, ϕ_{solar}, is

$$\frac{\phi_{rad}}{\phi_{solar}} = \frac{1}{\alpha}, \tag{7.9}$$

where α is the absorptivity of the package surface to the incident radiation spectrum.

The thermal radiation to simulate the insolation can be applied as either a constant value or as periodic heat flux as discussed in Sect. 7.5.2. The tests should be operated for a sufficient time for the spatial temperature distribution in the full-scale package to reach steady state for a constant heat flux or periodic steady state for a periodic heat flux.

Some simple analysis should be performed prior to the test. This analysis should assist in determining the placement location and the operating ranges of the instrumentation to be used in monitoring the performances of both the test and the package. Substantial differences between the test measurements and the analytical predictions should trigger a reexamination of the analysis and/or test unless the differences can be explained. A new analysis and/or test designed to validate the explanations of the substantial differences observed in the earlier analysis/test should be performed. Desirable measurements that should be taken during the steady-state tests include the temperatures of the environment, package surface, and several points within the package interior.

It should be recognized that heat-transfer tests carried out on reduced-scale models of the actual package will not provide experimental evidence of temperatures expected in the full-size package under specific environmental conditions.

7.9 HYPOTHETICAL ACCIDENT

The evaluation of the performance of a package in a hypothetical thermal accident is frequently not a part of the design process. As discussed in Sect. 7.8, the normal conditions of transport can be more limiting than the hypothetical accident conditions. However, the hypothetical fire accident is a case that must be evaluated to demonstrate that the package performs its functions during and after the hypothetical fire.

7.9.1 Thermal Evaluation by Analysis

The use of analysis to determine the response of a packaging to a thermal accident will allow the determination of the design margins for the thermal accident criteria selected. These transient calculations for generally nonlinear systems require the use of numerical methods. Because the package will have experienced other accident phenomena, including a 9-m (30-ft) drop onto an essentially unyielding surface followed by a puncture prior to the thermal accident, damage to the

packaging prior to the thermal accident must be considered in the analysis of the response of the packaging to the thermal accident. For example, large packages, such as for spent fuel shipments, have the impact limiters mounted external to the packaging. Because the impact-limiter materials are usually good thermal insulators, a conservative assumption is that these externally mounted limiters are separated from the package before the thermal accident. For drum-type packages where the impact limiter (which may also functionally serve as a thermal insulator) is located within the drum, the compression of the impact limiter from the dynamic loading experienced in the accident sequence prior to the thermal accident should be estimated with the thermal conductivity of the limiter adjusted to that of its crushed state.

The heat loads on a packaging during a thermal accident include the content decay heat in addition to the thermal radiation and convection from the external accident heat source. Following the accident, the packaging will radiate and convect the heat absorbed from the accident as well as from the contents to the environment.

For a given environmental temperature and a particular package surface temperature, the heat transferred by convection in a fire will be greater than that by natural convection in a quiescent environment. A comparison of the heat transfer coefficients as a function of surface temperature for a 1-m-diam horizontal cylinder in a 800°C (1475°F) air environment is given in Fig. 7.3 for natural convection and with forced convection caused by air moving at velocities of 5, 10 and 15 m/s. Note that typical velocities associated with a hydrocarbon fire range between 6 to 12 m/s (20 to 40 ft/s).[83,84]

The heat flux on the surface of a 1-m-diam, horizontally oriented packaging for a regulatory hypothetical accident is given in Fig. 7.2 as a function of the packaging surface temperature for several flow velocities. As noted, for a regulatory furnace test, radiation constitutes about 90% of the total heat flux, with the remainder from natural convection. For a regulatory fire test, radiation constitutes between 75 and 90% of the total heat flux, while the remainder is from forced convection of the heated air at various velocities normal to the axis of the horizontal cylinder.

The natural convection coefficient from a 1-m-diam horizontal cylinder during postaccident cooling is given in Fig. 7.4 as a function of surface temperature for an environment temperature of 38°C (100°F). The cooldown heat flux is also given as a function of temperature for a package emissivity of 0.8 in Fig. 7.4. When an analysis indicates that thermal conditions exist such that chemical decomposition or material changes can occur, a thermal evaluation of the packaging should be performed by testing.

7.9.2 Thermal Evaluation by Testing

If transient thermal tests to simulate hypothetical accident conditions on a transportation package are to be performed, they must be performed on a package that has been subjected to a sequence

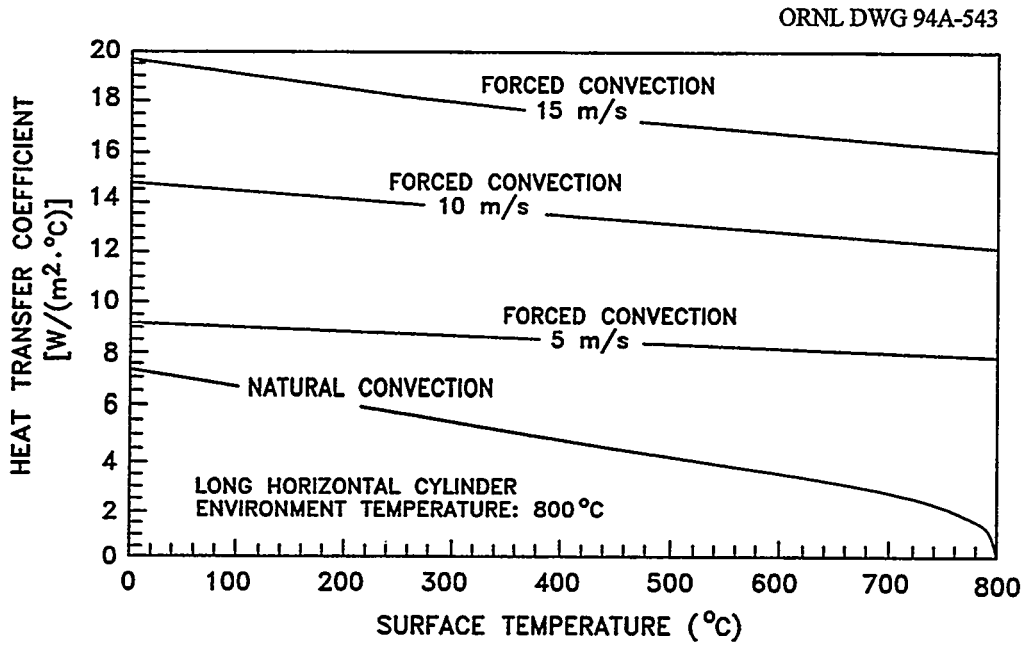

Fig. 7.3. Comparison of heat transfer coefficients for natural convection over a 1-m-diam horizontal cylinder with that of cross flow at several velocities over a cylinder as a function of surface temperature for air temperature of 800°C. The natural convection occurs during a furnace test (or following a fire test), while the forced convection occurs during a fire test.

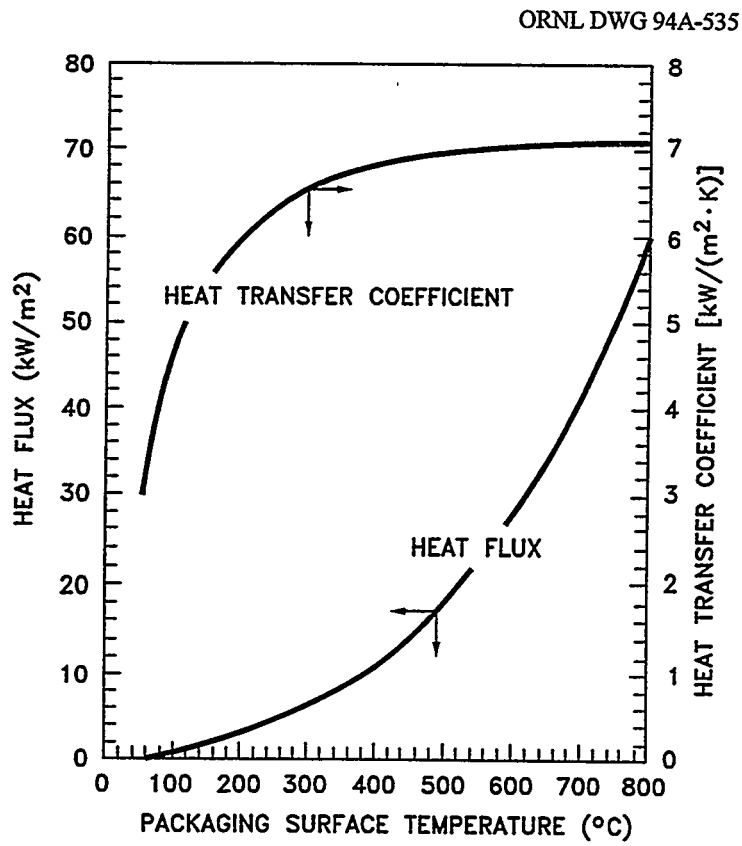

Fig. 7.4. Natural convection coefficient and heat flux as a function of surface temperature for a 1-m-diam horizontal cylinder to air at 38°C (100°F), with a package emissivity of 0.8.

of physical tests designed to produce the maximum damage to the package. Such thermal tests are usually performed in a furnace or pool fire facility. An excellent set of guidelines for thermal tests to simulate hypothetical accident conditions have been developed by J. H. VanSant, et al.[75] Testing is the recommended approach if the packaging includes materials that

- change phase at temperatures between –40° and 800°C;
- combust at temperatures between –40° and 800°C; or
- decompose at temperatures between –40° and 800°C.

Also, testing is recommended for small packagings [e.g., 208-L (55-gal) drums] because the cost of producing a sacrificial packaging and the cost of testing are small compared with the cost of analysis.

Two situations must be considered when planning a test program. If the contents of the package constitute a significant heat source, the first involves (1) the simulation of that heat source or (2) the determination of the effects of the source on the thermal response of the packaging if a simulated heat source is not present. The use of a preheated package to simulate the initial temperature profile caused by decay heat is not an appropriate substitution for the decay heat released during the test.

A second situation involves combustible materials if any are used in the packaging design. If these types of materials are designed to be in the package under normal operating conditions, they must be present in the thermal test; they must be allowed to burn until self-extinguished, which may add significant amounts of heat to the packaging and extend the duration of the fire beyond the 30-min test period.

7.9.2.1 Furnace Testing

High-temperature furnaces that produce temperatures greater than 800°C (1475°F) can be used in the qualification of packaging and radioactive contents in support of the transportation of radioactive materials. The furnace heat sources may consist of electric resistance heaters or the combustion of natural gases or other hydrocarbon fluids. The furnaces should be able to operate at a radiant source temperature of greater than 800°C for a time greater than 30 min to produce a uniform absorbed *radiant* energy flux of greater than 54.8 kW/m^2 (17,400 Btu/h ft^2) on a package having an initial surface temperature of 38°C (100°F). The furnace must also have dimensions sufficiently greater than the package dimensions to allow natural circulation of gases to take place within the furnace.

The net radiation flux exchanged between two bodies is

$$\phi_{rad} = \sigma \Im_{12}(T_2^4 - T_1^4) \ , \tag{7.10}$$

where σ is the Stephen Boltzman constant, and $\Im$ is the overall energy interchange factor between body 1 and body 2, given by the equation

$$\mathfrak{I}_{12} = \frac{1}{\frac{1}{\varepsilon_1} - 1 + \frac{1}{F_{12}} + \frac{A_1}{A_2}\left(\frac{1}{\varepsilon_2} - 1\right)} \tag{7.11}$$

for the case of an optically transparent fluid between the surfaces. F_{12} is the black surface overall energy interchange factor.

For the case of a furnace test in which the emissivity of the furnace or package is different from the regulatory prescription, the characteristics of a furnace test to meet regulatory radiant heat flux can be determined by

$$T_2^4 = \frac{\phi(T_1)}{\sigma}\left[\frac{\varepsilon_2 + \left(\frac{A_1}{A_2}\right)(\varepsilon_1 - \varepsilon_1\varepsilon_2)}{\varepsilon_1\varepsilon_2}\right] + T_1^4 , \tag{7.12}$$

where $\phi(T_1)$ is the regulatory radiant heat flux with the package temperature T_1 (subscript 1 refers to the package characteristics, and subscript 2 refers to the furnace characteristics). The furnace radiant temperature T_2 must be greater than 800°C (1475°F).

The use of a furnace test to generate a greater-than-regulatory heat flux on a package surface may have the disadvantage that, during a 30-min test time, certain package components may exceed the temperature that they would experience under an 800°C (1475°F) regulatory fire. If these components degrade at a temperature greater than 800°C, but less than the higher temperature used in a test to compensate for different emissivites of the furnace, an erroneous assumption may be made that the packaging is unsuitable to transportation of the contents. This assumption may cause an unnecessary redesign of the package.

For the furnace temperature of 800°C, the surface characteristics of the furnace and package necessary for the radiant heat flux of the furnace test to equal or exceed that of a regulatory pool fire test (at 800°C) require that,[76]

$$\frac{1}{\varepsilon_1} + \left(\frac{A_1}{A_2}\right)\left(\frac{1}{\varepsilon_2} - 1\right) \leq \left(\frac{1}{0.8} + \frac{1}{9}\right) = 1.361 . \tag{7.13}$$

For the case in which the package surface emissivity is greater than 0.8, but the radiation source emissivity is less than 0.9, Eq. (7.13) reduces to

$$\frac{A_2}{A_1} > 9 \cdot \left(\frac{1 - \varepsilon_2}{\varepsilon_2}\right), \tag{7.14}$$

which indicates that a large ratio of the furnace-wall area to the package-surface area will tend to compensate for a less-than-regulatory-specified furnace-wall emissivity. The bound of minimum furnace wall to package area ratio for a given furnace-wall emissivity (<0.9) is given in Fig. 7.5.

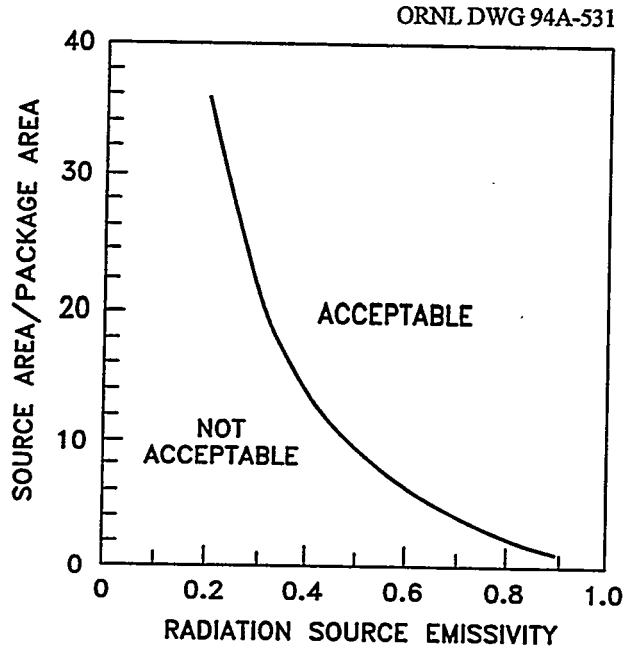

Fig. 7.5. Minimum surface area ratio of furnace wall to package for a furnace-wall emissivity less than 0.9 that will satisfy the regulatory radiant heat flux on a package with a furnace-wall temperature of 800°C (1475°F) and a package surface emissivity of 0.8.

In the event that the inequality given above is not satisfied, the furnace temperature, T_2, can be increased until the regulatory radiation heat flux is achieved.

For the case in which the furnace is fired by the combustion of natural gas or other hydrocarbon fluids within the test chamber, the products of combustion such as H_2O and CO_2 are not optically transparent; the radiation heat flux between the furnace walls and the gas to the package is given by[75,77]

$$\phi_{rad} = \frac{\sigma\varepsilon_p \left[E_w + \varepsilon_g(1/\varepsilon_w - 1)\right]\left(T_g^4 - T_p^4\right) - \left(T_g^4 - T_w^4\right)}{E_w\left[1 + (1 - \varepsilon_g)(1 - \varepsilon_w)E_p\right]}, \tag{7.15}$$

where

σ = Stefan-Boltzmann radiation constant (W/m^2-K^4),
E_w = $[\varepsilon_g (1 - \varepsilon_w) + \varepsilon_w] / [(1 - \varepsilon_g)\, \varepsilon_w]$,
ε_g = gas thermal emissivity coefficient,
ε_w = oven wall thermal emissivity coefficient,
T_g = gas temperature (K),
T_p = package temperature (K),
T_w = oven wall temperature (K),
E_p = $[\varepsilon_g (1 - \varepsilon_p) + \varepsilon_p] / [\varepsilon_g(1 - \varepsilon_p) + \varepsilon_w]$, and
ε_p = package surface thermal emissivity coefficient.

This equation is applicable to a furnace having uniform wall and gas temperatures, spectrally gray gases, and an area ratio and package-to-oven view factor equal to unity (equivalent to infinite parallel plates). For many ovens, radiation heat transfer to a test package cannot be adequately predicted using simple equations, such as Eqs. (7.10) and (7.11) (see ref. 78). The basic assumption implicit in these equations is that the surface of each body is isothermal. The basic assumption in Eq. (7.11) is that the surface heat flux is uniform over each of A_1 and A_2. These conditions are satisfied only for the concentric case, or approximately satisfied for eccentric cases in which the emissivity of each of the surfaces is so high (>0.9) that the reflections from the surfaces are relatively unimportant. If these conditions are not satisfied for the actual furnace facility, the analysis can be performed by dividing the system into smaller elements and solving for the radiant energy transferred by using flux algebra or a Monte Carlo analysis.

The total heat flux on the surface of a package consists not only of the radiant heat flux, but also of a convective heat flux driven by either the natural circulation of the gas within the furnace cavity or by convection resulting from combustion within the cavity. The current regulatory hypothetical accident requires the convective coefficient to be that value which exists if the package were exposed to the specified force.[2]

A furnace facility to be used for the simulation of a hypothetical accident condition on a package should be preheated to a temperature greater than 800°C (1475°F), have sufficient input power, and/or thermal capacity to ensure that the radiation source temperature does not drop below 800°C when the package is placed in the furnace. In addition, the initial air temperature in the furnace should remain near that of the furnace wall during the entire test. The package support should consist of a high-thermal impedance system to minimize the heat conduction between the package and the furnace through the support. The support should be designed and located to minimize the perturbance of the radiation flux on the package. The oxygen level in the furnace interior should not be allowed to drop below the concentration necessary to ensure that combustion of package components, if any, will proceed at rates unconstrained by the loss of molecular oxygen (producing carbon monoxide and dioxide) in a closed environment. If fresh air or oxygen must be furnished to the furnace interior, the makeup air/oxygen should be preheated to the desired furnace interior temperature.

Testing a packaging design in a furnace is not recommended if there are combustible materials or materials that thermally decompose into combustible materials as part of the packaging. The reason for this is that, usually, a furnace will be closed to prevent ingress of relatively cold ambient air and that will also limit the amount of oxygen that is available for combustion. Should substantial flames emanate from a furnace when the door is opened to remove a packaging at the end of a 30-min test, the flames will indicate that incomplete combustion of package components took place. Also, the presence of flames around penetrations for power or instrumentation leads indicates oxygen deprivation within the furnace.

7.9.2.2 Pool-Fire Testing for Packages

The use of pool fires to simulate a hypothetical accident condition on a package represents, perhaps, the more logical method of testing a transportation package. Frequently, the fires produced in a transportation accident are the result of the ignition of spilled fuel from the transportation conveyance. In principle, the use of a pool fire produces the appropriate amount of convective heat transfer from the flame and fluid motion to the package.

The characteristics of a regulatory fire test include minimum-incident emitted power ϕ, where ϕ is given by

$$\phi = \varepsilon\sigma T^4 \tag{7.16}$$

and is greater than 67.6×10^3 W/m^2 with a temperature greater than 800°C (1475°F). The package must be mounted 1 m above the surface of the fuel pool, which horizontally extends more than 1 m, but less than 3 m, from the external surface of the package.[79] The limits on the pool size relative to the package size are to ensure that oxygen is available to the fire center (maximum size) and that the fire is thick enough (minimum size) such that the emissivity produced will result in the fire achieving at least the minimum emitted power specified earlier.

The package support should consist of a high-thermal impedance system to minimize the temperature gradients in the package by heat conduction to the ground. Thus, the supports should have a length of greater than 1 m and have a small cross-sectional area for heat transfer. The supports should be designed and located to minimize the perturbance of the radiation flux, and the velocity of the flames and fluid on the package.

The fuels used in a pool-fire test range from natural gas to kerosene, diesel, or aviation jet fuel. The liquid hydrocarbon fuels, such as kerosene, diesel fuel, or aviation fuel, can be floated on a pool of water that will protect the bottom of the pool. In addition, the combustion products of these fuels have a sufficient number of carbon particulates to produce flames with a sufficiently high opacity such that the flames are optically thick.

The use of gaseous fuels to simulate the hypothetical thermal accident condition on a package is not encouraged. The use of natural gas will require a large number of burners to blanket the package because the flames have few entrained carbon particles, resulting in optically thin radiation environments. To compensate for the optically thin flames, the flame temperature of the natural gas may need to be increased by adjusting the fuel:air ratio to achieve the required minimum emitted power.[75]

A general recommendation for the fuel is that the fuel should comprise a distillate of petroleum with a distillation end point of 330°C (625°F) maximum and an open cup flash point of 46°C (115°F) minimum and with a gross heating value of between 46 and 49 MJ/kg (20 and 22 $\times$ 10^3 Btu/lb). This covers most hydrocarbons derived from petroleum with a density of less than 820 kg/m^3 (51.2 lb/ft^3) (e.g., kerosene and JP4-type fuels). A small amount of more volatile fuel may be used to ignite the pool as this will have an insignificant effect on the total heat input.[80]

The burn time can be controlled by controlling the amount of fuel placed in the pool. A typical recession rate of a fully developed JP-4 fuel pool fire is 6.4 mm/min.[81] Thus, a pool fuel depth of 0.20 m will produce a test time of about 30 min before the fuel is totally consumed, extinguishing the fire.

A major problem with conducting a pool fire test is the effect of wind velocity on the test. The wind will cause the fire plume to tilt (relative to the axis of the package), causing a variation of the flame thickness around the package and a nonuniform flame and fluid velocity around the periphery of the package at the location of any horizontal plane through the package. This results in a nonuniform heat flux on the package periphery, with the possibility that, instead of being surrounded by an engulfing fire, portions of the package surface may become visible through the

flames. A relatively modest breeze with a velocity of the order of 1 m/s can cause a tilt in the flame column by as much as 20°, as shown in Table 7.3 (ref. 82). In addition to the wind effects, the flame pinch reduces width of the flames to about 80% of the width of the base of the flames at a vertical height of about 2 m above the surface of the fuel. This phenomenon is caused by air entrainment induced by the buoyant fire plume. Empirical data for the tilt of the flame plume at a height of 3.0 to 4.6-m above the fuel surface as a function of the wind speed are presented in Table 7.3.

The relationship between wind speed and the tilt angle of flame plume is approximately:

$$\tan \theta = \frac{v_w}{0.25\ v_f}, \tag{7.17}$$

where

θ = the angle that the flame plume is tilted from the vertical in degrees,
v_w = the horizontal speed of the wind, and
v_f = the vertical speed of the flames in the same units as v_w.

The typical vertical velocities from forced flow caused by combustion in a hydrocarbon-based fire test ranges between 5 to 15 m/s (16 to 50 ft/s).[83–85]

Employing the above information in a practical example of a pool fire test, one concern might involve the determination of how quiescent the air needs to be during the test. Assume the package to be tested has a diameter of 1 m (3.3 ft) and a length of 1 m and the pool extends 1 m beyond the edge of the package in all directions. This results in a pool that is 3 m by 3 m square. In order to account for the pinching effect of the flame at the level at which the package is suspended above the pool, the dimension of the flame should be decreased by about 20% (to 2.4 m). This means that in order to prevent the wind from bending the flame too far and potentially exposing the edge of the package, its velocity should not exceed 1.1 m/s (2.5 mph). The geometry of this condition is shown in Fig. 7.6.

If the width of the fire is increased to 7 m (3 m between the package and the edge of the fuel), the maximum wind speed can be 2.67 m/s (6 mph) without causing any of the package to be outside the flame plume. However, in both examples, although the package remains within the flame plume, the emissivity of the fire is not uniform over the surface of the package because of the nonuniform flow thickness.

Table 7.3. Empirical flame-plume-tile data
(ref. 82)

Wind speed		Tilt of flame plume (degrees)
m/s	mph	
0.98	2.2	25
1.20	2.7	27, 21, 20
1.3	2.9	27

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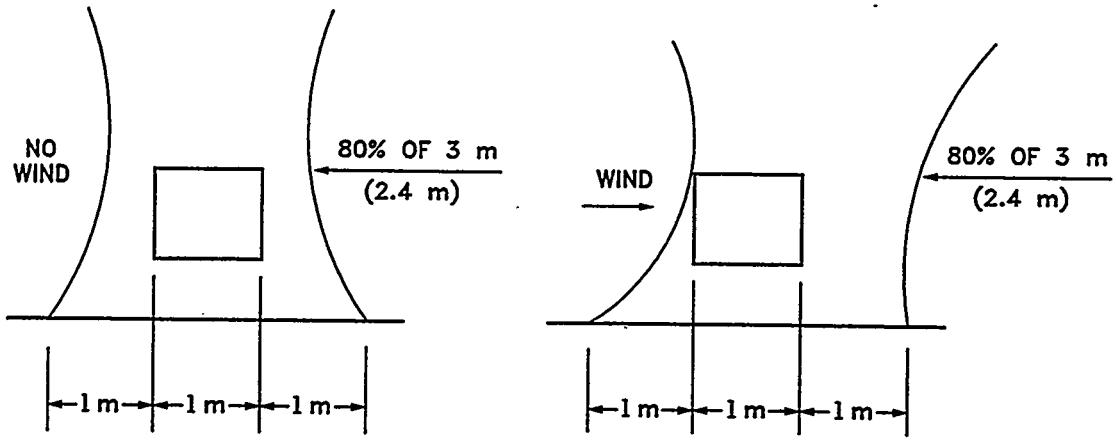

Fig. 7.6. Example of how the wind affects the flame thickness around a package in a fire test.

These wind speeds are very small, suggesting that a test program should focus upon the wind speed and be prepared to delay testing until the wind is sufficiently calm. Retesting may be necessary if the wind conditions change during the test.

To mitigate the effects of wind velocity on a pool-fire test, several options are available. These options include:

- Ensuring that the package is fully engulfed (remains invisible through the flames) for an integrated time of 30 min.
- Shielding the pool fire facility from the wind.[86]
- Allowing the test to be satisfactory if the average incident emitted power and the average fire temperature around the periphery of the package are greater than 67.6 kW/m^2 and 800°C (1475°F), respectively, for the 30-min test time; or for thin-wall, drum-type packages the surface temperature is maintained at 800°C or greater for 30 min.

These are issues for the above mitigation measures. The facility must have the ability to extend the fire time until the integrated time of 30 min is achieved for the fully engulfed fire. The effect of the screen may act as a heat source or sink by reflecting the radiant flame energy to the package or by radiating the absorbed flame energy to the facility external environment. The use of the spatial average incident emitted power and temperature will require the need to make measurements of the emitted power and temperature. These measurements may interfere with the incident emitted heat flux and the flame/package temperature. Thus, the method used to mitigate the effects of wind and/or other environmental effects on the package may need to be justified.

The discussion of the allowable wind speed also serves to illustrate the point that in planning and performing the tests to satisfy the regulatory requirements, often only one parameter of the many

that characterize a fire test can be selected. Of the parameters that characterize the fire test (flame temperature, flame emissivity, environment temperature, wind speed, and wind direction), only one can be specified, and the others must be accepted at whatever value results. Some of the parameters such as flame temperature and flame emissivity are specified by the choice of fuel material, and they are controllable in that changing the fuel will change these parameters; however, only a few combinations of these parameters are possible. Figure 7.7 presents the heat flux reaching a package during a fire for a range of flame temperatures. For a 1-m-diam cylindrical package with a surface emissivity of 0.8, a flame emissivity of 0.9, and a flame velocity of 5 m/s (11.2 mph), the initial surface heat flux will almost double with a 20% increase in the absolute flame temperature.

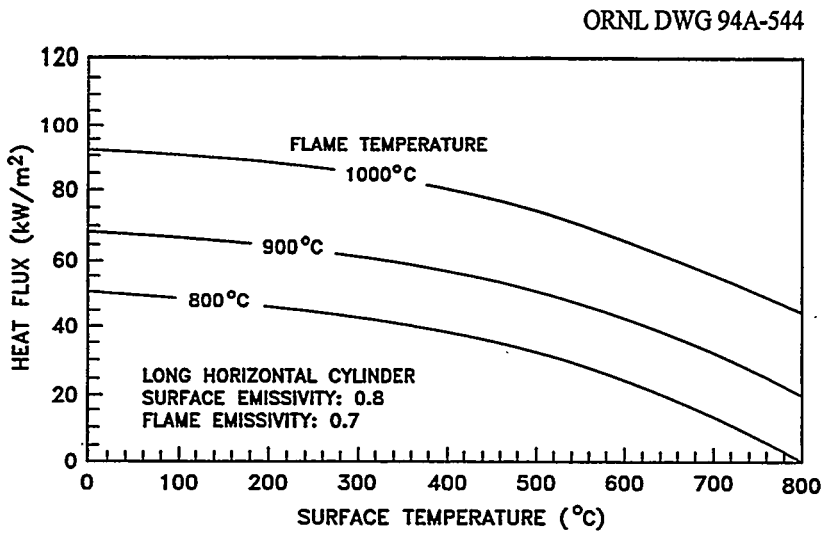

Fig. 7.7. Heat flux as a function of surface temperature on a 1-m-diam horizontal cylinder for several flame temperatures.

7.9.2.3 Comparison of Furnace Test with Pool-Fire Test

Several differences are noted between a furnace test and a pool fire test even with similar source and package emissivities. These differences primarily result from the type of convective heat flux on the surface of the package.

Convection coefficients for a fire test range between 10 and 92 W/m^2-K (1.7 and 16.2 Btu/h · ft^2 · °F) (refs. 83, 84, 87). These values were calculated using forced convection coefficient relationships for different fluids that may be considered to be relevant to a hydrocarbon-fuel pool fire. The importance of the assumption of the type of fluid is given by L. H. Russell and J. A. Canfield, where for a velocity of 4.6 m/s (10.3 mph), the calculated convection coefficient ranged from 15.3 W/m^2-K for carbon monoxide, to 15.9 W/m^2-K for dry air, to 92 W/m^2-K for isooctane, with an average convection coefficient of 56.1 W/m^2-K inferred from the experimental results.[88] The convection coefficient for a furnace test is typically about half that of a pool-fire test as shown in Fig. 7.3. Not only is the convection heat transfer different for a pool fire than for a furnace test, but the direction of the circulation of the fluid about the package differs. The flow over a package surface is upward in a pool fire, while the flow in a furnace test is downward because the package surface temperature is less than the furnace-wall temperature.

The simulation of the forced convection of a pool fire in a furnace is difficult to accomplish. The use of a natural gas or other hydrocarbon-fluid-fired furnace to heat the package may simulate the buoyant effect of the flame in a pool-type fire facility both with respect to the flame velocity as well as the direction of the gas velocity upward over the package. To simulate the effects of the forced convection of a pool-fire facility using an externally heated or a resistance heated furnace will require injecting high-temperature [800°C (1475°F)] air with a velocity of 5 to 12 m/s (11 to 25 mph) over the surface of the package. Forced convection can also be simulated by natural convection by increasing the test temperature. Thus, the use of elevated (>800°C) furnace temperatures are suggested for the simulation of the hypothetical accident condition of a pool fire. Note that the use of this higher temperature may, depending on the temperature, compensate for the forced convection heat transfer to the package.

Another difference between a pool-fire test from a furnace test is the composition of working fluid. For a pool-fire test, the working fluid is air, evaporated fuel, and combustion products that remain relatively constant during the test. For a furnace test, the composition of the working fluid may change during a test if combustion of any of the packaging occurs. This effect will increase the quantity of H_2O, CO_2, and other gases that will participate in the radiation and convection heat-transfer processes. In addition, the oxygen concentration in the furnace cavity will decrease over the test time which may impede the combustion process.

7.9.2.4 Content Simulation

Because of safety considerations, packages that are subjected to the hypothetical accident tests never use the actual contents for which the package is to be used. Under these conditions, the contents must be simulated. To the degree possible, the characteristics of the simulated contents should include the mass (for the dynamic load tests, especially the 9-m-drop test), the average specific heat, heat generation rate, and pressure generation characteristics of the actual contents.[75]

Although typically a secondary effect, the duplication of the mass and specific heat are necessary to simulate the response of the contents to the transient portion of the hypothetical accident test. The duplication of the pressure generation characteristic is desirable to simulate the increase in pressure load in the containment vessel caused by the response of the contents or back-fill gas in the hypothetical accident condition. This can be simulated with an initial fill of gas that will be sufficient to produce the maximum estimated pressure in the containment (caused by the fill gas expansion or a phase change and composition or out-gassing of the contents) during the hypothetical accident test.[75]

The simulation of the heat-generation rate of the contents is, in principle, not difficult. A simple electrical heater can be used to simulate the content heat-generation rate. However, the heater should be placed in the containment vessel before the package is subjected to the sequence of hypothetical accident tests with the intent of energizing the heater and letting it come to equilibrium before the package is subjected to the thermal test. One potential difficulty with such tests is that to energize the heater, it will have to be equipped with electrical leads that pass through the containment vessel, and most packages are not designed for this. Another potential difficulty is that the heater may fail as a result of being subjected to the 9-m drop and 1-m punch test that must precede the thermal test. A third concern is that the electrical leads may short circuit during the tests if care is not taken in the design of the simulation heating system.[75] For the case in which the damage to a package from the hypothetical accident conditions is small, a fresh package can be fully instrumented with temperature-measuring sensors, outfitted with a heater to

simulate the decay heat, and subjected to a fire or furnace test without being subjected to the remainder of the hypothetical accident conditions. These test results will furnish the designer a clue to the size of the thermal design margin.

7.9.3 Scale Testing and Analysis

The use of scale-model tests to qualify a package for the shipment of radioactive materials is a temptation if large-scale test facilities are unavailable to an applicant. However, testing of scale models in a thermal environment is discouraged since the model will not respond in the same way that a prototype will to a given thermal environment (e.g., a fire test). However, it would be appropriate to subject a scale model to a thermal test in order to benchmark computer software that is to be used in the package design and analysis.

7.10 SUMMARY OF THERMAL DESIGN ISSUE RECOMMENDATIONS

In the previous sections, several recommendations were made to help mitigate the effects of normal transport conditions and hypothetical thermal accident condition on the shielding, criticality control, and containment systems of a package. These recommendations include:

- Passive thermal protection systems are encouraged.*
- The use of materials that will exothermally decompose within the maximum temperature range of the packaging is discouraged.
- The use of materials that produce pneumoconiosis is discouraged.
- The use of materials that have a physical-state change within the maximum temperature range of the packaging is discouraged.
- The use of a coolant that can be lost from the cavity of a package in an accident is generally not allowed.
- The use of active environmental controls can be operationally efficient, but the package must meet the regulations if the active elements fail.

Note that the recommendations given above are not mandatory. However, there should be a reason for a variance from these recommendations. In addition, the effect of such a variance on the shielding, criticality control, and containment for normal conditions of transport or hypothetical thermal accident conditions should be thoroughly evaluated.

*Although a designer may choose to use an active system (e.g., forced circulation of a coolant) for convenience under normal operating conditions in order to reduce the normal operating temperatures of a package, it must be recognized that thermal systems important to safety under both normal and accident conditions of transport must be passive if regulators are to assume that they can operate properly after an accident.

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8. Containment

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8.1 INTRODUCTION

This chapter describes the principal aspects of demonstrating containment of the proposed contents of a package. It addresses the information required by Chapter 4 of the *Safety Analysis Report for Packaging* (SARP) as recommended for Type B packages by the U.S. Nuclear Regulatory Commission's (NRC) *Regulatory Guide 7.9, Standard Format and Content of Part 71 Applications for Approval of Packaging of Type B, Large Quantity, and Fissile Radioactive Material*, January 1980 (ref. 1).

The regulatory provisions for containment vary somewhat by package type, but usually no loss or dispersal of contents is permitted in normal conditions of transport. Under accident conditions, the requirements vary much more widely. For some package types, loss of contents is permitted, while for others, such as the Type B package, only very small quantities of material may escape from the package, over time. Within the regulations, limits on the releasable materials are specified as a function of the contents, and the actual quantity of material that may be released is dependent upon its isotopic constituents.[2,3] These requirements are established by both national and international regulations and are included in American National Standards Institute, Inc. (ANSI), N 14.5, the standard entitled *Leakage Tests on Packages for Shipment of Radioactive Material*,[4] which is referred to as the "Standard" throughout the rest of this chapter. This chapter was written to assist the designer in applying the requirements and formula of the 1987 edition of ANSI N 14.5. The Standard is being revised to make it easier to read and use, and will be issued as the 1997 edition. To the extent practical, this chapter incorporates the 1997 Standard, even though the Standard is in draft form. Where reference is made to the 1987 version of the Standard, the reference is noted.

The reader should also be aware that the A_1 and A_2 values of some radionuclides were revised effective April 1, 1996, to bring those values in line with those of the IAEA. Consequently, care should be exercised to ensure that correct values are used in leak rate calculations. Recalculation of leak rates for previously approved packages will likely be required at the time of application for an amendment to the package certificate. Recalculation of leak rate is also required for any amendment to certify a previously approved package to the NRC's 1985 requirements.

Containment of the contents of a package during both normal conditions of transport and under accident conditions is important to the health and safety of the public and of the package operators. To assist the designer in developing a package that provides adequate containment for the proposed contents, the regulators specify features that must be incorporated and also some that

cannot be incorporated into package design. The required conditions or features include positive-fastening devices for closures (such as torqued bolts), protection of fasteners from inadvertent operation, enclosures at penetrations to retain leakage from valves, and use of adsorbents for liquids. Pressure-relief valves are excluded from the enclosure requirement.[5,6] The regulations prohibit continuous venting devices, filters, and mechanical cooling of the package.[7,8] If the proposed contents of a package require double containment, then each containment boundary must independently meet the allowable leak rate for normal and accident conditions.

The designer may elect to show by analysis that leakage that might occur under normal and defined accident conditions is below the limits specified in the regulations, or that the package is *leak tight*. The Standard defines leak tight as having a leak rate equal to, or less than, 10^{-7} std cm^3/s (air). The Standard also establishes the methodologies for determining and measuring the leak rate. In applying both the regulations and the Standard, the designer should always consult the current editions of both because the requirements are subject to change, even if infrequently. The regulations should also be reviewed for exemptions that may apply to the package type or the proposed contents.

From the perspective of carrying out the required analysis to demonstrate compliance with the regulatory performance standards, designing to leak-tight conditions (as defined in the Standard) reduces the amount of information required in Chapter 4 of the SARP because no formal calculation of leak rate based on package contents or conditions is needed. Regardless of the strategy used to demonstrate package containment performance, Chapters 7 and 8 of the SARP must describe the methods of test and measurement that demonstrate that the performance requirement is met.

There are three principal elements to the determination of the package containment performance requirement:

- the quantity and isotopic makeup of the material in the package that could be available for release should a leak occur between the cavity of the package and the environment;
- the nature, pressure, and temperature of the medium that surrounds the contents within the cavity (typically air or an inert gas that does not contribute to an oxidation or corrosion process); and
- the components and configuration of the containment boundary and, in particular, the seals used in the boundary.

The term *medium*, as used herein, includes any gas or fluid which may not be itself radioactive, but which could carry radioactive material through a leak in the containment boundary.

The designer will apply knowledge of the package and contents to determine the amount of material available for release in normal transport and under the defined hypothetical accident conditions, will use the regulations to determine the allowable release (if any), and will apply the Standard to determine how the containment requirement is demonstrated. This chapter describes some of the calculations required by the Standard; similar calculations should be included in the SARP to show compliance to the containment requirement.

An international standard that is comparable to ANSI N 14.5 is being developed by the International Organization for Standardization (ISO). The ISO standard is entitled *Leakage Testing*

on Packages for the Safe Transport of Radioactive Materials and has some minor differences from the ANSI N 14.5 Standard, which could be important in specific applications to packages that may be used internationally. The ISO standard also contains examples of the application of analysis to packages and uses SI units throughout.

8.2 GENERAL CONTAINMENT CONSIDERATIONS

This section discusses some of the generic issues related to the containment boundary and configuration that can affect the performance of the package during transport as well as the use of the Standard. The demonstration that the package cavity can maintain an adequate seal under normal and hypothetical conditions of transport is usually presented in Chapter 2 (which concerns structural aspects) and Chapter 3 (which concerns thermal aspects) of the SARP.

8.2.1 Package Containment

The package containment boundary consists primarily of the cavity in which the contents are placed, the closure lid, the penetration closures, and the seals that are placed between surfaces to prevent leakage. ("Seal" or "seals" is intended to include the full range of O-rings, seals, and gaskets; metal; or other material that could be used to achieve containment.) The cavity shell, the cavity, and penetration closures and fittings are typically fabricated from one of several types of steel. The remaining portion of the containment boundary consists of the seals between a closure and its mating piece. Because the metal surface is largely impervious to the typical gases, slurries, and particles contained in the cavity, the performance of the seals in the boundary is the principal design concern. Infrequently, the designer may have to consider the use of a pressure-relief valve for the cavity or other closed space integral to the package (e.g., a neutron-shield tank) or the effects of tritium, which has been shown to migrate into steel over long periods.

The gases and particulate material that could potentially escape from a package could be contaminated with either (1) a mixture of isotopes that are available from the contents and, to a lesser degree, from residual radioactive material within the cavity or (2) a material that has been deposited on the surface of the contents. The constituents available for release from the contents are typically determined from analysis of the contents, using computer codes such as ORIGEN,[9] from knowledge of the process from which the contents are obtained or from direct sampling. Once the potentially releasable contents are known, a weighted average of the constituents is used in calculating the allowable leak rate for the package. The methodology of these calculations is taken from the applicable regulations, primarily 10 CFR Part 71, Appendix A, and is applied to a particular example in Sect. 8.3, which follows.

The pressure that is developed within the cavity either during normal transport or under the hypothetical transport accident conditions is the principal driving force for the release of material in the unlikely event that the containment boundary of the package leaks. Most packages are designed to be transported with dry contents sealed in a dry cavity. A major incentive for keeping liquid out of the cavity is that significantly lower pressures will exist in the cavity under both normal or accident conditions, reducing the likelihood of a release. For a cavity that contains water or other fluid, a much higher pressure typically is produced from the hypothetical fire condition. In addition, there may be some concern that water or liquid organics might undergo significant radiolytic decomposition, forming hydrogen gas. The formation of combustible gas is discussed in Sect. 8.7.2.

The cover gas medium surrounding the contents is usually air, but the designer may specify other gases, depending on the design requirements. If inerting is required because of the potential for oxidation of the contents or to otherwise limit free oxygen, helium or nitrogen is frequently used. Helium is a better heat conductor than air, and it is one of the few gases for which leaks in the normal range of interest for package containment can be detected and measured. Nitrogen can also be used to prevent oxidation of the contents and reduce the available free oxygen. It is somewhat less expensive than helium and is usually available at most nuclear facilities because it is often used as an instrument-line purge gas. The NRC has also suggested argon as a cover gas,[10] but it has not been routinely used. Argon is more expensive than helium and is less readily available. It is recommended that the designer specify more than one cover gas if inerting is required. (Note that only helium may be specified if credit is also taken for its presence in the package's thermal analysis.)

In addition to the primary opening for inserting and removing the contents of the package, the typical package cavity is penetrated by one or more openings intended for venting, filling, draining, and sampling the cover gas in the cavity. Usually, only two penetrations are used, one for sampling, venting, and filling, and one for draining. Each of these penetrations becomes part of the containment boundary and must be closed to establish the leak tightness required by the regulations. The penetrations may be either through the side wall of the package or through the package closure. Regardless of the location of the penetration, the design must ensure that the containment boundary performs as required under normal transport and hypothetical accident conditions.

In preparing the containment analysis for the package, the designer must identify the package containment boundary and its components. This identification includes a physical description of the key elements of the containment system, materials of construction, dimensions, methods used to seal the package, and operation of the penetration components. The physical description should be supplemented with a sketch showing the containment boundary, or reference should be made to the items as shown on design drawings included in the SARP. Also, the torque or bolts used to achieve positive closure must be specified. A bolt-torque table should be included in Chapter 7 of the SARP and referenced in the containment analysis. (The torque table provided in Chapter 7 would also contain the torque values for bolts not associated with the containment boundary. It is desirable that only one torque table be used in order to preclude the inadvertent use of different torque values in different tables.)

Leak testing is performed to verify the performance of the containment-boundary seals. The containment design should ensure that the seals can be adequately and efficiently tested. Other portions of the containment boundary, the cavity shell, closure and penetration structures, and the necessary welds are not routinely leak tested. These elements are qualified during fabrication and package-acceptance testing by any one of several methods, including nondestructive testing and inspection of the raw materials and the finished products. Unless an event occurs that raises doubt about the continued performance capability of the package during later use of the package, the integrity of the containment materials and welds is not typically reverified.

A significant portion of the package-handling time will be spent in the verification of containment each time the package is used and in the similar determination made during periodic (e.g., annual) inspections. The designer should consider how containment testing and necessary seal replacement will be performed in the operating environment. Operational and design aspects of the containment penetrations and seals are discussed in Sects. 8.2.4 and 8.6, which follows.

A confinement evaluation is also required for canisters used in dry storage systems. The canister has the same penetrations as a package; however, the penetrations are typically closed by field-installed welds. Although an allowable leak rate can be calculated for a canister, it is usually not worthwhile to do so, as "leak tightness" can be claimed for a welded closure. The integrity of the welded closures [lid(s) and penetrations] is demonstrated by a liquid penetrant inspection, and some welds may be helium leak tested. Satisfactory welds present a continuous metal containment boundary to preclude the possibility of loss of contents by leakage.

Noted, however, in passing that 10 CFR 72, which presents the regulatory requirements for storage of spent fuel, requires that the canister be backfilled with an inert gas so that there is no degradation of cladding over the storage period. Consequently, if an allowable leak rate is proposed by the designer, then the designer must show that (1) the allowable release rate is not exceeded, and (2) there is no in-leakage of air which could lead to a loss of fuel cladding integrity over the period of storage. Since the closures are welded, the designer is usually better served to demonstrate leak tightness by testing (limited helium leak test and liquid penetrant inspection of welds that can not be effectively leak tested) to demonstrate a continuous metal containment boundary.

Finally, even though the designer may demonstrate leak tightness for confinement, it is still necessary to calculate the concentration of radionuclides available for release in the canister due to accident conditions. The reason for this is that the designer must estimate the total dose at the site boundary (boundary of the owner-controlled area), assuming that the storage canister fails catastrophically. This failure is a hypothetical assumption made only to assess site-boundary conditions, consistent with how other activities at the site are evaluated. The analysis for the dry storage system, including the canister, will show that there are no design-basis accidents that could result in the failure of the canister or cause it to leak.

8.2.2 Normal Transport Conditions

The principal driving force for leaks under normal conditions of transport is the pressure increase resulting from heatup of the cavity medium. This medium, typically a gas, is heated by direct solar heating of the external surface of the package and the radioactivity that exists in the contents. The increase in pressure in the medium during transport is typically calculated using the ideal gas law and the maximum cavity or contents temperature and is generally not large. The calculated values of cavity temperature and pressure are obtained from the package thermal analysis (Chapter 3, Sect. 3.4 of the SARP).

Other factors may also affect the cavity pressure and perhaps increase the number and quantity of isotopes that might be available for release. These factors include

- the presence of residual or entrained water in the package cavity or contents which could form steam when heated,
- radiolytic decomposition of water or organics in the package cavity to form oxygen or hydrogen,
- the release of gas under pressure from the contents (e.g., a sealed source) to the cavity as a result of some failure mechanism,
- the presence of significant quantities of residual contamination in the cavity or surface contamination of the contents, and
- conservative assumptions by the package designer.

Infrequently, contents packaged in secondary containers in the cavity may make some small contribution to the internal pressure and/or to the radionuclide inventory available for release. Consequently, the designer should review secondary container performance under normal transport conditions so that contributions to pressure and inventory from secondary containers can be included in the calculations if appropriate.

The pressure and temperature conditions that exist in normal transport are often significantly lower than those for the hypothetical accident condition primarily because of the hypothetical fire accident. In practice, the hypothetical accident-condition environment usually influences the seal material and configuration requirements, while normal transport conditions influence the containment and leak-test requirements. Typically, no credit is taken for the presence of a secondary container unless it is a qualified containment boundary.

8.2.3 Hypothetical Accident Conditions

The principal driving force for leaks under the hypothetical accident conditions is the increase in pressure resulting from (1) heatup of the medium under thermal accident conditions and (2) any contribution to the cavity pressure from gases that may be released from the contents because of increased thermal energy, fracture of the contents, or failure of secondary containers.

In calculating the cavity pressure and radionuclide inventory available for release under the hypothetical accident conditions, the designer usually makes conservative assumptions that lead to the largest values for each.

As in the case for normal transport conditions, the designer must consider increased cavity pressure and releasable radionuclide inventory that could arise in the hypothetical accident conditions from the formation of steam from residual water or other liquids in the package or the contents and from release of radionuclides by structural failure of the contents or secondary containers. Cavity pressure and temperature for accident conditions are normally obtained from Chapter 3, Sect. 3.5 of the SARP.

In many cases, there may be little or no difference in the estimated inventory of releasable radionuclides between the normal transport and accident conditions. The designer uses the worst case (i.e., normal transport or accident) condition to determine the allowable leak rate.

An important factor in evaluating package containment capabilities under hypothetical accident conditions is the maximum estimated temperature at the locations where seals are used in the containment boundary. The designer must ensure that the thermal model used predicts accurately the maximum temperature of the sealing surface. The containment boundary seals must maintain containment capability over the temperature range from –40°C to the maximum temperature experienced by the sealing surfaces, which typically occurs during or immediately following the hypothetical fire-accident condition. This and other seal selection criteria are discussed in Sect. 8.5.

Finally, the containment boundary must also remain intact during the drop-and-puncture hypothetical accident conditions. Failure of any part of the containment boundary is not permitted. This condition must be demonstrated in the package structural analysis of the SARP, but it may be incorporated by reference to Sect. 4.3 of the containment analysis.

Infrequently, the designer may need to consider the use of a pressure-relief valve to prevent possible overpressurization of the cavity, which could then lead to failure of the containment boundary. Even though the use of a relief valve may have mechanical merit, regulatory approval of its use may prove difficult. Key among the concerns are that the operation of the relief valve is, for all practical purposes, an uncontrolled release and that the reclosing of the relief valve after venting can usually not be guaranteed. In operation of the relief valve, the designer may find it difficult to show that the allowable release has not been exceeded, and frequent testing (more often than annually) may be required to show that there is reasonable assurance that the relief valve reseats at some lower pressure. The designer may wish to consider the merits of dewatering, vacuum drying, or other actions that could result in lower pressures in the cavity as an alternative to use of the relief valve. Nevertheless, a relief valve may be considered if the allowable release limits can not be exceeded because of the character of the contents, special-form considerations, or other factors.

8.2.4 Operations

The most important aspect of the containment boundary is that it must provide the level of containment required for the proposed contents. The fact that the containment boundary must be reestablished each time the package is closed requires that its operational requirements and characteristics be considered concurrently with its design requirements. The containment design should be evaluated according to the requirements placed on the package operator or user in its operating environment. The purpose of this section is to address operational considerations in a philosophical sense. Configuration and design of a package is also addressed in some detail in Sect. 8.6 of this chapter and in Chapter 12.

The operational aspects of the containment boundary design are important because each user will have to carry out specific operations on the package to achieve and verify containment. All packaging operations normally occur in a radiation environment and, at least half of the time, with the package loaded. To the greatest extent possible, the designer should configure the containment features so that total radiation dose to the user is as "low as reasonably achievable" (ALARA).

Radiation to which users are exposed include the background levels found in and around the plant and the radiation field around the package itself emanating from its contents. The dose from background radiation can be reduced by reducing the handling time for the package in the plant. Consequently, the design should allow for rapid, efficient closure and testing of the package. Dose from the contents is reduced by the same method and by locating penetrations near the edge of the package so that a user can work in a comfortable position at arms' length from the package. Fittings and fixtures should not be of unique designs that may (1) preclude a user from anticipating correct operation, (2) require a number of motions or special tools to operate or actuate, or (3) require unnecessarily long replacement or maintenance times. The configuration and operation of each penetration and its associated fitting should reduce or eliminate the potential for error when containment is established and verified.

The package designer should consider the need for controlled venting of the package cavity before opening. The package cavity may be under pressure, and the cavity medium may contain respirable particles or gases. The operator should be able to vent the cavity to a facility waste-handling system without undue risk of exposure to airborne contamination. In addition, a need may exist to sample the medium in the cavity before opening the package in order to verify the condition of the contents. All ports (e.g., vent and sample) should be readily accessible to the user

and unequivocally marked for function. Typically, a package is designed with a vent port, which is used also for sampling, and a drain port, which is connected to the bottom of the package cavity by appropriate pipe or drilled passage. Even though drain lines are frequently omitted from packages that are used dry, the designer may wish to include a drain line to facilitate (wet) chemical or steam decontamination of the package or to increase the future flexibility in the use of package. If the package must be decontaminated without the drain, then liquids may have to be absorbed, mopped, or vacuumed dry.

Seal placement should reflect the need for visual inspections of seal surfaces and the replacement of seals on a per-use or other basis. The package design should not require the operator to reach into or work in high radiation areas and should allow the wipe down of the area around the seal to reduce transferable contamination. The seal should be designed so that it is retained in its correct position under reasonable handling conditions. The operator can be expected to have reduced visual acuity and limited manual dexterity because of the use of a respirator faceplate and multiple layers of gloves. Contending with small fasteners or retention clips or retrieving fallen seals often creates problems for the user.

The package and the penetration closures are normally bolted; these fasteners are torqued to values specified in the operating instructions. The designer should consider the difficulty of achieving high bolt-torque values in the field. Torque exceeding about 815 N·m (600 ft-lb) is difficult to apply manually, especially if the operator must work from temporary scaffolding when accessing the top of packages.[12] Even though some mechanical aids are available, maintaining and demonstrating calibration of air- or electric-driven torque wrenches can also be difficult.

It is important to recall that the package will be designed and built once, but it may be used hundreds of times. The design cannot be fixed in the field. Stepping through the required handling actions for the package while considering the requirements on the operator and the environment in which the package is used assists the designer in ensuring that the containment requirement is safely, efficiently, and correctly met.

8.2.5 Use of ANSI Standard N14.5

The Standard specifies a method of estimating a leak rate which can be compared with the allowable release rate for the package. It provides guidance in specifying the leak-test conditions that demonstrate compliance of the package to its containment performance requirement. It also provides for estimating the leak rate by an actual measurement of leaked quantities in real time. This Standard is endorsed by the NRC in *Regulatory Guide 7.4, Leakage Tests on Packages for Shipment of Radioactive Materials*, as a generally acceptable procedure for demonstrating compliance with regulatory requirements. Reliance on other methods can be acceptable, but may result in the need to provide additional information or documentation to the regulator to prove the method works and is accurate.

The Standard presents the steps that should be followed once the package-containment requirements are known. It does this chiefly by providing formulas to be applied under various conditions or circumstances and by providing examples of the application of the formulas. The Standard also contains a chart, which is reproduced in Fig. 8.1. As can be deduced from the figure, the strategy is to ensure that specified leak tests for package (1) postfabrication acceptance, and (2) periodic, and (3) per-use conditions are adequate to demonstrate package compliance with its allowable leak rate (if any).

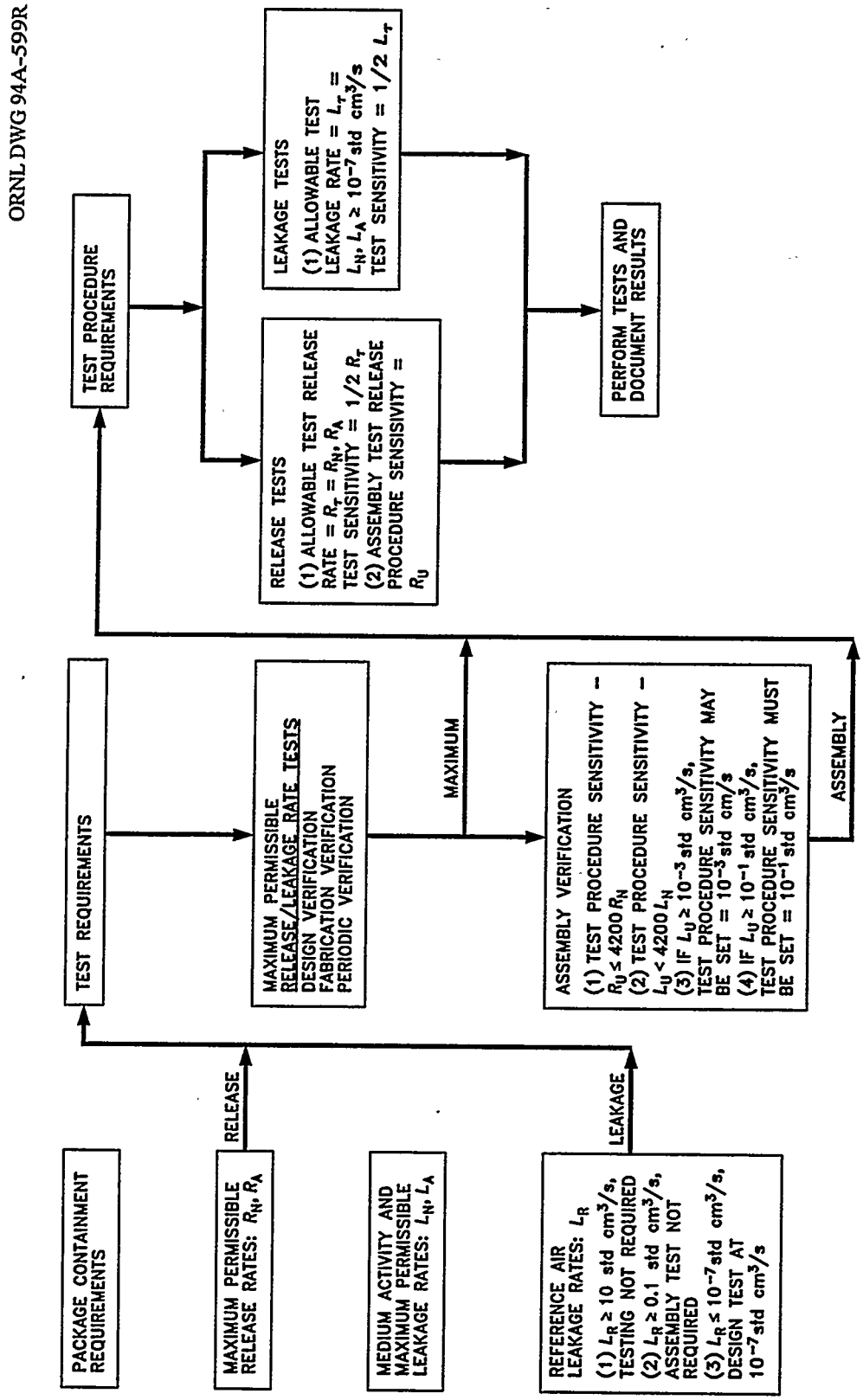

Fig. 8.1. How to use the Standard. This figure is reproduced with permission from American National Standards Institute, Inc., ANSI N14.5, copyright 1990, by the American National Standards Institute, 11 West 42nd Street, New York, NY 10036.

Before the application of the Standard, the designer needs to determine the allowable release rate using the calculational methods given in Appendix A of 10 CFR Part 71 or the similar table in the IAEA Safety Series 6.[13] The designer must have considerable information about the proposed contents, their form and individual constituents, and their proportions, and normal transport and accident condition characteristics and behavior.

A significant portion of this chapter is devoted to stepping through a leak-rate calculation to show how the Standard is applied to a hypothetical package and its contents. The intent is to aid the designer in visualizing the process and to supplement the examples provided in the Standard.

8.3 CONTAINMENT ANALYSIS FOR GASES

A common containment analysis that is performed is to determine the allowable leak rate for gases in the package cavity, and to then establish the containment boundary test that demonstrates that the allowable leak rate is not exceeded.

This section presents an example of the typical calculations that are required for the analysis of mixed gases that might be found in a package cavity, within the containment boundary. For the example, it is assumed that the designer has identified krypton (^{85}Kr, 3660 Ci), xenon (^{131m}Xe, 225 Ci, and ^{133}Xe, 360 Ci), and tritium (400 Ci) as the radionuclides that could be available in the free volume of the package cavity. This mixture of gases is used to demonstrate the use of the formula presented in Appendix A of 10 CFR 71 Part 71 and later in ANSI N 14.5 that relates to the calculation of standard leak rates. It is not intended to be representative of any particular package or shipment. As discussed below, the actual gases included in any mixture are determined by the designer and are based on the proposed package contents.

8.3.1 Package Source Term and Determination of the A_2 Value

The source term associated with the contents of the package may be calculated (1) by using a computer code, (2) by applying knowledge of the process that formed the contents, (3) by sample or assay, (4) by direct measurements, or (5) by any similar strategy that identifies the principal radioactive inventory by nuclide and quantity. This "constituents and quantity" knowledge of the material is also needed to complete the description of the package contents required in Sect. 1.2.3, and in the shielding and criticality analysis chapters of the SARP.

Once the radionuclide inventory is known or has been estimated, the designer can use Table A.1 of 10 CFR Part 71, Appendix A to determine the A_2 value for the constituents of the package contents. The values listed in Table A.1 for A_2 (and for A_1) have the units of curies. Calculation of the A_2 values is a first step in determining the package allowable leak rate.

To determine an A_2 value for the proposed contents, the designer must know or must estimate the radionuclide constituents under both normal transport and accident conditions. Note that the A_2 value itself represents a level of activity of any radionuclide that would prevent radiological effects in humans from exceeding a level consistent with the protection standards established by the International Commission on Radiological Protection.

If the designer does not know the specific radionuclides that make up the source term, 10 CFR Part 71, Appendix A, contains a number of rules for determining the A_2 value, most of which

simplify the process. For example, it allows the use of $A_2 = 0.4$ Ci, for mixed fission products if a detailed analysis of the mixture is not carried out. Some of the rules apply to particular kinds of radionuclides; therefore, the designer must review the rules to ensure that the appropriate method of determination of the A_2 value is applied.

A common application of Appendix A is the calculation of a composite A_2 value for a gaseous mixture of radionuclides in the cavity of a package. An example of this application follows. As previously noted, the designer needs to know the expected radionuclide constituents of the mixture and the quantity (in curies) of each of the components. The radionuclides may be daughter products of known radioactive species, by (usually slow) release of entrained gases that form during the principal use of the contents, or by other means. The designer must also note that the gaseous (and other) constituents in the cavity (and potentially available for release) could change in proportion or quantity when normal transport conditions and accident conditions are considered. In this event, an A_2 value must be calculated for both conditions.

Example

For this example involving mixed gases, it is assumed that the designer has identified that the activity in the package contents includes 3660 Ci of ^{85}Kr, 2250 Ci of ^{131m}Xe, 3600 Ci of ^{133}Xe, and 400 Ci of tritium.

From Table A.1 of 10 CFR Part 71, the A_2 value for ^{85}Kr, tritium, and ^{133}Xe is 1000 Ci, and for ^{133m}Xe, it is 100. The A_2 column is used because the material is not "special form."

It is assumed that from knowledge of the contents or process, or by historical measurement, only 3 to 5% of the existing xenon will escape the contents (e.g., by slow migration) and be available for release from the cavity under transport conditions. For this example, it is conservatively assumed that 10% of the xenon escapes (about twice the expected quantity) to the package cavity free volume, plus all of the ^{85}Kr and tritium.

The determination of a composite A_2 value of the assumed releasable contents is generated manipulating the information in Table 8.1 from which the composite A_2 value is calculated as: $1/3.16 \times 10^{-3} = 316$ Ci.

Table 8.1. Determination of composite A_2 value

Material	A_2 (Ci)	Activity in the contents (Ci)	Activity available for release (Ci)	F_i	F_i/A_2 $(Ci)^{-1}$
^{85}Kr	270	3660	3660	0.788	2.9×10^{-3}
Tritium	1080	400	400	0.086	7.9×10^{-5}
^{131m}Xe	1080	2250	225	0.048	4.4×10^{-5}
^{133}Xe	541	3600	360	0.078	1.4×10^{-4}
Totals			4645	1.000	3.16×10^{-3}

In this table, "activity in the contents" is the total activity (in curies) of each of the gaseous radionuclides in the contents, and "activity available for release" is the activity of each radionuclide (also in curies) that is considered to be available for release. This latter column reflects the assumption that 10% of the xenon in the contents is available for release. F_i is the fractional contribution of each of the radionuclides, and that it is dimensionless. For example, the F_i value for tritium is 400 Ci (of tritium)/4645 Ci (total) = 0.086. Because the sum of the fractional parts cannot be greater than the whole, the total F_i is 1.

F_i/A_2 is the fractional activity of each radionuclide divided by that radionuclide's A_2 value, or for tritium, F(tritium)/A_2 (tritium) = 0.086/1080 = 7.9×10^{-5}/Ci. These individual values are then summed, and the composite A_2 is obtained by taking the reciprocal of the total.

This table format is used because its construction lends itself to using a spreadsheet program and because it is somewhat easier to manipulate than the summation formulas provided in Appendix A of 10 CFR Part 71.

For contents consisting of a single radionuclide, the A_2 value is that of the element. For mixtures of radionuclides that have the same A_2 value, the A_2 value of the mixture is the same as that of any element, regardless of the number of curies of each.

The calculation of the A_2 value for aerosols, liquids, and solids, including particulates, is performed in the same way as it is for gases.

The remaining steps in the leak-rate calculation follow the method of the Standard, and the equations used in the following are taken from it.[4]

8.3.2 Calculation of Permissible Leakage Rates

The A_2 value is used to first calculate the maximum permissible release rate and then the maximum permissible leakage rate for both normal transport and accident conditions.

The maximum permissible release rates are generated from equations specified in the regulations for normal transport and for accident conditions. These rates are designated as R_N for the normal transport release rate and as R_A for the accident conditions release rate. From the regulations,

$$R_N \leq A_2 \times 10^{-6} \text{ Ci/h, and} \tag{8.1}$$

$$R_N \leq A_2 \text{ Ci/week (except 10 } A_2 \text{ Ci of } {}^{85}\text{Kr per week)} \ . \tag{8.2}$$

Something that is often overlooked is that the regulations (10 CFR 71.51) specify that the package leak tightness must be demonstrated to a *sensitivity* of $A_2 \times 10^{-6}$ Ci/h. Since test sensitivity is specified as $R_N/2$ in Sect. 8.4 of the Standard, the allowable leak rate is then: $R_N = 2A_2 \times 10^{-6}$ Ci/h. Ordinarily, the leak rate is calculated, and reported, as $A_2 \times 10^{-6}$ Ci/h, and the test sensitivity is set to one-half of this value.

Using the composite A_2 value calculated in the example presented in Sect. 8.3 (Table 8.1), A_2 = 316 Ci,

$$R_N = 3.16 \times 10^{-4} \text{ Ci/h, and}$$

$R_A = 3.16 \times 10^2$ Ci/week .

In specifying the R_A value, the exception allowed for a ^{85}Kr source (except 10 A_2 Ci of ^{85}Kr per week) has been conservatively ignored; and the A_2 value calculated in Table 8.1 includes the contribution of the ^{85}Kr in the contents. It is suggested that this strategy be used unless ^{85}Kr is the only gas expected to be released by the contents. As a practical matter, no good example of an application of the exception may exist.

The maximum permissible release rates (R_N and R_A) are used to calculate the maximum permissible leakage rates, L_N and L_A, where, from the Standard:

$$L_N = R_N/C_N \,, \tag{8.3}$$

and

$$L_A = R_A/C_A \,. \tag{8.4}$$

C_N and C_A are the activities per unit volume of the medium that could potentially escape from the containment system under normal and accident conditions. The units are curies per cubic centimeter (Ci/cm^3).

The activity per unit volume available for release under accident conditions could be much greater than that under normal transport conditions. One example is the assumed failure of 1–3% of spent fuel rods in normal transport of irradiated fuel for the purpose of calculating R_N and the assumed failure of all the rods for calculating R_A. Another example is one in which processed material, such as powders or resins, are placed in a secondary container and that container is assumed to be sealed in normal transport, but to fail under the accident conditions. As previously discussed, the task for the designer is to estimate, based on knowledge of the contents and its behavior, the quantity of radionuclides that are potentially available for release in both normal transport and accident conditions. The volume that must be considered is the free space, or void volume, within the package cavity which contains the medium that could be released. The free space is the space within the containment boundary not otherwise occupied by solid contents, spacing materials, dunnage, secondary container materials, etc.

In calculating the activity concentrations, C_N and C_A, the designer must consider the various contents and contents configurations that could be used in the package during shipment and then select the worst-case conditions for analysis.

The free space in the cavity does not usually change between the normal transport and accident conditions, but it is possible for it to do so in some designs. For example, if there is an impact-absorbing material within the containment boundary intended to absorb the drop energy of the contents in the 9-m drop accident, the free volume could increase under the accident condition as the impact-absorbing material is crushed. The designer should conservatively ignore the increase in volume, which decreases concentration because it is often impossible to know the extent of crush that could occur in a random accident.

Usually, the designer will find that the R_N value controls the allowable release rate because it is normally the lower of the two allowable release rates. (Here, for example, the R_A value is 316 Ci in 1 week, but $R_N = 3.16 \times 10^{-4}$ Ci/h $\times$ 168 h/week = 0.05 Ci in 1 week.)

To calculate L_N and L_A, it is assumed that the package has a free volume, V, of 0.0212 m^3 (0.75 ft^3) available in both normal transport and accident conditions for the releasable contents described above (A_2 = 316 Ci). In the example, the available releasable material was found to be 4,645 Ci, from which

$$V_N = V_A = 0.0212 \text{ m}^3 = 2.12 \times 10^4 \text{ cm}^3 ,$$

and

$$C_N = C_A = 4{,}645 \text{ Ci}/2.12 \times 10^4 \text{ cm}^3 = 0.22 \text{ Ci/cm}^3 .$$

Then, from Eq. (8.3),

$$L_N = R_N/C_N = (3.16 \times 10^{-4} \text{ Ci/h})/(0.22 \text{ Ci/cm}^3) = 1.44 \times 10^{-3} \text{ cm}^3/\text{h} ,$$

$$= 1.44 \times 10^{-3} \text{ cm}^3/\text{h} \times \text{h}/3600 \text{ s} ,$$

$$= 4.0 \times 10^{-7} \text{ cm}^3/\text{s} ;$$

and from Eq. (8.4),

$$L_A = R_A/C_A = (316 \text{ Ci/week})/(0.22 \text{ Ci/cm}^3) = 1.44 \times 10^3 \text{ cm}^3/\text{week} ,$$

$$= 1.44 \times 10^3 \text{ cm}^3/\text{week} \times \text{week}/168 \text{ h} = 8.5 \text{ cm}^3/\text{h} ,$$

$$= 8.5 \text{ cm}^3/\text{h} \times \text{h}/3600 \text{ s} ,$$

$$= 2.37 \times 10^{-3} \text{ cm}^3/\text{s} .$$

For the example chosen, the permissible activity release rate for normal transport (L_N) is found to be more restrictive than the permissible accident release rate, (L_A). Consequently, only the more restrictive L_N is considered in the subsequent analysis.

8.3.3 Calculating the Reference Air-Leakage Rate

In the preceding section, a maximum permissible leakage rate was calculated for normal conditions (L_N) and for accident conditions (L_A). One purpose of calculating these rates is to identify an appropriate leak test that must be performed on the package to demonstrate that the allowable leak rate is not exceeded.

Table A.1 of the Standard presents the nominal sensitivity of various kinds of leak-test methods. In the table, the test sensitivity is specified in standard cm^3/s, where "standard" refers to dry air at 1 atm absolute pressure at 25°C (298 K). Therefore, the maximum permissible leakage rates, L_N and L_A, must be converted to an equivalent air-leakage rate under standard conditions before Table A.1 can be used to identify possible leak-test methods.

In addition, the designer should note that Sect. 6.2 of the Standard states that the reference air leak rate is the leak rate that is to be the acceptance criteria for periodic leak testing, package acceptance and post-repair leak testing. This criteria is reasonable in that, in the event of a leak,

the molecular weight of the leaking gas is expected to be closer to that of air (29) than to helium (4).

To make the conversion, it is necessary to know the media (air, inert gas, etc.) that are in the free volume of the package and the maximum normal transport and accident-condition temperature and pressure conditions in the package. Continuing with the example, it is assumed that the package's operating procedure specifies that the package will be closed, vacuum-dried, and backfilled with helium to 204 kPa (15 psig). It is also assumed that the package is closed at room temperature (25°C) and that the thermal analysis of the package establishes that the maximum temperature of the containment cavity in normal transport results from solar insolation and is 47°C. From this information the internal pressure is found to be 220 kPa (17.2 psig). For the accident conditions, the temperature and pressure values of the cavity are 250°C and 360 kPa (37.4 psig), respectively.

Leaks that are in the range of interest of the package designer, from 1×10^{-2} cm³/s to 1×10^{-7} cm³/s, are considered to be made up of two leak-flow regimes: One is continuum leakage, F_c, which is characterized by laminar, or viscous, flow, and the other is molecular flow, F_m, which is characterized by low leakage through fine capillary paths. Molecular flow is associated with leak rates of about 1×10^{-5} cm³/s and less.

In establishing the reference air leak for normal transport conditions, L_R, it is assumed that the leak is composed of both continuum and molecular flows; Eq. (B.2) of the Standard [see Eq. (8.5) herein] is applied. The strategy is to solve Eq. (8.5) (after some substitutions for F_c and F_m) for the diameter of the leak pathway that results from the maximum permitted leak rate, L_N, found in Sect. 8.3.2. Then Eq. (B.20) [see Eq. (8.5) below] is solved to find the rate of air leakage through that same-diameter leak under standard conditions.

The practical result is that, at least analytically, the permitted leak rate is related to a theoretical leak of a given-diameter capillary. That diameter is subsequently used to establish the allowable reference air leak at standard conditions.

Continuing, Eq. (8.5) is applied, substituting in it the equations for F_c and F_m, which are given in the Standard as Eq. (B3) (see Eq. 8.6 below) and (B4) (see Eq. 8.7 below), respectively.

$$L = (F_c + F_m)\,(P_u - P_d)\ \text{cm}^3/\text{s}\ ; \tag{8.5}$$

$$F_c = (2.49 \times 10^6)\,(D^4)/(a\mu)\ \text{cm}^3/\text{atm-s; and} \tag{8.6}$$

$$F_m = (3.81 \times 10^3)\,(D^3)\,([T/M]^{1/2})/(aP_a)\ \text{cm}^3/\text{atm-s.} \tag{8.7}$$

In these equations:

L = a volumetric leakage rate,
F_c = the coefficient of continuum flow,
F_m = the coefficient of molecular flow,
P_u = the upstream pressure (pressure at origin of leak),
P_d = the downstream pressure (pressure at destination of leak),
D = the leakage hole diameter,
a = the leakage-path hole length,
μ = the gas viscosity,
T = the gas absolute temperature,

M = the gas molecular weight, and
P_a = the average stream pressure $[(P_u + P_d)/2]$.

The leakage-path hole length, a, is determined by a review of the package design drawings as the distance across the compressed seal in the containment boundary. The shortest distance should be used if more than one size of seal is used. In this example, it is assumed that the leakage distance is found from the package design drawings to be 0.635 cm (0.25 in.).

The viscosity used is that of the leaking gas. Frequently, the gas is helium or air. In the event of an actual leak, the leaking gas is typically helium or air. As in the working example, the leaking gas is often composed of several gases, and the viscosity, as well as the molecular weight (M) and partial pressures, should be calculated for the mixture of gases. The mixtures can be calculated using the same technique as was used to calculate the fraction activity of the cavity gas. The designer is referred to Eqs. (B-6), (B-7) and (B-8) of the Standard, and to the example presented in Sect. 8.6.3 of this chapter. It is reasonable to use the cover gas in the calculation of the leak rate because it is typically the gas that is used in the qualification leak test.

The viscosity of helium at 47°C is found by interpolation of the published values[14] to be 0.0205 c_p.

P_d = 101 kPa (1.0 atm—the leak is to the atmosphere),
P_u = 220 kPa (2.17 atm),
P_a = $1/2\ (P_u + P_d) = 1/2\ (220 + 101) = 161$ kPa (1.59 atm),
M_{he} = 4 gmol, and
T = 47°C = 320 K.

From the values specified or calculated, then:

$$F_c = (2.49 \times 10^6)\ (D^4)/(a\mu)\ \text{cm}^3/\text{atm s}$$
$$= (2.49 \times 10^6)\ (D^4)/(0.635 \times 0.0205)$$
$$= (1.91 \times 10^8)\ (D^4),$$

$$F_m = (3.81 \times 10^3)\ (D^3)\ [(T/M)^{1/2}]/(aP_a)\ \text{cm}^3/\text{atm s}$$
$$= (3.81 \times 10^3)\ (D^3)\ [(320/4)^{1/2}]/(0.635 \times 1.59)$$
$$= (3.81 \times 10^3)\ (D^3)\ (8.94)/1.01$$
$$= (3.57 \times 10^4)\ (D^3),$$

and

$$L_N = (F_c + F_m)\ (P_u - P_d)\ \text{cm}^3/\text{s}$$
$$= [(1.91 \times 10^8)\ (D^4) + (4.98 \times 10^4)\ (D^3)]\ (2.17 - 1.0)$$
$$= [(1.91 \times 10^8)D + (3.57 \times 10^4)]\ (D^3)(1.17)\ \text{cm}^3/\text{s}.$$

But from Sect. 8.3.2 above,

$$L_N = 4.0 \times 10^{-7}\ \text{cm}^3/\text{s}\ .$$

Setting these two values equal to each other,

$$L_N = 4.0 \times 10^{-7} = [1.91 \times 10^8\ D + 3.37 \times 10^4](D^3)(1.17)\ ,$$

from which

$$D = 1.8 \times 10^{-4}\ \text{cm (diameter of leakage hole)}\ .$$

Looking ahead, it is noted from Table B.2 of the Standard that a reference air leak rate on the order of 2×10^{-7} cm³/s should be expected, since the postulated diameter of the leakage hole is 1.8×10^{-4} cm.

To calculate the reference air leak, this value of D is returned to the equations for the coefficient of continum flow, F_c (Eq. 8.6) and for the coefficient of molecular flow, F_m (Eq. 8.7). These equations must be modified to account for air at standard conditions as the leaking media by using the viscosity of air ($\mu = 0.0185$) in Eq. (8.6). In Eq. (8.7), the mass of air ($M = 29$) and $P_a = 0.505$ should be used. Consequently:

$$F_c = [(2.49 \times 10^6)(D^4)]\ /\ [(0.635)(0.0185)]$$
$$= 2.22 \times 10^{-7}\ \text{cm}^3/\text{atm-s}\ ;$$

$$F_m = [(3.81 \times 10^3)(D^3)\sqrt{\frac{298}{29}}]\ /\ [(0.635)(0.505)$$
$$= 2.21 \times 10^{-7}\ \text{cm}^3/\text{atm-s}\ .$$

The reference air leakage rate, L_R can be found by using Eq. (B.5) of the Standard [see Eq. (8.8) below],

$$L_R = (F_c + F_m)(P_u - P_d)\ \text{cm}_3/\text{s} \tag{8.8}$$

where $P_d = 0.01$, $P_u = 1.0$;

$$L_R = (2.22 \times 10^{-7}) + 2.21 \times 10^{-7})(0.99)$$
$$= (4.38 \times 10^{-7})\ \text{std cm}^3/\text{s}.$$

This reference leak rate is less than the maximum permissible leakage rate calculated for normal conditions of transport. This is reasonable because it is expected that air would leak at a slower rate than helium, based on the difference in molecular weight.

Returning briefly to the concept of continuum vs molecular flow, it is sometimes helpful to determine, which of the flows predominates using the ratio:

$$r_f = F_c/F_m\ . \tag{8.9}$$

The advantage of this is that the flow that does not predominate (either F_c or F_m) can generally be set to zero. This simplifies the leak-rate equations that must be solved, with little loss of accuracy. In this example, the flows are equal, and r_f, the ratio of the flows, is found to be 1.0.

This ratio indicates that neither flow predominates, so neither flow can be ignored. Consequently, both flows should be considered, as in the method previously used to solve for L_R.

Usually, if r_f is greater than 10, then the molecular flow is less than 10% of the total flow, and F_m can be ignored [F_m = 0.0 in Eq. (8.8)]. Alternatively, if the ratio is less than 0.1, then the continuum flow is less than 10% of the total flow, and F_c can be ignored.

If r_f is between 0.1 and 10, the designer should consider the contribution of both flows in the calculation of the total leak rate. Alternatively, the designer could initially assume that one or the other flow is zero, depending on the r_f value, and then add back a percentage of the nonzero flow based on the value of r_f. For example, if $r_f = 5.0$, the designer could assume $F_m = 0.0$, and then add back 50% of the F_c flow as representative of the actual F_m flow contribution.

A final observation or two about the allowable leak rate. First, the leak test would be performed using helium as a tracer gas. As previously noted, helium is one of the few gases for which test equipment is readily available to either verify leak rate or to merely detect the presence of a leak in this range. Therefore, it is advantageous to specify the allowable leak rate in terms of helium, rather than in terms of air. Second, in this example, the allowable leak rate is sufficiently close to the definition of "leak tight" that the designer should specify the test that would demonstrate leak-tight conditions.

Using an identical process, except ignoring the F_m contribution and any increase in pressure due to accident conditions, the reference air leak rate for the accident condition, L_A, is found to be 7.12×10^{-4} std cm^3/s. Because this is a significantly larger permitted leak rate than the normal transport-conditions-permitted leak rate, the leak-test method chosen must be based on the normal transport requirement. As a practical matter, the designer will often find that the leak-rate test requirement is driven by the package normal conditions of transport.

A comparison of the leak-test requirement to a leak-test method is presented in Sect. 8.6.3, "Periodic Inspections."

8.4 CONTAINMENT ANALYSIS FOR NONGASES

The designer must frequently consider the presence of nongas material in the package that could leak during normal transport and accident conditions. This material includes aerosols, particulates, and liquids. Of these, the estimates of particulate contribution to the concentration of the material available for leakage may be the most difficult to quantify, primarily because of the lack of data on the solid material that may be available for release. The effects of nongases may also be considered in the evaluation of containment of spent fuel. Spent fuel volatiles and particulates are discussed in Sect. 8.4.1.1.

8.4.1 Aerosols and Particulates

Aerosols are formed by the suspension of vapor or fine particles in a gaseous medium. The aerosol may also include volatile species, such as cesium, from the contents or in the package cavity as residual material. The leakage rate for aerosols is calculated in the same manner as for gases, following the method presented in Sect. 8.3. This method is considered to be conservative, resulting in a predicted leak rate that is greater than the (expected) actual leak rate. Particles are

subject to settling and plating out on cooler surfaces, a process which tends to remove them from the gaseous medium that could leak. Particles also tend to block leak paths that may exist. There are typically no postaccident forces which operate to introduce new material into the medium that may leak or to keep entrained material suspended in it. For example, cesium boils at a temperature of about 670°C (1238°F), which is well above the temperature that would exist in the interior of a transport package in the hypothetical fire accident event. Consequently, only a small amount of volatile cesium could be expected to be available for release from the typical package contents following the hypothetical accident.

One difficulty in establishing the curie concentration of the medium (the aerosol) that could escape is in estimating (1) the amount of nongas material that may be entrained in it and (2) the distribution of particle sizes within the nongas materials that become available for release. Clearly, particles that are too large to pass through the postulated leak opening cannot escape the containment boundary, but they could act to clog the leak path, preventing the escape of smaller particles and gas. Even particles that are smaller than the leak-hole diameter can combine to bridge the hole, causing blockage. Further, the analysis assumes a single leak opening having a flow rate equivalent to the total leak rate. The actual leak path more likely consists of multiple smaller holes, with geometries that are variable in terms of straightness and smoothness. These irregular, small paths are more likely to become clogged by even very fine particles.[14]

Obviously, particle size can be important to the designer because even if the contents become fully fractured under accident conditions or even if the contents consist entirely of powder, release would not occur if all of the available particles were larger than the leak orifice. If the proposed contents are in a powdered form and the designer knows or can determine the distribution of particle sizes, then a leak rate based on particle size can be established (i.e., if the grain size of the powder is larger than the calculated diameter of the postulated leak path, then escape of the powder cannot occur).

Finally, an assumption of 100% dispersibility for the contents of some packages may not necessarily result in a leak-tight containment requirement. An extensive analysis of containment requirements for gross residual contamination in spent fuel casks shows that comparatively large permissible leak rates can be calculated even if it is assumed that 100% of the contents are in a fractured form.[15]

Because particle size is an important factor in the postulated formation of an aerosol, a significant amount of research has been devoted to particle size and its relationship to estimated leak rates. Empirically, it is expected that the particle size has a Gaussian, or normal, distribution with the vast majority of the particles sufficiently large as to be outside our range of interest. Some specific contents may have the size distribution skewed—all of the contents consist of fine particles, or none of the contents is fine particles. The designer may be able to apply knowledge of the contents or knowledge of the process to assist in making reasonable determinations about the particle size distribution and the dispersal of the contents within the package cavity. Unfortunately, there is usually little or no information about particle size distribution of most source material under normal transport conditions and even less knowledge about particle size under postaccident conditions. The question becomes: What is a reasonable assumption to make regarding the availability of fine particles for release during normal transport and for hypothetical accident conditions?

Much of the research performed to date to study particle size distribution involves the use of uranium oxide (UO_2) as a model for plutonium or plutonium oxide.[16–19] This research involves the

use of powder consisting of very fine particles (more than half of the particles in the 1–5 μm [microns] range), subjected to induced vibration which provides the mechanism for particle dispersal in the cavity and continued entrainment in the medium. Orifices of various sizes were used in each of these tests. The general conclusions from these studies is that orifices up to about 200-μm-diam clog readily, even at high test pressures [6,890 kPa (1000 psi) pressure drop across the orifice was used in about 200 experiments] and with the imposition of vibration on the package.[16] For perspective, the calculated diameter of the leak pathway in the working example was 0.000183 cm, or 0.18μm. If the allowable leak rate had been on the order of 1×10^{-3} standard cm^3/s, then the leak pathway diameter would be approximately 2 μm.

Even though these studies have shown that even small particles do not easily pass through a leak hole, they give only empirical evidence that the percentage of solid material that might become airborne, and stay airborne, for purposes of a leak-rate calculation, is low. Other studies have been conducted on the distribution of fine particles from spent fuel rod failures.[20–22] At least one study indicates that only about 10% of the available bulk material appears as widely distributed fine particles.[22] A 10% aerosol of fuel fines was used in one major study on the containment requirements of spent fuel shipping casks.[23,24]

The method of leak rate evaluation used by experimenters has been to apply a mass leak rate based on the measured density of the suspended material (g/cm^3). As might be suspected from the previous discussion, the maximum measured densities are low, being on the order of 10^{-6} to 10^{-9} g/cm^3, depending on the powder. Of the results that have been reported, the maximum measured density has been 9×10^{-6} g/cm^3 (ref. 20). Conservatively using this value as an expected density, the average activity (C_n) would be calculated as:

$$C_n = [\text{Specific Activity of the powder (Ci/g)}] \times (9 \times 10^{-6}\ g/cc^3)\ .$$

For a mixture of powder contents, the designer could calculate the specific activity of the mixture, or apply the specific activity that is most conservative. The allowable release rate (R_n) is calculated using the appropriate single or composite value for A_2.

The leak rate that is calculated is considered to be very conservative. This rate is based on the use of the maximum reported material density, and on the demonstrated properties that argue against high rates of leakage of fine particles through a small hole - plugging of the leak path by larger particles, and clogging of the pathway by bridging of smaller particles caught on rough surfaces within the pathway. One study has concluded that the predicted loss from a 5-μm-diam leak would be about 0.09 μg of powder, assuming a hypothetical fire accident event, and that the probability of particulate loss greater than 0.3 μg from a leak larger than 10 μm in diam is 10% (ref. 25). For perspective, a 5-μm-diam leak path represents a leak rate of 0.044 cc^3/s, while 10 μm represents a leak rate of 0.68 cm^3/s, both at standard conditions.

The designer should always be satisfied that a proposed strategy for demonstrating an allowable leak rate is conservative, based on the knowledge of the properties of the contents and package design. Otherwise, a leak-tight condition should be specified for the package.

The A_2 value of the dispersed solid particulate material should be calculated as shown in Sect. 8.3.1. The (dispersible) solid material is then treated as an aerosol (gas), using the appropriate A_2 values for the individual constituents, and their concentrations to establish the composite A_2 value for the mixture. The equations from the Standard for gases are used because it is assumed that the mixture is an aerosol behaving as a gas. The Standard does not address the

possible leakage of solid material. For an example of how an assumed particulate aerosol is addressed, the reader is referred to examples D6 and D7 of the ISO Standard, *Leakage Testing for the Safe Transport of Radioactive Materials.*[26]

Finally, the designer can establish the (actual) leak rate for suspended solid particles in a given containment boundary configuration by testing. These tests would establish the actual leak rate by sampling the external surfaces around the boundary to detect particles (e.g., by counting smears) after the package was exposed to the normal and accident conditions of transport. The designer should recognize that as a practical matter, such tests could prove challenging. The designer would also have to estimate reasonably the contents worst-case response to normal transport and accident conditions and conduct the test (or tests) using these conditions. Even though it is clear that some solid material could escape the containment boundary in some instances, it is more likely that significant release of solid material is reserved for catastrophic failure of the boundary, which has an extremely low probability of occurring.

Physical testing may have some applicability in the study of source terms for leaking coolants if the contents consist primarily of powder. However, it is likely that the results of a specific series of tests could be highly variable. The designer could expect to find a lack of reproducibility between and among tests because of (1) the way in which actual leaks might be formed, (2) the way in which particles behave relative to the actual leak path that is formed, and (3) the randomness of the particle size distribution in the vicinity of the leak.

8.4.1.1 Spent Fuel

Historically, designers of spent fuel shipping casks have only considered fission product gas in the calculation of cavity concentration and allowable leak rate. In those analyses, it was assumed that 10 to 30% of the available fission gas would escape a fuel rod that failed, depending on the gas. This gas, primarily krypton, was postulated to be in the plenum area of the rod. The remaining gas was entrained in the lattice structure of the fuel pellets where it was formed. (The total curies of available gas are determined from an ORIGEN,[9] or similar, calculation.) The gas only slowly migrated to the surface of the pellet, where it was then available for release. Largely as a matter of convention, 3 to 10% of the spent fuel rods were postulated to fail under normal transport conditions, and all of the rods were postulated to fail under the hypothetical accident conditions. Clearly, if no rods were considered to fail, then no allowable leak rate for the spent fuel could be calculated.

The potential failure rate of spent fuel rods that may occur if a shipping package carrying them were to be involved in an accident severe enough to match the regulatory hypothetical accident conditions was investigated by Sandia National Laboratories (SNL). A Sandia report[23] estimates that the most likely event that would cause rods to fail would be a 9-m oblique corner drop onto a solid, unyielding surface. Under these conditions, for General Electric 7 × 7 BWR assemblies, the highest failure rate is approximately 8×10^{-6} per rod per accident event. For one rail cask that might contain as many as 50 BWR assemblies, this translates to a possible failure of less than one pin if the cask were involved in a severe impact accident. The report also examined the situation with transporting PWR assemblies. In this case, for Babcock and Wilcox (B&W) 15 × 15 PWR assemblies, the highest failure rate was estimated to be 2×10^{-4} per rod per accident event. For one rail cask that might contain as many as 20 PWR assemblies, this translates to a possible failure of again less than one pin if the cask were involved in a severe impact accident.

Recent studies, performed by SNL and Lawrence Livermore National Laboratory, recommend that, in addition to the fission gas products, crud on fuel assembly surfaces, fission products considered to be volatile, and "fines" that resulted from the fracturing of fuel pellets within the rods be considered in the leak rate calculation.[24,27] Crud is a corrosion product whose activity comes primarily from cobalt. The principal volatile radionuclide is cesium.

The quantity of crud assumed to be present is calculated based on the surface area of the fuel assembly and an assumed surface contamination of 140 $\mu Ci/cm^2$. The surface contamination value is obtained from PNL-6237, based on assembly cooling time.[28] It can be arduous to calculate the surface area of fuel assembly, and it is not a number that the package designer normally would calculate for another purpose. Note, however, that NUREG/CR-6487 presents the surface area of typical fuel assemblies in Tables 6-4a and 6-4b.

The principal volatile is cesium, which boils at about 770°C, a temperature not likely to be reached even in the hypothetical fire accident event. Cesium can be expected to be available in only very small quantities. NUREG/CR-6487 (ref. 27) establishes a release fraction of 2×10^{-4}, while *The Standard Review Plan for Dry Storage*, NUREG-1536 (ref. 29) proposes a release fraction for cesium of 2.3×10^{-5}. The principal difference in these two values is the assumed temperature of the contents, which is lower for spent fuel in dry storage. For even a large quantity of spent fuel, this release fraction does not result in an A_2 value that is significantly different than if the cesium is ignored.

NUREG/CR-6487 proposes a release fraction of 0.00003 for fuel fines, where "fines" includes all of the radionuclides not classified as gas or volatile. This release rate is based largely on conditions not related to normal transport or hypothetical (transport) accident conditions, but there are only limited data available upon which to postulate a release fraction. This release rate should be considered highly conservative since the estimate is the maximum release fraction observed, there are no mechanisms that would tend to hold the fines in uniform suspension, and that only a small percentage of the released material is expected to be of sufficiently small size (less than 0.1 μm) so as to possibly escape a postulated leak. (In point of fact, none may be sufficiently small, given the fabrication grain size of pellets.)

The methodology for incorporating the effects of fines, volatiles and crud, with the fission gases, involves calculating the postulated quantity of the releasable material, and then calculating the A_2 value of the mixture. The table below compares the estimated releasable curies in each category for a typical PWR and BWR fuel assembly with those recommended by NUREG/CR-6487:

	Crud	Gasses	Volatiles	Fines	Total
Typical PWR	1.7	25.2	0.57	0.15	27.62
NUREG PWR	5.46	34.26	0.43	0.28	40.43
Typical BWR	8.9	9.8	0.21	0.05	18.96
NUREG BWR	18.81	12.43	0.237	0.101	31.58

The typical PWR and BWR radionuclide inventory was taken from an ORIGEN analysis of a Westinghouse 16 × 16 and a GE 9 × 9 spent fuel assembly, respectively. The point of this comparison is to emphasize that the curie values shown in NUREG/CR-6487 are conservative.

The calculated A_2 values shown in the table below can be compared in a similar way:

	Crud	Gasses	Volatiles	Fines	Total
Typical PWR	10.8	282	5.3	0.13	20.2
NUREG PWR	10.8	418.6	11.4	0.264	23.36

Here the A_2 values are approximately the same. The reason for this is that the A_2 value depends on the ratio of radionuclides rather than the quantity. As previously noted, designers have historically considered only the release of fission gas products. The inclusion of crud, volatiles and fines, has effectively lowered the A_2 value by an order of magnitude compared with what it would be (282 Ci) had only fission gases been considered. The lower A_2 value results in a lower allowable leak rate for the package. The allowable leak rate is very conservatively established.

Even so, the allowable leak rate remains in the range of 1×10^{-4} or 1×10^{-5} standard cm^3/s (air), depending on the free volume within the package cavity.

A final observation about NUREG/CR-6487: Even though very different releasable quantities of radionuclides were obtained for a typical PWR and BWR fuel assembly than was postulated by NUREG/CR-6487, the A_2 values were remarkably similar. Consequently, it may be reasonable to apply the (generic) A_2 value established by that report, and thereby avoid the calculations that are required to establish the specific A_2 value. This assumes that the A_2 values are accepted by the NRC and that the design-basis fuel is one of those considered in the report.

8.4.2 Liquids

Liquid associated with the contents of a package typically consists of residual water from incomplete draining of the contents during pool loading or from entrained water (moisture) from some process (e.g., the manufacturing of uranyl nitrate). The principal concern for the designer may be the greatly increased pressure that results from the formation of steam at temperatures that could be developed during transport or postulated accident conditions. Potential radiolytic decomposition of the liquid, which could form an explosive or flammable mixture of gases, may also be a design concern.

The interaction of any liquid with the contents during normal transport or accident conditions may give rise to the formation of a slurry, in which solid or volatile materials are suspended or dissolved to a greater or lesser degree. A slurry may also be formed by the interaction of a liquid with residual contamination (e.g., crud) in the package cavity, independent of the interaction of the liquid with the contents.

In the calculation of a leakage rate for the liquid or slurry, the designer must estimate the concentration of contaminants within the liquid, which may prove to be analytically difficult. Assumptions similar to those made for the aerosol formation in solid contents are needed to establish the concentration of radionuclides in the liquid. If steam is formed, the medium should be treated as a gas.

Compared with gases, liquids have a relatively high viscosity. Liquids are not considered to be able to flow through the fine capillary leaks associated with molecular gas flow, nor is liquid able to penetrate features such as containment seals. Because there is no molecular flow, $F_m = 0$ in Eq. (8.5), resulting in Eq. (B9) for liquids in the Standard [see Eq. (8.10)], as noted below:

$$L = F_c\,(P_u - P_d)\ \text{cm}^3/\text{s}\ . \tag{8.10}$$

As was found in Sect. 8.3.3, the calculation of F_c requires an estimate of the leakage-path diameter and length. F_c is found from Eq. (8.6) as

$$F_c = (2.49 \times 10^6)(D^4)/(a\mu)\ \text{cm}^3/\text{atm-s}\ .$$

The length of the leak path (a) can be determined from the compressed seal and seal surface dimensions; the estimate of the leakage path diameter (D) requires that Eq. (8.10) be solved. If the leak-path diameter found in the example in Sect. 8.3.2 is used and if the medium is water, then

$$D = 1.8 \times 10^{-4}\ \text{cm}\ ,$$

$$a = 0.635\ \text{cm}\ (0.25\ \text{in.})\ ,$$

$$\mu = 0.89\ \text{c}_\text{P}\ \text{at}\ 25°\text{C (water), and}$$

$$F_c = (2.49 \times 10^6)(1.8 \times 10^{-4})^4/(0.635)(0.89)$$
$$= 4.62 \times 10^{-9}\ \text{cm}^3/\text{s}\ .$$

From Eq. (8.10),

$$L_n = F_c(P_u - P_d)\ ,$$

where $P_u = 2.17$ atm and $P_d = 1.0$ atm. Substituting,

$$L_n = 4.62 \times 10^{-9}\ (1.17) = 5.4 \times 10^{-9}\ \text{cm}^3/\text{s}\ .$$

This leak rate is two orders of magnitude less than the equivalent air leak rate. The result is intuitive because water would be expected to leak much more slowly than air or another gas from a leak having a given hole size. The leak rate can be converted to a reference air leak to establish the required leak test using Eqs. (8.5–8.8) or the equivalent equations from the Standard.

As has been noted, it may be difficult for the designer to estimate the makeup of any slurry that is formed in the package, and it may be unreasonable to assume that the liquid does not pick up constituents from the contents or residual material within the cavity.

If the package is routinely expected to contain free liquids that must be considered in the leak analysis, it is suggested that the designer specify a leak-tight configuration rather than address the assumptions regarding the activity of the liquid that could escape. A related issue is the quantity of radionuclides that could become airborne in the space above the liquid due to a low vapor pressure. Even though the designer may "know" that there is little or no airborne activity, it may prove difficult to establish the technical basis for that position.

8.5 SEAL SELECTION AND USE

This section describes some of the principal concerns in seal selection and use. Issues related to the design configuration, operations, and maintenance are described primarily in Sect. 8.6. *Seal* includes the full range of elastomeric or metallic O-rings, seals, and gaskets that are intended for use in a package and which will become part of the containment boundary. Elastomeric seals include any material that is not metal, including rubber, neoprene, viton, plastic, silicone, polymeric, Teflon™, and similar materials. Metallic seals include both those that are solid and hollow.

Seal performance is perhaps the most important element in containment performance because the containment boundary is typically a metal structure interrupted by the containment penetrations. The seals at the penetrations may be metallic, but they are often fabricated from an elastomeric material that is less expensive than metal and requires less torque to seal.

The containment-boundary penetrations of a package are normally closed by bolted flanges, some of which, like the package closure, can be very large. The typical design for an O-ring seal incorporates a groove that permits the seal to deform until there is metal-to-metal contact of the flanges; that is, the closure flange does not "float" on the seal. Movement of the closure (or penetration) cover under either normal or accident conditions could damage the seal, thus resulting in the formation of a leakage path. Metal-to-metal contact ensures that the seal remains in compression and prevents incidental movement of the flange. The configuration of the seal seat prevents distortion (from compression) and, in the worst case, bursting. A second seal configuration that is less-frequently employed is referred to as a gland-type seal. Gland seals are commonly used between vertical surfaces, somewhat like piston rings between a piston and cylinder wall. In either case, the intent of the seal is the same. Several possible seal configurations are shown in Fig. 8.2 (ref. 30).

In practice, the designer specifies at least two aspects of the surfaces that the seal will contact. The first aspect is the dimensions of a machined groove, shoulder, or other feature that is expected to hold the seal in the location and position intended by the design. For some seals, the groove dimensions and configuration ensure that there is adequate room for the seal to change shape in the compression process and that the seal flattens against the mating surface. The second aspect is to specify a finish of the surface of the flanges at the area of the flange where the seal will mate to the flange. This finish is typically much smoother than the remaining surface of the package, and it usually extends about 1 in. on either side of the expected bearing point of the seal. The smoother finish improves the sealing capability of an O-ring (or gasket) by reducing the extent of interpenetration of imperfections (sealing surface roughness) that the seal must accommodate. Usually, metallic seals require a much smoother seal surface than do elastomeric seals. Groove dimensions and surface finish are normally specified by the seal manufacturer.

Initially, the designer is often concerned about the seal performance during the hypothetical fire accident, which usually leads to high temperatures in many of the structural portions of the package. The placement of the seals for the lid closure and other penetrations affects the temperatures experienced by the seal during the fire accident. The temperature performance of seals is discussed in Sect. 8.5.1.

Every seal is subject to exhibiting by-pass and permeation leakage. By-pass leakage is the leakage around the seal that the designer seeks to limit or preclude. Permeation is leakage through the seal material by the cavity medium or by the tracer gas used to test the containment boundary. Many

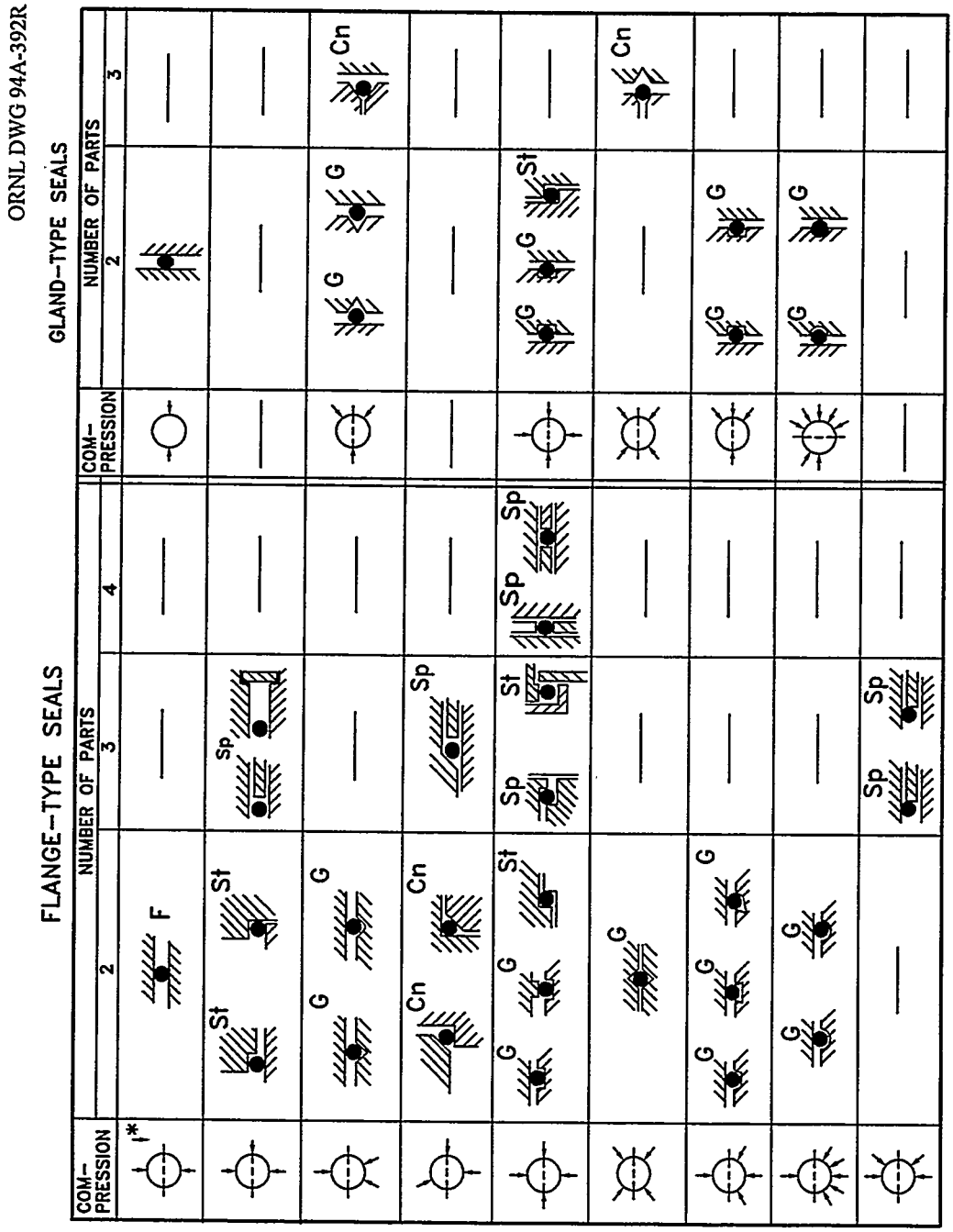

Fig. 8.2. The cross section of typical seal designs. (Adapted from M. M. Warrant and C. A. Ottinger, *Compilation of Current Literature on Seals, Closures, and Leakage for Radioactive Material Packagings*, SAND88-1015, Sandia National Laboratories, Albuquerque, N.M., January 1989).

factors affect the ability of a specific seal in a given configuration to seal and effectively prevent by-pass leakage. Some of these factors are not unique to the seal. On the other hand, the permeation rate for a seal with a given gas is uniquely related to the seal material. The permeation rate of a seal to a particular gas is usually important in selection of the containment-boundary test method and, more infrequently, to the selection of an inerting gas if air is to be excluded from the package cavity. Permeation is discussed also in Sect. 8.5.3.

Other design factors may be important to the operating capability of the seal. These factors include resiliency and compression set, radiation resistance, thermal expansion, shelf life, or cost and availability. Some of these factors are discussed below.

A comparison of characteristics of elastomeric, plastic, and metallic seals over a temperature range between 400°C to that of liquid nitrogen (–195°C) is provided in Table 8.2 (ref. 30). Even though most packages will never experience temperatures of those extremes in normal operations, temperatures could exceed 400°C under accident conditions.

8.5.1 Temperature Performance

The seal must be able to retain its containment capability throughout the temperature range it could experience under both normal transport and accident conditions. The low end of the temperature range is –40°C, the regulatory limit for operations. It assumes that the contents either generates no heat or generates insufficient heat to raise the temperature above this lower limit. Performance of a seal based on some minimum level of heating would require design assurance of that minimum level of heat. Thermal analysis of the package is used to determine the high temperature to which the seal will be subjected; the analysis considers the temperatures of the seal bearing and retainer surfaces. This temperature typically occurs sometime shortly after the end of the hypothetical fire-accident condition. The selected seal must retain its seal capability against internal and external pressures throughout this temperature range. To protect the seals during high-temperature transients, one design strategy is to either place the seals as far from the package external surfaces as possible or to install a thermal barrier between the seal and external surface.

Metallic seals, which exhibit good performance at high temperatures, are frequently used to meet the upper-temperature requirements. Nonmetallic seals usually have limited life at high temperatures, and few are rated above about 200°C. Metal seals can be used at temperatures of about 550°C for extended periods.
For perspective, the reported temperature range for some typical seal materials are

Silicone	–55 to +230°C
Neoprene	–45 to +150°C
Fluorocarbon	–25 to +260°C
321 Stainless steel	–40 to +260°C
Inconel (Alloy 600)	–40 to +425°C
Inconel (Alloy 750)	–40 to +980°C

Low-temperature ranges for metallic seals are frequently not specified, but they are usually in the cryogenic range (–185°C or less). The values given here for the elastomer seals are taken from Parker guide, *O-Ring Handbook*.[31] The values for the metal seals are taken from a Helicoflex document, *Metallic O-Rings*.[32]

Table 8.2. Comparison of the characteristics of elastomeric, plastic, and metallic gaskets[a,b]

Gaskets	Compressibility	Elasticity	Permanent set	Hardness	Outgassing	Permeability	High-temperature behavior	Low-temperature behavior	Chemical behavior
Rubber	Incompressible; the seal should allow for constant volume deformation	Very elastic; the seal requires less pressure and maintains it	Small permanent set; groves or spacers required; gasket may be reused	Very soft; gasket does not require very high surface finish	High; the surface exposed to a vacuum should be kept to a minimum	Fair, except silicone rubber	Use limited to moderate temperatures; hardening accelerated by temperature	Becomes brittle; not recommended at low temperatures	Resistant to oils and mercury
Plastic	Compressible; higher local pressures require flanges with ridges indicated	Inelastic; the pressure of the seal must be often increased, bolts retightened or spring loading used	Cold flow occurs under sealing load; gaskets may be reused after annealing	Soft; the seal is based on parallel grooves and ridges on surface of flanges	Teflon has very low vapor pressure	Teflon has very low permeability	Fair, especially Teflon; high expansion; the seal should be tightened at lowest service temperature	Teflon can be used at low temperatures; seal should be tightened at low temperatures	Teflon is chemically inert
Metal	Compressible; very high pressures required; sealing surfaces kept to minimum	Equal torque on all bolts required	Hardens under load; gasket cannot be re-used many times	Harder than other gaskets; very good surface finish required	Very low vapor pressure even at high temperatures	Very low	Some metallic gaskets may be used for bakeable seals; seal should allow for differential expansion	May be used at low temperatures if differential expansion allowed for	Some oxidize; others may be attacked by mercury vapor

[a]Summarized from ref. 30.
[b]Considering temperatures ranging from that of liquid nitrogen to about 400°C.

Note that temperature ranges published for some materials can vary considerably among manufacturers because of differences in chemical makeup and fabrication processes. The designer should, therefore, refer to the manufacturer's specifications to obtain information about a specific seal. Further, if the seal is to be used near the limits of its reported range, the designer may wish to confirm its performance by testing. For example, SNL has reported that performance tests at low temperatures have shown that some seal materials did not perform at the low-temperature limits reported by the manufacturer.[33] However, the same study shows that the great majority of the seal materials tested do perform acceptably to –40°C.

Leakage of seals at low temperatures is caused primarily by thermal contraction of the elastomer material. The temperature at which leakage is initiated is a function of the O-ring material, and some research on Viton™ formulations and silicon show good low-temperature performance.[34] Seal compression set may also become more important at low temperatures.

8.5.2 Seal Resiliency, Deformation, and Compression Set

Sealing is normally achieved by compression of the seal between two surfaces, with the compression accompanied by some level of deformation of the seal. Typically, compression of the seal is on the order of 25% for elastomer seals and 15% for metal seals, where compression refers to the decrease in seal cross section between the mating flanges. However, specific seals have different compression requirements; such requirements can best be determined from literature published by the manufacturer. The compression requirement is typically established by the seal-groove dimensions recommended by the manufacturer. For example, for an O-ring seal having a cross-section diameter of 0.625 cm (0.250 in.), the groove depth may be specified as 0.51 cm (0.200 in.), thus establishing a compression of 20%.

The deformation of the seal gives rise to the issue of the amount of deformation that remains after the load is removed from the seal—its compression set. Ideally, the seal has sufficient resiliency so that if there is no compression set, the seal returns to its original dimensions as the compressive force is removed. In fact, the seal usually retains some small percentage of compression. SNL has tested elastomeric seals to assess the amount of compression set that exists after about 72 h following a compression of 25%. Most of the seal materials tested showed a compression set of about 1%. An exception were two Teflon™ compounds, which showed a compression set of between 5 and 10% (ref. 35). The designer should note that seal resiliency may be lost at low temperatures, and compression may be "set" until the seal reaches a higher temperature even if the compressive force is removed.

Compression set is important for two reasons: the first reason is that a hypothetical accident condition might cause a penetration flange to flex under dynamic loading conditions. If the flexing is larger than the compression set, then a transitory breach of the containment boundary could occur. However, the amount of resiliency required can be determined from the structural analysis and is seldom more than a fraction of a millimeter (a few thousandths of an inch), and, as a result, most seal materials can be expected to recover sufficiently to maintain the seal during this transient. Usually, the designer should avoid the potential for flexing of a penetration cover during any accident transient. Otherwise, the regulator could suggest that dynamic load testing be performed to confirm independently the vendor's estimates of seal resiliency. Note that metallic seals usually are compressed less than elastomeric seals, but the metallic seals retain somewhat greater compression set, sometimes as much as 5 to 7%. Usually, compression set is greater because of deformation of the seal by bending, which allows less return to the original shape.

The designer should pay close attention to the amount of bolt torque required for seal compression. Metallic seals are designed to be compressed less, but they still require substantial bolt torque to achieve the seal. For example, a 0.0625-cm (0.250-in.)-thick, 50-cm (20-in.)-diam type, 321 stainless steel O-ring requires a seal loading of 140×10^3 N/m (800 lb/in.) of diam.[32] It can be very difficult to apply torque greater than about 815 N · m (600 ft-lb) in the field, especially if temporary scaffolding must be erected to access the top of the package.

Temporary scaffolding is usually required if the top of the package (closure-bolt location) is greater than about 1.2 m (48 in.) above the floor. Few user facilities have recesses (to "lower" the package work height), or solid work platforms (to "raise" the height of the operator) that allow convenient access to the top of a specific package. Temporary structures make the manual application of high levels of bolt torque difficult and, frequently, dangerous.

A second aspect of compression set is the prospect of reuse of installed seals in operations. Seal materials that have good resiliency lower operating costs because the seals can be used more than once, and the seal is more forgiving of minor defects (scratches or nicks) in the sealing surface. Radiation exposure to personnel and turnaround time are also reduced if seals do not have to be replaced. In general, the designer should plan and design for reuse of seals with seal replacement occurring annually or on an as-needed basis.

Based on experience, reuse of metal seals may create a problem. For example, a spent-fuel-truck cask uses a metal O-ring seal for primary containment. This seal is replaced on a per-use basis even though the designer initially anticipated replacement after about ten uses. Per-use replacement evolved over some years of experience after it was found that the seal would occasionally leak after several uses. Replacement of the seal requires that the cask be returned to the spent fuel pool and the closure lid be removed for seal replacement. Consequently, it was found to be operationally expedient to replace the seal after each use, which markedly reduced the probability of failure and simplified the loading procedure and records keeping.[36] In this case, the difference between the planned replacement of the O-ring and the actual replacement schedule can be attributed to a compression set of the metal O-ring.

8.5.3. Permeability

As previously described, permeation is the diffusion of gas molecules through the molecular structure of the seal material. As the density of the seal material increases, the rate of diffusion decreases, and as the gas becomes (molecularly) lighter, the rate of diffusion increases. Other factors that are important to the rate of diffusion are the differential pressure across the seal, the temperature, and the face cross section of the seal. Because compression of a seal effectively increases its density, compression acts to slow permeation.

Permeation can be important to the containment of some few radioactive gases such as krypton, xenon, and perhaps tritium, but it is not a mechanism by which solids or liquids entrained in the diffusing gas escape the containment boundary. An important permeation concern is the diffusion of helium across the seal because helium is frequently used to confirm the presence and performance of a seal before the package is released for off-site transport. Helium is one of the few gases that can be detected in very low concentrations and for which detection equipment is commonly available. If helium permeation of the seal occurs before the leak test can be completed, the leak test will fail because the detector cannot distinguish between permeation leakage and by-pass leakage.

Tests performed by SNL on the permeation of elastomeric seal material have shown that most formulations of silicone, fluorosilicone, and Teflon permeate within minutes.[29] Use of these seal materials may pose problems in future leak testing if helium must be used to demonstrate containment. Other materials, including some elastomers and many coated metals, have permeation times that are so long as to be of little concern.[33]

To overcome the effects of helium permeation in testing, the designer has at least two options other than the use of an effectively impermeable seal material: The first option is to estimate the time required to perform the leak tests and select a seal material that has a permeation time of about twice the estimated testing time. In estimating the test time, the designer must consider that normally the required leak tests of the closure and all penetrations are usually done sequentially.

The second option is to select a seal material that has a permeation rate somewhat below the allowable leak rate. The test procedure specifies that a leak rate may be observed but that the observed leak rate must be less than one-half the permitted leak rate. Both leak rates should be specified in the procedure. Use of this option suggests that the seal material needs to be tested to establish its actual permeation rate under typical test conditions rather than to rely only on published data.

8.5.4 Other Factors

Several other factors relative to the seal material may be important to specific applications. These include radiation resistance, thermal expansion, shelf life, or cost and availability.

Radiation resistance refers to the ability of the seal to resist significant degradation when exposed to gamma and neutron environments. Such degradation typically results in changes in the mechanical properties of the seal. Elastomeric seal performance may be expected to degrade after about 10^4 Sv (10^6 rad) of radiation at about room temperature.[31] Metal seals are largely unaffected by radiation up to about 10^7 Sv (10^9 rads) or neutron flux rates greater than 10^{19} neutrons/s. Seal-material resistance to radiation is usually not important in the design of most transport packages. However, it may be an issue in packages used for long-term storage.

Thermal expansion of the seal material may need to be addressed in some applications. The coefficient of thermal expansion of elastomeric seals is normally about ten times that of steel. A higher coefficient contributes to greater sealing stress at elevated temperatures. If thermal expansion of a seal (e.g., a metallic seal) is less than that of the steel, then sealing stress is reduced, and some potential for loss of containment at higher temperatures could exist.

For the convenience of operations personnel, the seal material selected should have a reasonable shelf life. A shelf life of at least 2 years is desirable because it assists in ordering seals in less expensive lot sizes and in accommodating delay within the procurement or Quality Assurance systems and scheduled shipping programs.

Like shelf-life, seal cost and availability are future operations and maintenance concerns. When practicable, off-the-shelf seals using common materials are preferred to seals using special material formulations or having patented designs. The identification of a unique or proprietary seal material can result in the (at least temporary) loss of the package for use if the vendor terminates the manufacture of that particular seal, if some flaw is later found in the manufacturing process of the material, or if that particular seal becomes unavailable for some other reason. Frequently, it is

helpful to consider materials having a military specification (MilSpec) because there is typically more than one vendor of the material. Cost of MilSpec material may be noticeably higher because of the detailed specifications the material must sometimes meet.

8.5.5 Sources of Information

Several sources of information exist regarding seal materials, performance, and technology that are readily available to the designer. The principal sources are the existing seal vendors and fabricators, existing package designers and operators, and technology development and demonstrations programs, such as those conducted by SNL.

Existing vendors and fabricators can be located by referring to industry "yellow pages," such as the *Thomas Register*.[37] The designer may also review directories of certified packages such as NUREG-0383,[38] consult other package designers, or consult industry publications, and research papers or reports of studies conducted by the national laboratories. One comprehensive study is the *Compilation of Current Literature on Seals, Closures and Leakage for Radioactive Material Packagings* (ref. 30). This study provides some insight into many of the issues related to seal selection and presents some tables of properties and comparisons of the family of seals currently in use. Ongoing research programs at SNL, primarily in support of the DOE Office of Civilian Waste Management, are investigating seal performance in several areas, but primarily in low-temperature performance and permeation.[30,33,35,39]

At some point the designer will select a seal material and configuration from a qualified vendor. Frequently, a seal vendor's engineering department provides test results, materials and fabrication data, performance data, and other information to assist the designer in ensuring the correct application and configuration of a given seal. The information provided includes the temperature-performance range, pressure range, material compatibility, wear and shelf-life data, recommended sealing surface smoothness, and groove width and height. Even though some seal materials have limitations on the diameter of the seal, most seals can be fabricated in virtually any diameter required, but, as previously noted, designers are best advised to specify seal diameters that the manufacturer offers "over the counter" or "off the shelf."

8.6 ASSEMBLY VERIFICATION AND PERIODIC INSPECTION

This section addresses the containment-boundary testing, which is required for the continued use of the package. Typically, containment testing is performed on a per-use basis and periodically, usually annually, to verify continued compliance of the package to its performance requirements. A significant portion of the time for each package handling is devoted to establishing or verifying containment. To the extent that the designer simplifies this process, corresponding reductions in package-handling time and operator radiation dose can be achieved.

8.6.1 Operational Design

In a typical package design, the cavity is penetrated by one or more openings intended for venting, filling, sampling, and draining the cavity in addition to the opening for loading and unloading the package. Each of these penetrations becomes part of the containment boundary and must be closed to establish the leak tightness required by the regulations. Depending upon the contents, this

requirement means that each of the penetrations that are used to carry out operating functions must be leak tested when the package is initially accepted, annually, and at each use (per use).

The number of individual leak tests required may be reduced by design. For example, more than one penetration may be grouped under a single closure, with only the closure leak tested. The design may allow either the vent or drain to be used for sampling cavity gases to eliminate a sample port, or the test ports for more than one penetration can be interconnected so that only one leak test is performed.

The containment design should ensure that the boundary closure and penetration seals can be adequately and efficiently tested. A significant portion of the package-handling time will be spent in the operation (per-use) verification of containment and in the similar determinations made during periodic inspections (annuals). The test procedure should account for the fact that (1) leak testing is normally performed in a radiation environment on a package that may contain significant sources of internal, residual radiation; (2) personnel may be encumbered by anticontamination clothing, gloves, and respirator gear; (3) uncomplicated test regimes can assist in reducing the potential for operator error and reduce lost time due to failed required devices; and (4) simplified and unsophisticated test regimes can shorten handling times, reducing operator total radiation exposure. Careful consideration of the testing process is justified because the package is designed once, but it may be used hundreds of times.

The containment boundary is generally established using either a single-seal or a double-seal design at the penetrations, including the closure. Single-seal designs are tested usually on a per-use basis by bubble, pressure-drop, or pressure-rise testing and periodically by detection of some tracer gas, frequently helium. Pressure-drop testing is done by pressurizing the package cavity and observing for a pressure drop over some period of time using a pressure gauge attached at a penetration. The test is repeated for each penetration. This test is an assembly verification for the closure and the penetrations. Once this test is complete, the cavity is vented to a waste-gas-handling system. At the periodic inspection, the cavity is pressurized with the tracer gas, and a detector is moved slowly around the outside of the containment boundary seals in search of evidence of leaks. The penetrations and closure are tested one at a time.

The double-seal design uses two seals at each penetration (including the closure) which are concentric. The seal closest to the cavity is the containment-boundary seal. This seal and the outer seal form an annulus, the interspace, which can be sampled for a tracer gas or used for pressure tests, as shown in Fig. 8.3.

The test port to the interspace is outside of the containment boundary and is typically closed by a pipe plug or similar fitting. Even though the outer seal contributes to the ultimate leak tightness of the package, no credit is taken for its presence in the containment-performance assessment.

This configuration is also frequently tested by the pressure-drop method on a per-use basis and by detection of a tracer gas on a periodic basis. Pressure-drop testing is carried out by pressurizing the interspace and observing for a pressure drop over some period of time using a pressure gauge attached to the fixture at the test port. A similar concentric seal set and test port is used at each penetration, and each penetration is tested separately (although careful design can result in the interconnection of two interspace annuli, allowing the simultaneous testing of two penetrations). Periodic testing is accomplished by pressurizing the package cavity with a tracer gas, connecting the detector to the test port and observing for indications of the presence of the tracer. This

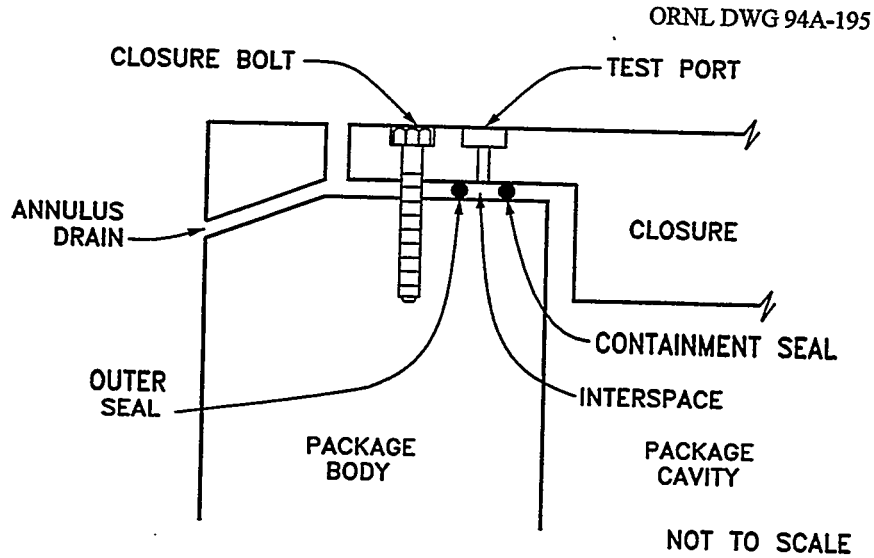

Fig. 8.3. Concentric seal-closure containment boundary.

connection is repeated at any untested penetration. At the conclusion of the test, the tracer gas is vented to the waste-gas-handling system.

The use of the two-concentric seal design has proved to be more operationally convenient than testing a single-seal configuration, and single-point testing has the effect of "totalizing" leakage in a given seal, which is more consistent with the spirit of the Standard. When compared with single-point testing, the circumferential testing required by single-seal designs tends to extend testing time and increase operator dose, but it does not lead to any improvement in confidence in leak tightness. Pressurizing the cavity on a per-use basis requires a (sometimes large) volume of gas to be vented from the package and processed through a facility waste-handling system, thus increasing the facility waste handling requirement.

During normal operation of the package, the O-rings or gaskets may fail to seal properly. This problem can have several causes, including sealing surface damage or the accidental introduction of debris on the sealing surface during handling operations. Any small nicks, gouges, or other flaws that are introduced into the sealing surfaces are often polished out using jeweler's rouge or polishing stones.

A disadvantage of the two-concentric-seal configuration is that a pressure-drop-test failure means that both seals must be replaced because the operator has no way to determine which seal is leaking. A similar condition exists if two (or more) seal tests were conducted simultaneously. For example, if the closure and one other penetration interspace volumes were interconnected and a leak was detected, the operator would replace the penetration seal and inspect its seal surface first. The leak test would be repeated before the closure seal was replaced or inspected because that job requires substantially more time and effort.

The frequency of failed leak tests is low based on experience with spent fuel casks.[36–40] The excellent experience stems primarily from the fact that the sealing system is simple and that there are very few failure mechanisms. Most failures result from debris or minor scratches on the sealing surface or on the seal that occurs during underwater loading. However, scratches can result from carelessly removing a seal from a groove, resting a penetration on its sealing surface, and

placing other items (bolts, screwdrivers, wrenches, etc.) on a penetration seal or sealing surface. To reduce the possibility of operations personnel damaging the sealing surface during removal, the designer should provide a slot that will accept the tip of a flat-bladed screwdriver in the O-ring seal groove. The slot provides a point at which the O-ring can be pried out of the groove to allow replacement. Without a slot or other aid, it can prove to be difficult to extract the O-ring, without damaging the sealing surface or groove.

For the most part, seal groves are placed in the "removable portion" of the penetration or closure (e.g., the lid) rather than in the body of the package. The reason for this is that in the event of seal failure, it is usually only the lid or cover that has to be removed to replace them. Suppose, for example, that a spent fuel package that is loaded underwater fails during a leak test. The operator would return the loaded package to the pool and, if the seals were in the closure lid, would pull it to the surface, replace the seals, and return the lid to the package. If, however, the seals had been mounted in the body and the package failed the leak test, the operator would have to return the package to the pool and unload it before it could be brought to the surface for seal replacement.

The designer should ensure that some method of holding the seal in its place exists if the seal is on the underside of a closure lid. Such methods include a press fit, clips, inverted V-shaped grooves, etc. Without some positive means of holding the seal in place, it is possible that the seal could be dislodged when moving the closure lid to the package and the seal lost in the bottom of a loading pool or hot cell.

The placement of grooves in penetration covers is affected by two considerations: The first is that in the event of a failed leak test, the cover plate can be removed from the package and taken to a lower radiation area for seal replacement and surface inspection. The second is that in the event of significant damage to the seal surface or seal grooves, the entire cover plate can be replaced. Reworking the package body to complete repairs can be difficult and certainly can be more time-consuming than simply dealing with a cover plate.

As has been discussed in other sections, the bolt torque on the closure bolts should be limited to what can reasonably be achieved by manual application [e.g., up to 815 N · m (600 ft-lb)]. Higher torques can be difficult to apply under the operating conditions frequently found in the field. Penetrations should be placed near the edge of the package, preferably in the package side wall if the package stands more than about 1.2 m (48 in.) tall. Edge and side-wall placement allow the operator to work at arms/length, thus reducing whole-body exposure, and frequently results in a better view of the penetration. The penetration configuration should be designed to reduce or to eliminate the potential for operator error during package operations.

Finally, the seals are a routinely replaceable component of the package containment boundary. The designer may establish a replacement schedule based on any number of factors, including seal material, configuration, contents parameters, expected rigor of use, and other considerations. The Standard does not require that more than an assembly verification test be performed if a routinely replaceable item is replaced during use of the package. Consequently, provided that a written procedure is followed and that inspected and approved seals are used, an assembly verification test is normally sufficient to demonstrate containment if only the seal is replaced.[41]

8.6.2 Assembly Verification

The designer must specify a per-use leak test for most contents proposed for the package. The per-use test is permitted by the Standard to be less sensitive than the leak test used during the periodic inspection and is basically a "form and fit" or assembly verification. The philosophy is that the package has been assembled according to a written procedure in the same configuration that has routinely passed the much more stringent periodic leak test. Thus, there is no reason to suspect that this assembled package would not also pass the periodic test if it were applied. Clearly, the procedure used to establish the containment boundary on a per-use basis must be the same procedure used to establish the boundary for the periodic leak test.

The designer may also need to specify a "form and fit" leak test for shipping the unloaded package if the package is going to have any significant residual contamination in the cavity after the authorized contents are unloaded. (The package is *unloaded*, but it is not necessarily *empty*.) Residual contamination is considerably more of a problem for shipments that are loaded in storage pools than for packages that are handled dry. It is usually not necessary to specify a leak test for the unloaded package if (1) there is no significant residual contamination, (2) the contents are separately containerized, or (3) the method of package handling essentially precludes generating residual contamination in the cavity.

Less-sensitive tests are generally considered to have an operational benefit because the test procedure and equipment used are usually less complex. For example, if it can be shown that a pressure-drop test to an interspace seal configuration can be used for assembly verification, then the small interspace can be pressurized using facility air pressure, and the pressure can be measured over some reasonable period of time (e.g., 10 min) using a facility-calibrated pressure gauge. The instruments (gauge and clock) are common and unsophisticated and can be read directly. Recovery from failure of an instrument is easily accomplished, and success or failure of the test can be determined by personnel with routine facility training.

The reduced sensitivity of the assembly verification test is described in Sect. 7.6 of the Standard. This criteria provides some flexibility in establishing the procedures to test the package and allows the designer to apply a test that is less stringent than the periodic test. However, during the application of the per-use testing, the package must always meet the per-use test requirement without evidence of a leak greater than that permitted by the contents.[42] For example, suppose that a package has an annual leak test requirement of 1×10^{-6} std cm^3/s, a per-use test requirement of 2×10^{-3} std cm^3/s, and a per-use test procedure that can measure 1×10^{-5} std cm^3/s. If during the per-use test, a leak rate of 1×10^{-4} std cm^3/s were measured, the package would be assumed to have failed its leak-rate test. By referring to the philosophy presented in the first paragraph of this section, it can be seen why this assumption is proper. The package has not demonstrated "form and fit," and corrective action must be taken to ensure that the package meets its containment performance requirement.

As an example of a form-and-fit test, assume that a package has a concentric O-ring containment boundary, as described in Sect. 8.6.1. From the package drawings, the designer finds that the annulus formed between the two O-rings is machined to be 0.406 mm (0.016 in.) in height, the outboard radius of the inner O-ring is 358 mm (14.075 in.), and the inboard radius of the outer O-ring is 368 mm (14.495 in.). (Note that the annulus height is machined in the closure lid between the O-ring grooves. The O-rings are fully compressed, and the closure and the package body are otherwise in metal-to-metal contact.)

A typical concentric O-ring configuration is shown in Fig. 8.4. In this figure, metal-to-metal contact of the penetration closure with its base occurs inside of the inner O-ring and at the outside of the outer O-ring. The machined surface between the O-ring grooves creates the test annulus. As shown in the figure, a leak-test port is drilled through the closure to intercept the annulus.

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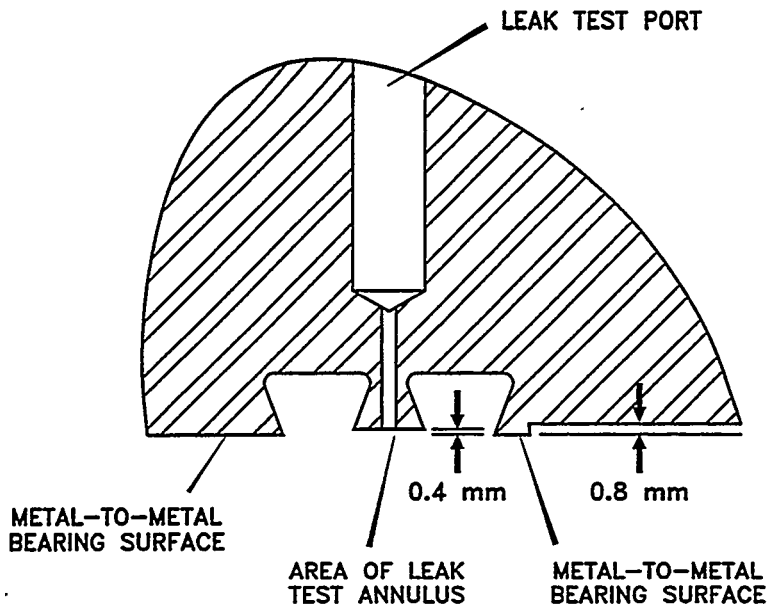

Fig. 8.4. Typical concentric O-ring configuration.

The volume of the annulus can be found from:

$$V = \pi r^2 h \,, \tag{8.11}$$

where h is the annulus height and r is the difference between the square of the inboard and outboard radii, or

$$V = \pi[(368)^2 - (358)^2]\ (0.406) = 9{,}270\ \text{mm}^3 = 9.27\ \text{cm}^3 \,.$$

The leak rate, based on pressure drop, can be found from an application of the ideal gas law:

$$L = VT\ (P_1/T_1 - P_2/T_2)/t \,, \tag{8.12}$$

where

V = the volume of the test annulus, cm^3;
T = a reference temperature, 298 K;
P_1 and P_2 = the pressures at the end and start of the test, respectively, atm;
T_1 and T_2 = the temperatures at the end and start of the test, respectively;
t = the duration of the test, s.

The procedure for the leak test would require that no pressure drop be observed during the time of the test. In performing the test, if the measured pressure at the start of the test (P_2) is equal to

the pressure at the end of the test (P_1), then from the equation (and intuitively), the leak rate (L) is equal to 0.0 cm^3/s.

To explore the sensitivity of the pressure drop test, assume that the operator could observe a pressure drop of 6.9 kPa (1 psi) on a gauge and set the test pressure at 1.5 times the normal transport condition (gauge) pressure. Using the pressure from the example in Sect. 8.3.3 of 220 kPa (17.2 psig or 34.9 psia), the test pressure can be established as 274 kPa (25 psig or 39.7 psia).

As determined from Eq. (8.12), the leakage rate that can be measured decreases as the duration of the test becomes longer. Operationally, a convenient duration is about 10 min (600 s). Longer test times can lead to temperature variations in the test volume, which, in turn, will affect the pressure.

$$L = [(9.27)(298)\ (2.7/298 - 2.63/298)]/(600)$$

$$L = 9.2 \times 10^{-4}\ \text{atm cm}^3/\text{s}\ .$$

Referring to Sect. 7.6.4 of the Standard, testing to a sensitivity of 1×10^{-3} std cm^3/s is adequate for a form-and-fit test. The designer should note that, technically, the form-and-fit test should be corrected to standard conditions.

Several areas should be revisited in this example. The first is that it works only because the test volume is (very) small. As the test volume increases, the sensitivity of this method decreases. The second is that the proposed test requires that the calibrated gauge used has adequate divisions of the scale to allow the user to properly determine pressure of the system and its variations. Third, no loss of pressure in the actual test is permitted, even though a loss of pressure is assumed in calculation. Any loss of pressure must be taken as indicative of containment failure. Finally, the test checks both the inner and outer O-rings; if a leak is detected, the operator cannot know which O-ring is actually leaking, so both must be replaced (and seal surfaces checked).

In spite of the argument that a less-sensitive form-and-fit leak test is acceptable for per-use shipments, testing identical to the acceptance level or annual inspection testing can also be used for per-use testing. Testing to the more sensitive level may be appropriate to dedicated packages that are used between two facilities and that have the test equipment installed in the package-handling area. However, as the required tests increase in sensitivity, both the test procedure and test equipment tend to become more complicated, which may require special equipment, training, and perhaps additional specially qualified personnel.

8.6.3 Periodic Inspections

Periodic inspections of a package are performed to ensure that it has not degraded during use and that it continues to perform as described in its SARP. The frequency of the periodic inspection is often once per year, but the frequency is set by the designer and may be based on any number of design or use considerations. Periodic inspections are discussed in some detail in Chapter 14.

A key element in the periodic inspection is the package leak test. This test must be performed according to the procedures or methodology presented in Sect. 8.2 of the SARP. It may be possible to apply the same test procedure or method that was used in the package-postfabrication-

acceptance testing; however, the designer should recognize that at the time of the periodic inspection, the package may be externally contaminated and is likely to contain residual internal contamination. If the package is large, personnel are usually dressed for contamination protection and may need to wear respirators. If leak testing according to the original acceptance procedure will not work, the designer must develop a procedure appropriate for a contaminated package. The periodic test normally specifies the replacement of the seals used in the containment boundary and may specify the rebuilding, replacement, inspection, or testing of other boundary hardware.

The leak test that is performed must have sufficient sensitivity to demonstrate that the package meets its containment performance requirement. According to the Standard, the sensitivity of the leak test must be at least half the maximum allowable leak rate, or, from Eq. (B16) of the Standard,

$$S \leq L/2 \text{ cm}^3/\text{s} , \tag{8.13}$$

where S is the sensitivity of the test method.

Returning to the working example, the maximum allowable leakage rate (air) was found to be 2.2×10^{-7} cm³/s (from Sect. 8.3.3), from which then $S = 1.1 \times 10^{-7}$ cm³/s. If the containment criteria state that the package be leak tight, then the sensitivity of the leak test must be 5×10^{-8} std cm³/s or better [from Sect. B14 (ref. 43) of the Standard.] According to Table A.1 of the Standard, "Leakage Tests," the acceptable leak test methods for this level of sensitivity involve a sensing of helium as a tracer gas.

Detection of helium at low concentrations requires specialized test equipment, which may require special training or qualifications to operate or for interpretation of results. For the most part, the ability to direct-read the solid-state detectors currently available are within the skill level of the technicians normally working within nuclear facilities. However, the required skill level may be set by the package owner or by the facility operator in which the package is used.

To continue with the example, suppose that the designer proposes a periodic inspection (and acceptance test) procedure that requires that the package cavity be drawn to a vacuum, backfilled with helium to 203 kPa (2 atm), and the boundary seals be circumferentially tested using a mass spectrometer for detection of helium leaks. Such a procedure is consistent with the SARP, which specifies a leak-tight condition. The procedure is also consistent with the Standard, in which Table A.1 indicates that helium detection methods should be used.

Upon reviewing the leak-test procedure proposed by the designers, the operations personnel question the use of a mass spectrometer because it can be difficult to calibrate and requires the use of liquid nitrogen. They oppose establishing a vacuum in the cavity because mixing of the cavity gas wastestream and the vacuum pump oil could create a mixed waste. The operators propose that an existing procedure using halogen (gas) detection be investigated to use on the new package. The operations personnel advise that the halogen gas is fully recoverable at the (package) user facilities, so there is no environmental damage (threat to the atmospheric ozone layer) concern. The user, however, would not be able to dispose of any mixed waste that may be created by use of the vacuum pump.

The designer finds that the existing halogen test procedure calls for the package cavity to be initially closed in air at atmospheric pressure (101 kPa) and for the cavity then to be pressurized to 152 kPa (7.5 psig), using Freon-12™ refrigerant.

Because the detector senses only halogen gas, the allowable leak rate, L_R, must be converted to an equivalent leak of pure halogen, taking into account the air-halogen mixture in the cask during the leak test and the pressure differential between the leak test and the reference air leak specified by the Standard.

From Eq. (8.5),

$$L_R = (F_c + F_m)\ (P_u - P_d) \text{ std cm}^3\text{/s, (air)} ,$$

and

$$L_m = (F_c + F_m)\ (P_u - P_d) \text{ cm}^3\text{/s, for the air-halogen mixture} ,$$

where F_c = continuum flow, F_m = molecular flow, P_u = upstream pressure, and P_d = downstream pressure in atms.

To simplify the calculations, Eq. (8.9) is used to determine if one flow, F_c or F_m, of the reference air leak predominates, allowing the other flow to be set equal to zero.

From Eq. (8.9),

$$r_f = F_c/F_m = 654\ DP_a/[\mu(T/M)^{1/2}] ,$$

where

$\mu_{(air)} = 0.0185\ c_P$,
$M_{(air)} = 29$,
$T = 25°C = 298$ K,
$D = 3.53 \times 10^{-4}$ cm,
$P_a = 1.25$ atm,

from which $r_f = 3.0$. Because $r_f > 1$, then continuum flow dominates, but r_f is not large enough to allow an assumption that $F_m = 0$.

Even though the equations for L and L_m given above could be solved for the mixture flow, it is somewhat easier to solve the equations for the continuum flow (F_c), and then correct this flow to include the molecular flow, F_m.

The strategy that is applied is that the postulated reference air leak, L_R, and the postulated leak of the air-halogen mixture (say L_m), would be leaking through the same (postulated) hole. The only difference between the leaks is that the air and the air-halogen mixture have different viscosities. All other values (leak-pathway length and diameter, and the pressure difference) are constant because they are the same for either postulated leak.

From Eq. (8.5),

$$L = (F_c + F_m)\ (P_u - P_d) .$$

The molecular flow is assumed to be negligible, so $F_m = 0.0$.

Therefore,

$$L = F_c\,(P_u - P_d)\ . \tag{8.14}$$

But, from Eq. (8.6),

$$F_c = (2.49 \times 10^6)\ (D^4)/a\mu\ \text{cm}^3/\text{atm-s}\ ,$$

where

D = leak-hole diameter,
a = leak-hole length, and
μ = viscosity of the leak medium.

Substituting in Eq. (8.14),

$$L = [(2.49 \times 10^6)(D^4)(P_d - P_u)]/a\mu\ .$$

Because the leak-pathway parameters, and pressure difference are the same for these leak rates (the equivalent leak is leaking through the same hole), then these values can be replaced by a constant (e.g., K),

where

$$K = (2.49 \times 10^6)(D)^4\ (P_u - P_d)/a\ . \tag{8.15}$$

Therefore,

$$L_R = K/\mu_R\ ,\ \text{and}\ L_m = K/\mu_m\ , \tag{8.16}$$

where,

μ_R = viscosity of air, and
μ_m = viscosity of the mixture.

From Eq. (B8) of the Standard (see Eq. 8.17), the viscosity of the mixture, μ_m, is

$$\mu_m = \sum_{i=1}^{n} \frac{P_i \mu_i}{P_m}\ . \tag{8.17}$$

That is,

$$\mu_m = P_a\ \mu_a/P_t + P_r\ \mu_r/P_t\ , \tag{8.18}$$

where

a = air,
r = halogen refrigerant Freon 12™,

and where from the procedure, P_t = 152 kPa, of which P_a = 101 kPa and P_r = 51 kPa.

From Table B-1 of the Standard, $\mu_a = 0.0185\ c_P$, and $\mu_r = 0.0124\ c_P$. Therefore,

$$\mu_m = (101)(0.0185)/(152) + (51)(0.0124)/(152) = 0.0164\ c_P\ .$$

From Eq. (8.16), setting $K = K$, yields

$$L_m = L_R\ (\mu_R/\mu_m) = (4.90 \times 10^{-6})\ (0.0185/0.0164)\ ; \qquad (8.19)$$
$$L_m = 5.53 \times 10^{-6}\ \text{cm}^3/\text{s}\ .$$

This leak rate must be corrected to include the molecular flow component, which was previously ignored. Because $r_f = 3.0$, the calculated flow should be increased by about 1/3 to include the molecular flow component.

Therefore,

$$L_m = 5.53 \times 10^{-6} \times 1.33 = 7.35 \times 10^{-6}\ \text{cm}^3/\text{s}\ . \qquad (8.20)$$

The leak rate must now be corrected because the detector senses only the halogen component of the leak-test air-halogen mixture and because the standard leak rate assumes a pressure differential of 1 atm.

The halogen gas component of the leak is calculated as the ratio of pressures.

P_r/P_t (from Dalton's Law) = 51/152 = 0.336 ;

that is, the halogen detector senses only 33.6% of the actual leak rate.

The pressure correction is calculated as a ratio of pressures above atmospheric. Because the designer finds that the existing halogen-test procedure calls for the package cavity to be initially closed in air (101 kPa) and for the cavity to be then pressurized to 152 kPa (7.5 psig), using Freon-12 refrigerant, but the standard leak rate assumes a 1-atm pressure difference. Using the Perfect Gas Law (see Sect. 8.3.3),

$$P_{\text{Test Condition}}/P_{\text{Stand. Condition}} = (152 - 101)/(101 - 0) = 0.5\ . \qquad (8.21)$$

Correcting Eq. (8.20) for these two conditions yields a leak rate, L_{R12}, of

$$L_{R12} = (7.35 \times 10^{-6})\ (0.336)\ (0.5) = 1.23 \times 10^{-6}\ \text{cm}^3/\text{s}\ .$$

This is the Freon™ leak rate that is equivalent to the reference air-leak rate. The following question remains: Can the detector that is used measure this leak rate? If so, the operations recommendation could likely be accepted.

The operations personnel advise that the halogen leak test uses a detector having a sensitivity of 0.5-(fluid) oz/year, equivalent to approximately 14 cm^3/year. Converting,

$$D_S = 14\ cm^3/year \times year/3.15 \times 10^7\ s\ , \qquad (8.22)$$
$$D_S = 4.45 \times 10^{-7}\ cm^3/s\ .$$

According to Eq. (8.13), the minimum usable response of the detector is twice this value; therefore,

$$D_T = D_S \times 2 = 8.9 \times 10^{-7}\ cm^3/s \text{ (of Freon-12 refrigerant)}\ .$$

From the previous, the test leak rate was found to be

$$L_{R12} = 1.23 \times 10^{-6}\ cm^3/s\ .$$

Comparing D_T and L_{R12}, we find that $L_{R12} > D_T$ and the halogen leak test could be used for periodic testing of the package. If desired, this test would also be suitable for per-use testing, but usually halogen-based systems pose environmental safety questions that are increasingly difficult to answer. The objections raised to the designers' proposed procedure are valid, and diaphragm vacuum pumps that do not mix the discharge stream and the compressor oil should be considered. Solid-state helium detectors rather than mass-spectrometry systems should also be considered. Detection at the atomic level, which can be done with a mass spectrometer, is frequently not necessary.

8.7 MISCELLANEOUS CONSIDERATIONS

This section briefly addresses some of the issues that may arise in considering specific proposed contents or package design.

8.7.1 Confinement and Special-Form Containment

Occasionally, the package designer may find that the allowable leak rate for the proposed contents is so high that no testable containment boundary for the contents is required by the Standard. In this case, the designer should present the analysis demonstrating the condition and need not describe a leak test as a step in the handling procedures for the package described in Chapter 7 of the SARP.

Alternatively, the contents may qualify (or be qualified) as "special form." Special form ensures nondispersibility of the radioactive contents. If the contents are in a special form (e.g., in a capsule), the capsule must meet the criteria described in 10 CFR Part 71.75 and 10 CFR Part 71.77. The package need not meet a containment requirement because the requirements for special-form material ensure that the leakage of the encapsulated or solid material (such as irradiated hardware) does not occur under hypothetical accident conditions. Also note that there is no containment requirement for items qualified as "sealed sources." Sealed sources are typically designed to meet either the special-form requirements or contain substantially less than an A_2 quantity of material. In fact, most contain less than a "Limited Quantity." Usually, greater than Type A quantities of material that qualify as special form can be shipped in a Type A package.

The usual purpose of the package when only confinement is required is to provide shielding. The package internal configuration must ensure that the contents do not move so that shielding, criticality, or other performance requirements are not compromised.

Even if containment testing is not required, a well-designed package should contain a seal at the package closure and penetrations to prevent the inleakage of rainwater or other incidental water and to provide a barrier against the migration of residual contamination in the package or on the surface of the contents. In addition, seals may be needed for pressure or other tests that may be required at postfabrication acceptance or periodically that demonstrate other performance aspects of the package.

8.7.2 Combustible Gas

The quantity of combustible gas (hydrogen) generated by the contents during transport must be limited to less than 5% H_2 by volume. The major source of this gas is the radiolysis of hydrogenous materials. Other sources include desorption of protium (H+) in the contents by radiation or desorption of hydrogen isotopes in decomposed hydrides. Control of hydrogen generation also may be required to protect some package material from possible hydrogen embrittlement.

The designer should consider the generation rate of combustible gases in the contents to ensure that the volumetric limits are not exceeded. The principal tools for estimating combustible gas quantities are the computer codes WASTEH2,[44] and the more recent RADCALC.[45] The designer should note that these codes use a relatively small set of radionuclides in the calculational data base and that the codes do not account for the creation of decay products (from the original assumed contents) that could be more active in radiolysis.

The principal method of controlling the generation of combustible gases is to ensure the removal of water and to inert the package cavity. Removal of the water removes the significant portion of the hydrogenous material and inerting limits the oxygen concentration within the cavity. Water removal must consider the free-standing water and water that is entrained or otherwise captured in the contents. The performance of previously loaded secondary containers must also be considered.

Depending on package type, it may also be possible to use a catalytic recombiner or other similar device to suppress the formation of combustible gas.

If these steps are not sufficient to prove that a combustible gas mixture will not exist in the package cavity, then limitation on the package contents may be required.

8.7.3 Metal Fuels

Metal fuels are a class of primarily highly enriched research reactor fuels fabricated from uranium alloys. ORIGEN analysis of the irradiated fuel can show significant quantities of fission gases that are available for release. For conservatism, the designer can assume that these gases are immediately released by catastrophic fracture of the fuel in the hypothetical accident event. Fission gases that are generated in the metal fuel are formed within the matrix of the metal and migrate only very slowly to the surface of the metal where they might be released. Because the metal fuel

is designed to have at least self-supporting structural integrity, fracture to powder in any event is not likely. A study conducted by Oak Ridge National Laboratory (ORNL-2616),[46] showed that release rates of fission gases are generally less than 1% of the calculated inventory unless the fuel is melted.

Based on the available data, the designer may find that there is not significant releasable material in either normal conditions of transport or the accident conditions, and no requirement to calculate an A_2 value. If so, the rigor of any required operational or periodic leak test may be reduced, thus avoiding a requirement to use sensitive test instruments in the operating environment. Pressure testing of the closure may prove adequate.

8.7.4 Plutonium

Packages intended for the shipment of over 20 Ci of plutonium must provide for separate containment in addition to the containment provided by the package. Each of the containment boundaries (inner container and outer package) must be leak tight to $10^{-6}A_2$ Ci/h in normal transport. In the hypothetical accident conditions, they must both satisfy the release limit of A_2 curies per week (see 10 CFR Part 71.63). However, irradiated reactor fuels and metal alloys containing plutonium in solid form are exempt from the from the double-containment requirement (see 10 CFR Part 71.63.b).

If the amount of plutonium in a shipment exceeds 20 Ci, it must be shipped as a solid.

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9. Shielding

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9.1 INTRODUCTION

The shielding for a transportation package must be designed to maintain radiation dose rates external to package surfaces below established regulatory limits under defined normal and accident conditions. These regulations, discussed in Sect. 9.2, can vary depending upon whether the package is transported with other goods in general freight or whether it has exclusive use of the vehicle that transports it. Two forms of radiation are of most concern in package design: gammas and neutrons. Gamma radiation requires dense material for efficient shielding (e.g., lead, steel, and depleted uranium). Neutrons, when present, require a light material often containing significant quantities of hydrogen in order to shield the radioactive material. Each type of radiation requires somewhat different techniques to determine the proper shielding thicknesses to reduce external dose rates to acceptable limits. This chapter discusses how the radiation source can be characterized (Sect. 9.3); analysis methods that may result in a preliminary package design (Sect. 9.4); and, finally, calculational techniques that may be applied to a package design in order to predict external dose rates (Sect. 9.5).

Even though this chapter discusses shielding topics that will be applicable to any package design, one of the more challenging problems is the shield design for a spent fuel cask. The exposure of such fuel to the neutron flux in an operating reactor over a several-year period (with possible movement of the fuel assemblies within the core) can be much more difficult to characterize than a single isotope whose radiation characteristics are well known. In addition, for fuels having burnups in excess of about 20,000 MWd/t, neutrons become a significant source of radiation in addition to the gamma rays. It is for these reasons that this chapter focuses on shielding problems associated with packages designed to carry spent nuclear fuel (SNF).

In order to develop a source term for a package designed to carry SNF, a target maximum burnup and minimum cooling time (age) for the fuel is generally specified; this is often augmented in the Safety Analysis Report for Packaging (SARP) with burnup/age curves defined to demonstrate the operational flexibility of the cask design under alternative loading conditions (i.e., greater than design burnup and/or less than design age). The cask shielding design process must interact with materials selection and structural analyses efforts since the design goals usually focus on minimization of cask weight while maximizing cask payload. Both goals affect the design of the cask body, which remains the primary structural component of the package and whose integrity

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must be assured even under severe accident conditions. In addition, the cask body must be capable of transferring decay heat away from the SNF (or other radiation sources) under normal operating conditions, as well as preventing excessive heating of the payload that might be caused by an external fire should the package be involved in a severe accident.

The primary focus of the interaction between the shielding and criticality design activities is the determination (based primarily on criticality concerns) of the number of assemblies to be carried and fixed neutron absorber sizes in the cask basket. These activities largely determine the sizing of the cavity and, hence, the inner dimensions of the cask body, the payload, and the total cask weight.

Modern trends toward high burnup in SNF and the corresponding high initial enrichments tend to complicate today's cask design efforts. In the past, only primary gamma radiation contributed appreciably to external dose rates. Under these conditions, simple hand calculations and point-kernel techniques proved to be quite useful tools. Now neutrons, while still low in a well-designed cask, must be properly accounted for. These trends, as well as rapid advances in personal computers, have driven the cask designer away from hand calculations toward more sophisticated-yet-simple cask design tools. The PC-based program, CAPSIZE, has been developed to simultaneously study material selection (depleted uranium, lead, and steel gamma shields are considered), shield optimization, payload determination, and dose estimation. The CAPSIZE methodology includes a number of approximations, but also allows for a number of preliminary trade-off and cask design decisions to be made.

Once the scoping work for preliminary cask designs has been completed, a number of one-dimensional (1-D) transport methods are available for more detailed analysis. The ANISN and XSDRNPM 1-D discrete-ordinates codes are widely used for preliminary, and sometimes final (depending on the application), design work. These codes can utilize SNF source terms (typically generated from point-depletion codes such as ORIGEN2 or ORIGEN-S), as well as a number of multigroup cross-section libraries, including the SCALE 27-neutron, 18-gamma group library; the CASK 22-neutron, 18-gamma group library; and the BUGLE-80 47-neutron, 20-gamma group library. A number of excellent two-dimensional (2-D) and three-dimensional (3-D) methods are available for detailed final design calculations should they be needed. These include the 2-D discrete-ordinates code DORT and the 3-D Monte Carlo codes MORSE and MCNP. These codes and data are referenced in Sect. 9.5.

These issues, as well as others, are discussed in more detail in the sections that follow.

9.2 APPLICABLE REGULATIONS AND REQUIREMENTS

This section summarizes the current U.S. regulatory policies for radiation dose limitation as mandated under Title 10 of the *Code of Federal Regulations*, Part 71 (10 CFR 71) and Title 49, Part 173 (49 CFR 173).

The U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Transportation (DOT) govern the enforcement of these regulations for packaging and transportation of radioactive materials. These regulations have evolved over the years along a somewhat independent path from the corresponding international regulations as recommended by the International Atomic Energy Agency (IAEA). In recent years, both national and international regulations in the area of

radioactive material transport have been converging such that only small differences currently exist.

Several basic definitions are necessary to achieve an understanding of the applicable regulations. A Type-A package is a transportable package that carries at most A_1 or A_2 curies of a specific material. A package carrying an activity more than A_1 or A_2 is labeled a Type-B package. The A_1 and A_2 values are given in Table A-1 in 10 CFR 71 as well as Table 1 in IAEA Safety Series 6.

Both package types are subjected to the 10 CFR 71.47 requirements that state:

> (a) Except as provided in paragraph (b) of this section, each package of radioactive materials offered for transportation must be designed and prepared for shipment so that under conditions normally incident to transportation the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, and the transport index does not exceed 10.
> (b) A package that exceeds the radiation level limits specified in paragraph (a) of this section must be transported by exclusive use shipment only, and the radiation levels for such shipment must not exceed the following during transportation:
> (1) 2 mSv/h (200 mrem/h) on the external surface of the package, unless the following conditions are met, in which case the limit is 10 mSv/h (1000 mrem/h);
> (i) The shipment is made in a closed transport vehicle;
> (ii) The package is secured within the vehicle so that its position remains fixed during transportation; and
> (iii) there are no loading or unloading operations between the beginning and end of the transportation;
> (2) 2 mSv/h (200 mrem/h) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vehicle planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle; and
> (3) 0.1 mSv/h (10 mrem)/h) at any point 2 meters (80 in.) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and
> (4) 0.02 mSv/h (2 mrem/h) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10 CFR 20.1502.
> (c) For shipments made under the provisions of paragraph (b) of this section, the shipper shall provide specific written instructions to the carrier for maintenance of the exclusive use shipment controls. The instructions must be included with the shipping paper information.
> (d) The written instructions required for exclusive use shipments must be sufficient so that, when followed, they will cause the carrier to avoid actions that will unnecessarily result in increased radiation levels or radiation exposures to transport workers or members of the general public.

Section 71.43(f) also applies to both Type-A and Type-B packages and states:

> A package must be designed, constructed, and prepared for shipment so that under the tests specified in paragraph 71.71 (Normal Conditions of Transport) there would be no loss or dispersal of radioactive contents, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging.

Additional requirements in 10 CFR 71.51 are placed on Type-B packages, which are required to be subjected to both normal condition tests (10 CFR 71.71) and hypothetical accident condition tests (10 CFR 71.73).

The IAEA regulations, with respect to radiation dose limitations, are presented in summary form in Table 9.1. These regulations are essentially identical to those in 10 CFR 71 with a couple of exceptions. The U.S. regulations state that as a result of normal condition tests, the package should experience "no significant increase in external radiation levels," whereas the IAEA regulations quantify this increase as no more than 20%. In addition, the IAEA regulations differentiate between so-called special-arrangement and non-special-arrangement packages for exclusive-use shipments.

Table 9.1. Summary of radiation dose rate limitation of IAEA regulation

	Exclusive use			Not exclusive use
	Special arrangement	Not special arrangement		
		By rail By road	By air By vessel	
Transport Index[a] (TI)	No limit			<10
Normal Transport (surface of package)	<10 mSv/h	<10 mSv/h	<2 mSv/h	<2 mSv/h
After Normal Condition Test (surface of package)	No more than 20% increase of radiation level			
After Accident Condition Test (at 1 m from surface)	<10 mSv/h			
Vehicle[b] (surface of vehicle) (at 2 m from surface)	<2 mSv/h <0.1 mSv/h		Not regulated	
Conveyance[c] (surface of conveyance) (at 2 m from surface)	(only in case of accumulation of packages) <2 mSv/h <0.1 mSv/h			

[a]TI means the maximum radiation level at a distance of 1 m from the external surface of the package in units of mrem/h. The conversion is 1 mSv = 100 mrem.

[b]Vehicle means a road vehicle, railroad car, or railway wagon.

[c]Conveyance means all transportation methods of packages.

9.3 CHARACTERIZATION OF THE RADIATION SOURCE FROM SNF

Equations used in particle transport theory contain a source term to simulate the emission of the particles (photons, neutrons, etc.) being transported. Once the isotopic concentrations for radioactive waste is calculated (or known), the basic *techniques* for generating the radiation sources do not usually depend on the waste type. Since the shielding computation typically covers a wide energy range, the sources must be cast in units of emission rate per unit volume as a function of energy. Methods of calculating these gamma and neutron source spectra are discussed in this section.

9.3.1 Gamma Source Strengths and Spectra

Isotopic concentrations generated by a depletion and decay code are sometimes used within the same code to determine, as a function of energy, the intensity of the gamma (or photon) rays emitted from the radioactive waste. Some depletion and decay codes rely on auxiliary codes to actually evaluate the gamma source strength and spectra. In either case, for completeness, five different types of photon production processes should be considered in generating gamma sources:

- gamma rays and X rays from radioactive decay,
- gamma rays from (α,n) reactions,
- prompt fission gamma rays from spontaneous fission,
- delayed gamma rays from spontaneous fission, and
- bremsstrahlung radiation from positron and electron deceleration.

The extent to which these photon production processes are considered in the creation of the gamma-ray source terms is important to a spent fuel shielding analyst. An additional source of secondary gamma-rays produced from neutron absorption is typically treated in the radiation transport analysis. The following discussion of analytic techniques for evaluating these processes is based on the procedures used in the ORIGEN2[1,2] and ORIGEN-S[3] codes. The primary variation likely to be seen between these procedures and that of other codes that include these production processes lies with the data employed.

The predominant part of the photon source spectrum from spent fuel is from the gamma rays produced by the radioactive decay of fission and activation products. The energy-dependent photon intensities for both the gamma rays and X rays need to be included in the photon database (a good data source is the Evaluated Nuclear Structure Data File or ENSDF[4,5]). Photon intensities, I, in the database and their associated energy, E_I, are typically converted to energy group intensities, G, by conserving energy. The following equation is applied for this adjustment:

$$G = I(E_I / E_G) \ , \tag{9.1}$$

where E_G is the arithmetic average of the energy group boundaries. These procedures maintain the conservation of energy rather than photon intensity and provide a more accurate computation of dose rates in multigroup shielding analyses. Caution is advised for thick-shield applications since even the application of energy conservation can produce large errors if the group structure is too broad (see Sect. 9.5.4).

Of particular interest to the analyst are the relative importances of various fission and activation products with respect to the resulting radiation doses from typical spent fuel. Previous results,[6] shown in Fig. 9.1, give the fraction contribution to the total gamma-ray dose from six fission-product isotopes (^{144}Pr, ^{90}Y, ^{137}Cs, ^{110m}As, ^{154}Eu, ^{106}Rh) and one activation product (^{60}Co). These results are for PWR spent fuel burned to 35 GWd/MTU and cooled for 2 to 20 years. The resulting source particles were transported through a simple geometry, 38-cm-thick cast-iron cask (cylindrical geometry). The substantial contributions from ^{60}Co decay arise from activation of an assumed 0.5% ^{59}Co impurity level in the Inconel grid spacers. Efforts in recent years to substantially reduce the cobalt impurity levels in assembly designs should have a large effect on these relative importances. Care should be taken to accurately estimate the initial impurity levels. The remaining dose contributors all arise directly or indirectly from the fission process. Of these isotopes, ^{144}Pr dominates for short cooling times, while the ^{154}Eu dominates cooling times beyond 7 years.

Many of the isotopes in the spent fuel emit alpha particles that may interact with the ^{17}O and ^{18}O atoms in oxide fuel, producing neutrons and prompt gamma rays during the process. The measured gamma-ray spectra produced by the reaction of alpha particles from ^{238}Pu with ^{18}O atoms are typically used for developing[7–9] approximate spectra for the various alpha emitters in the data library. The spectral distribution of the prompt and delayed spontaneous fission photons can be computed from equations fit to the photon spectrum from neutron-induced fission of ^{235}U (ref. 10). The intensities of the spontaneous fission-product gamma rays can be assumed to be 0.75 times that from prompt fission photons.[11] The photons from (α,n) reactions and spontaneous fissions normally are a very insignificant part of the source in spent fuel. However, essentially all of the low-intensity gammas above 4 MeV in the spent fuel spectrum are due to spontaneous fission.

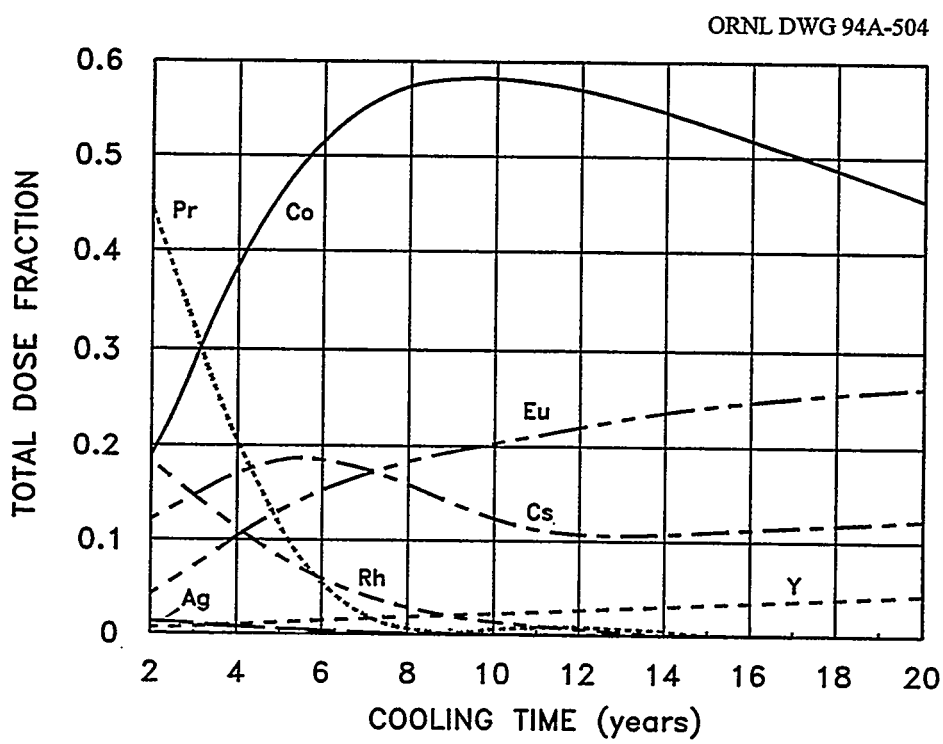

Fig. 9.1. Dose rate fractions for 35-GWd/MTU burned fuel. (Effect of cobalt is dependent on initial impurity levels).

A significant source of photons in spent fuel is the bremsstrahlung radiation, which is generated as a continuous spectrum when electrons or positrons are decelerated in the Coulomb field of nuclei. The theory indicates that only a relatively small bremsstrahlung intensity is derived from alpha particles, and so this contribution is typically not included. First, in the determination of bremsstrahlung data, the continuous energy-dependent electron and positron spectra of the nuclide are computed from the Log-f Tables derived[12] from the theory of beta decay. This model, also, is used in producing average beta energies for ENSDF.[4,5] Then, a model for computing the bremsstrahlung spectrum from the beta emission spectrum is applied (for example, see ref. 13).

The bremsstrahlung radiation is an important part of the total photon source spectrum from spent fuel. It has significant intensities for the energy range from 0 to 4 MeV. As the photon energy increases, the intensity from bremsstrahlung usually decreases more rapidly than that from delayed nuclei decay. Calculations that compared the bremsstrahlung source with the total photon source indicated about one-third of the photons and from 7 to 10% of the total energy are from bremsstrahlung. By observing these spectra and the energy-group dose rates of a somewhat typical shielding problem, it was indicated that 10 to 30% of the total photon dose rate was due to bremsstrahlung radiation.[14] Most, but not all, source generation codes account for these effects.

9.3.2 Neutron Source Strengths and Spectra

As with the calculation of gamma sources, calculation of neutron production is highly dependent on the availability and use of reliable data. The neutron source for spent fuel and other high-level waste (HLW) is a result of spontaneous fission of heavy nuclides and interaction of energetic alpha particles with a wide variety of light elements. The other potential source of neutrons is from photofission or photoneutron processes, neither of which are typically considered in codes used for radioactive sources from spent fuel or other HLW. During the first hour, or possibly several hours, after a reactor shutdown there is a significant neutron source from photoneutron reactions. At longer cooling times only insignificant fractions (<0.1%) of the neutron source are caused by the usual photoneutron target materials, the uranium fuel and the tritium in a water coolant.

Methods for computing a neutron source and spectrum emitted from the radioactive waste require data of the following types:

- decay half-lives and α particle energies;
- neutron yields per fission and half-lives for spontaneous fission;
- thick target neutron yields per (α,n) reaction in the waste medium (e.g., UO_2 for spent fuel) as a function of alpha energy;
- neutron spectra from spontaneous fission; and
- neutron spectra from (α,n) reactions caused by alpha decay from heavy nuclides.

An additional source of neutrons, typically called "source multiplication," arises when sources are present in a fissile system. These neutrons are often treated automatically in the radiation transport analysis.

The following paragraphs provide reliable (but not necessarily unique) sources for the above data. The reference sources are those employed by the ORIGEN-S code,[3] although other sources for the required data can be found (see ref. 2).

Data required to compute the neutron production rate from spontaneous fission include the spontaneous fission half-life, the average neutron yield per spontaneous fission, υ_{sf}, and the concentration for each contributing nuclide. Spontaneous fission half-lives are available from ENSDF[4] and from Kocher's compilation of decay data,[5] both of which contain measured data that have been evaluated. For several less important nuclides, calculated half-lives are available from ref. 15. These data were estimated with a correlation between measured data and so-called fissility parameters.[16] Measured values for υ_{sf} are available in ref. 15 for 21 nuclides, including the most significant. An equation, derived[15] to compute υ_{sf}, produces values that are within two experimental standard deviations for all except three nuclides. This equation can be applied for nuclides that do not have measured data.

As noted above, (α,n) neutrons are produced by alpha interactions with light elements in the waste. The principal (α,n) sources for most waste in the commercial nuclear fuel cycle are ^{17}O and ^{18}O. Thin target cross sections for $^{17}O(\alpha,n)$ and $^{18}O(\alpha,n)$ reactions and alpha stopping power data may be applied to compute neutron yields of spent fuel material. Measurements[17] of thin target cross sections for the $^{17}O(\alpha,n)$ and $^{18}O(\alpha,n)$ reactions produced improvement over earlier data.[18,19] Additionally, thick target energy-dependent (α,n) yields for $^{238}U^{NAT}O_2$ were computed,[17] having estimated accuracies within 10%. Yield data from ref. 17 can be applied to weighted energy averages of alpha energy-intensity data[5] of nearly all nuclides. Data for nuclides, such as ^{214}Bi, ^{241}Pu, and ^{249}Bk, that have very small alpha branching fractions, can be found in ref. 20. Decay constants and alpha decay branching fractions are also required to compute the (α,n) source and can be obtained from refs. 4 and 5.

The isotopes ^{242}Cm and ^{244}Cm characteristically produce all except a few percent of the spontaneous fission and (α,n) neutron source in spent pressurized-water reactor (PWR) fuel over the initial 10-year decay period. The next largest contribution (approximately 1 to 2% of the source) is usually from the (α,n) reaction of alphas from ^{238}Pu. Neutron energy spectra of both the spontaneous fission and (α,n) reactions have been determined for the curium isotopes[21,22] and ^{238}Pu (ref. 8). The measured spontaneous fission neutron spectrum of ^{244}Cm was found to be quite similar to that from ^{235}U and ^{252}Cf. Thus, the spontaneous fission spectrum for ^{242}Cm was computed[21] from these measurements. The (α,n) neutron spectra were determined by extrapolating the neutron spectrum from Po-α-O source measurements[23] to the alpha energies of ^{242}Cm, ^{244}Cm, and ^{238}Pu. The energy distribution of the spontaneous fission neutron spectrum can be computed from the spectra for ^{242}Cm and ^{244}Cm by using the calculated concentrations of those two isotopes and then renormalizing to include the total neutron source from all isotopes capable of spontaneous fission. A similar approach using the data for all three isotopes can be performed for the (α,n) neutron spectrum. The spectra must be collapsed from the energy group structure of the data to the desired group structure used by the code. One procedure is to assume uniform distribution of particle energies within each group and simply sum the quantities based upon energy fractions common to both groups in the two group structures. The total neutron source spectrum is then computed as the sum of the spontaneous fission and (α,n) spectra.

The relative importance of the spontaneous fission and (α,n) sources for PWR and BWR fuel at different burnups is well illustrated by the neutron radiation sources shown in Figs. 9.2–9.3 (ref. 24). It is important to notice that the total neutron source at discharge increases approximately as the fourth power of the burnup. However, the long half-life of the ^{244}Cm causes the neutron source to decrease slowly with cooling time after discharge. Since the gamma source changes linearly with burnup (following the fission-product trend) and the primary gamma-ray-producing isotopes have relatively short half-lives, it follows that the overall importance of the neutron source relative to the gamma source increases with burnup and cooling time. Finally, note

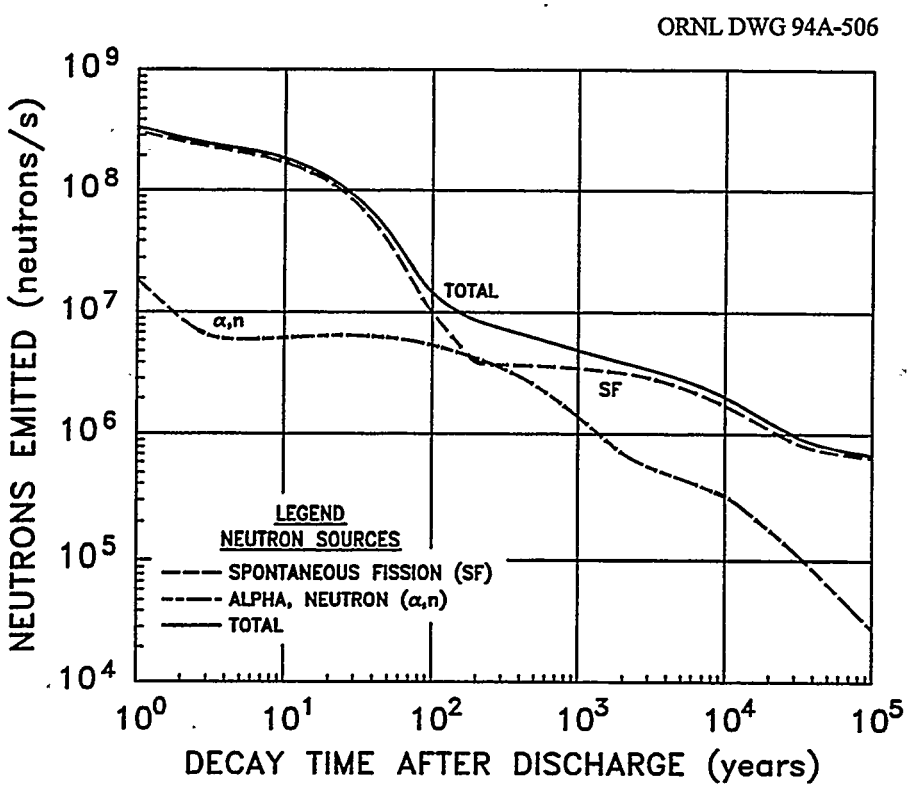

Fig. 9.2. Neutrons emitted by 1 metric ton of initial heavy metal (MTIHM): PWR/33,000 MWd/MTU.
Source: ref. 24.

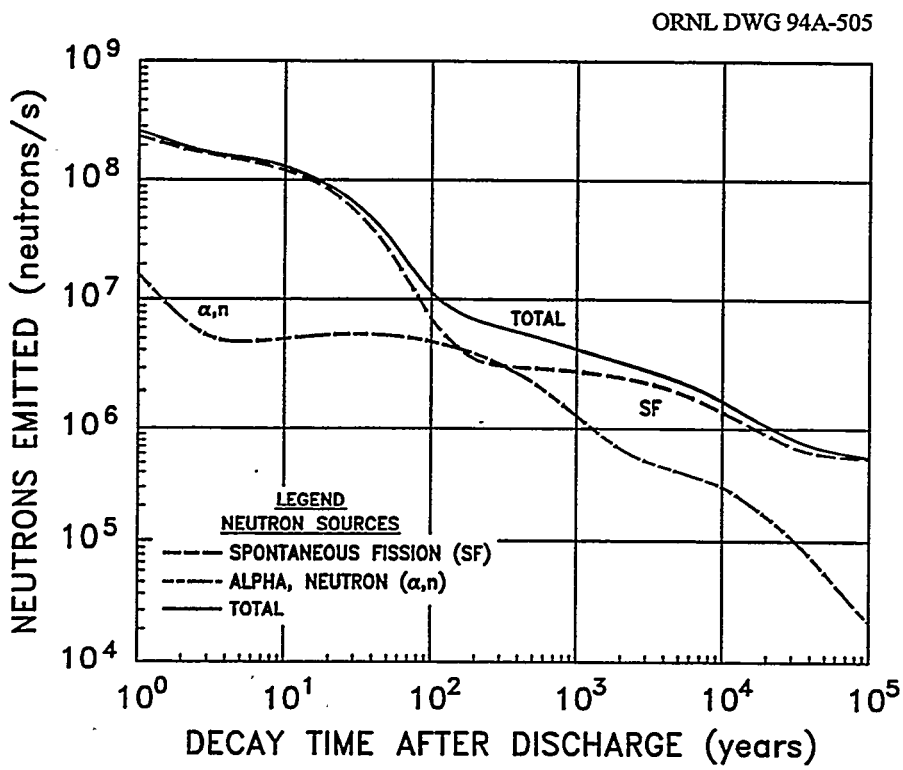

Fig. 9.3. Neutrons emitted by 1 metric ton of initial heavy metal (MTIHM): BWR/27,500 MWd/MTU.
Source: ref. 24.

that for applications involving vitrified HLW, light elements present in the glass can yield a significant (α,n) source that should be accounted for, and additional (α,n) yield data and spectrum data besides that for oxygen must be obtained.[25]

9.3.3 Range of Applicability

As evidenced from Sects. 9.3.1 and 9.3.2, the reliability of the radiation sources produced by a code is primarily dependent on the number and accuracy of the given isotopic concentrations and the completeness and accuracy of the data employed. This section discusses briefly the facets of spent fuel source terms that an analyst should be aware of prior to the use and/or selection of a source-term generation procedure.[26–28]

9.3.3.1 Source Strength

One of the major contributors to uncertainty in the radiation sources can be the uncertainty in the calculated isotopic inventory. The calculation of isotopic inventories for spent fuel is more sensitive to the use of proper cross sections for the production of heavy metals than it is for the production of fission products. This high sensitivity of spent fuel actinides inventory to the burnup-dependent cross sections used in the point-depletion calculation causes the radiation sources from actinides to have higher *potential* inaccuracy than from either the fission or activation products. Thus, as the importance of the actinide contribution to the total radiation source increases with burnup and cooling time, the shielding analyst needs to be aware of this potential inaccuracy and seek to verify the accuracy of the actinide inventory or investigate its importance (sensitivity study) on the final dose results. Of primary importance is ensuring that the burnup-dependent cross sections used in the point-depletion model enable an adequate representation of the physics in the reactor of interest.

The photon source from high-level radioactivity is usually dominated by the contributions from fission products, and possibly one or two activation products. However, for shielding applications, it is often the energy of the emitted gamma rays, rather than the intensity, that is important. Thus, such activation products as ^{60}Co in spent fuel assembly structures can produce a relatively small proportion of the overall assembly source strength, but can still be a dominant factor in the dose rate outside a shield. The reason for this is the relatively high-energy (1.17 and 1.33 MeV gammas/disintegration) gamma rays emitted by ^{60}Co. The shielding analyst needs to be aware of the importance of this activation product and seek accurate information on the ^{59}Co content in the as-built assembly.

Note that shortly after discharge and thereafter, the photon emission rates in the higher energy range come primarily from the actinides. Typically, this is true for photons whose energies are above 4 MeV at all cooling times and at energies above 3.5 MeV after 10 years. The major part of this higher range of the spectrum is contributed by ^{244}Cm after 90 days, and at least 90% appears to be from ^{244}Cm during the range of 10 to 50 years cooling time.[29] The low intensity from high-energy photons from the heavy-metal isotopes can contribute a dose rate fraction comparable to that of activation and fission products only for long cooling times and extremely thick shields.

Given accurate concentrations of heavy-element nuclides in spent fuel, the major uncertainty in the computed neutron source strength is in the neutrons produced by spontaneous fission of ^{242}Cm

and ^{244}Cm. At discharge, the neutron source is almost equally produced by ^{242}Cm and ^{244}Cm. The half-lives of ^{242}Cm and ^{244}Cm are about 163 days and 18 years, respectively. The neutron emission is mainly from ^{242}Cm at shorter cooling times and from ^{244}Cm at longer cooling times. The source computed from a given quantity of ^{244}Cm is more accurate due to less uncertainty in its spontaneous fission half-life. For the data referenced in Sect. 9.3.2, it has been estimated that over a 10-year cooling time the combined uncertainty in the computed neutron source from the spontaneous fission is in the range 4 to 8%, with a decrease in uncertainty as a function of time resulting from the decay of ^{242}Cm. The total source strength uncertainty for given isotopic concentrations is assessed as being <10%.[27]

As noted in Figs. 9.2 and 9.3, the neutrons from spontaneous fission most often dominate the neutron source strength. However, for lower burnups and intermediate cooling times the (α,n) contribution to the neutron source can be extremely important and should not be ignored.

The above discussion, together with that of Sect. 9.3.1, points out that computer programs that omit one or more of the isotopic categories (activation products, fission products, and actinides) or production processes may result in an inadequate definition of the source strength. The importance of the missing isotopes and/or production processes should be determined by the user for the application at hand.

9.3.3.2 Source Spectra

As will be discussed in Sect. 9.5, many radiation transport codes require the radiation source to be provided in a multi-energy-group format that corresponds to the multigroup cross-section set being utilized for the radiation transport analysis. Figure 9.4 is a plot comparing three "broadly averaged" groups of a PWR spent fuel gamma-ray spectrum with that generated for the same source in 0.01-MeV group intervals. It illustrates the pronounced variations of the intensities in the detailed spectrum. As described in Sect. 9.3.1, the best technique for calculating the intensity associated with each energy group is to conserve the total energy of the gammas produced. Thus, for the graphical representation of Fig. 9.4, the area under each of the "broadly averaged" groups times the group average energy should equal the sum of the products of the area and average energy of each fine group within the broad group. Attempts to conserve only the intensity when allotting particles to a group can cause either an overprediction or underprediction of the results depending on the procedure used. For example, 1.1-MeV gamma rays of intensity, I, binned directly into an energy group with limits 1.0 to 1.5 MeV would have an effective energy of 1.25 MeV (group average) that would result in a higher dose than if the energy was conserved (i.e., use of a modified intensity $I_{mod} = I(1.1)/1.25$). A sample comparison of dose results for an intensity or particle-conserved source vs an energy-conserved source is shown in Sect. 9.5.4.1.

The dose rates computed for two monoenergetic photon sources shielded with a typical shield thickness are highly dependent on their respective photon energies. Importance calculations have indicated that the importance of photons in two adjacent energy groups may differ by up to a factor of 2 or 3 for significant groups in a typical 18-group structure (see discussion of data libraries in Sect. 9.5.1.3). Thus, the benefit of finer group intervals should be properly balanced against computational efficiency in selecting an appropriate energy group structure. Also, due to the importance of using a correct energy-intensity combination, the conversion of one photon spectrum having a broad group structure to a spectrum of another group structure may significantly reduce its quality.

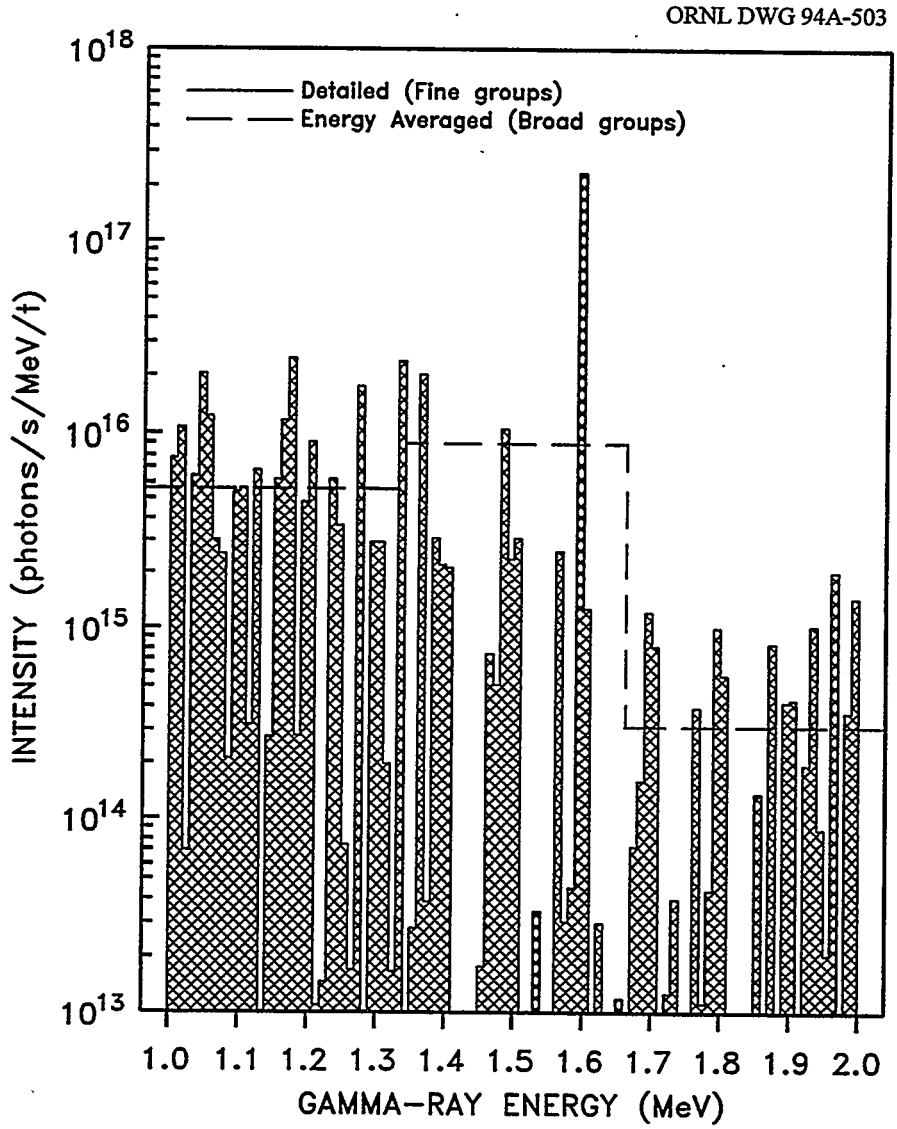

Fig. 9.4. Part of spectrum, comparing intensities for different energy-group widths. *Source:* ref. 28.

There are no reported uncertainties in the data applied for conversion to the neutron source spectra given in the references of Sect. 9.3.2. However, the use of measured spectral data should be an improvement over the use of standard fission spectrum formulae (e.g., Watt fission spectrum) that are often employed by shielding analysts when a spectrum is not provided with the neutron strength (see Sect. 9.5.4.1).

9.4 APPROXIMATE METHODS FOR PRELIMINARY CASK DESIGN

The use of approximate techniques for evaluating the radiation dose attenuation through a single shield or a series of shielding materials can be an extremely useful tool in the cask preliminary design phase. The use of simple point-kernel techniques, as well as the application of material attenuation factors, are two means of shield evaluations that can be performed by hand. These techniques are useful as a starting point in the shielding design, as a means of performing simple

trade-off and optimization studies, and as a quick check on the final designs by a reviewer. Two such techniques will be described herein with the goal of allowing a wide variety of cask design applications to be represented. The first technique is generally applicable to individual isotope package design and review. The second is primarily applicable to light-water-reactor (LWR) spent fuel cask design.

9.4.1 Use of Isotopic Radiological Properties in Cask Design and Review

The methods identified in the following parts rely upon a data compilation entitled *Isotopic Source/Shield Design Compilation* (IS/SDC) by E. D. Arnold and published here as Appendix D. That compilation lists radiation and thermal emission properties for isotopes that have half-lives exceeding 10 days. The material included in the following parts are used to estimate gamma and bremstrahlung shielding requirements for radioisotope shipping casks. The IS/SDC addresses other forms of isotope decay or emissions (such as alpha emission, spontaneous fission, (α,n) reactions, conversion electrons) for most isotopes (see Appendix D). The techniques are also applicable to other shielding applications within bounds determined by size.

These shield-estimating methodologies are based on point-source estimates. Refinements can be made by estimating the degree of conservatism imposed by using point sources. Actual sources have self-absorption and may require different and more exact methods based on geometry. The IS/SDC contains approximations for converting spherical and cylindrical volume sources to point sources as well as techniques accounting for the source self-absorption.

The geometric limits for using the point-source approximation have been determined to be a volume of ~2000 cm^3 or 20 kg at ρ = 10 g/cm^3 or 36 kg at ρ = 18 g/cm^3 for a geometric error less than ~20%.

A sequential methodology to determine a first-estimate shield thickness using isotopic radiation and thermal properties from the IS/SDC or other data sources is the following:

1. Determine source volume and compare it to 2000 cm^3. If the volume is greater than 2000 cm^3, then do not use these approximations, but instead use volumetric source geometry or use a computer program with volumetric options.

2. Determine self-absorption factor (SAF) by using the following approximation:

$$SAF \approx e^{-\frac{\mu_s r}{2}}, \; for \; \mu_s r < 3, \tag{9.2}$$

   where μ_s is the absorption coefficient for the source material at the source energy and r is the radius of the source sphere or cylinder. For $\mu_s r > 3$, see Figs. 6.4-11 and 6.4-19 of ref. 30.

3. Proceed if the source volume is less than 2000 cm^3 and SAF can be determined.

4. Estimate the required number of curies of a given isotope in the cask.

5. Determine the specific gamma ray constant, Γ_γ [units of rem × cm²/(h × Ci)]. The values of Γ_γ are given in Appendix D.

6. Determine the unshielded dose rate, D_{calc}, at a given dose point,

$$D_{calc} = \frac{S_c \cdot SAF \cdot \Gamma}{R^2}, \tag{9.3}$$

where S_c is the number of curies in the source, Γ is the total specific gamma-ray constant, $\Gamma = \Gamma_\gamma + \Gamma_x$, and R is the distance from the dose point to the source.

7. Include bremstrahlung via obtaining the value of Γ_x if it is significant for the given isotope. Bremstrahlung methodologies and values of Γ_x are included in Appendix D.

8. Determine an effective photon energy (E_{eff}). The values for most isotopes are tabulated in Appendix D.

9. Use appropriate shielding texts or shielding handbooks[31–32] to find μ_{eff}, which is the shield absorption coefficient evaluated at E_{eff}.

10. Estimate the required attenuation factor by evaluating

$$AF = e^{-(\mu t)^*} = \frac{D_{req}}{B\,D_{calc}} = \frac{D_{req} R^2}{B\,S_c \cdot SAF \cdot \Gamma}, \tag{9.4}$$

where the buildup factor, B, is approximately 5 for lead, 25 for iron, and 40 for concrete; then estimate the required mean-free paths, $(\mu t)^*$, of shielding material:

11. Evaluate buildup factor (B^*) at [E_{eff}, $(\mu t)^*$] by using the same references as in Step 9.

12. Reevaluate $(\mu t)^*$ using

$$(\mu t)^* = -\ln \frac{D_{req}}{B\,D_{calc}} = -\ln\left(\frac{D_{req} R^2}{B\,S_c \cdot SAF \cdot \Gamma}\right). \tag{9.5}$$

$$(\mu t)^* = \ln B^* - \ln \frac{D_{req}}{D_{calc}}. \tag{9.6}$$

13. Estimate shield thickness t^* from $t^* = (\mu t)^*/\mu_{eff}$.

14. Adjust estimated radius R if t^* + cavity radius > R.

The following example is presented for the parent-daughter decay chain of ^{106}Ru—^{106}Rh. The necessary data for ^{106}Ru—^{106}Rh are obtained from the data tables given in Appendix D.

For ^{106}Ru—^{106}Rh

Specific activity = 3.346×10^3 Ci/g of pure ^{106}Ru isotope.
Specific gamma-ray constants (Γ_γ and Γ_x) are
Γ_γ = 1354 rem · cm²/(h · Ci) for gamma rays,
Γ_x = 584.2 rem · cm²/(h · Ci) for bremstrahlung, and
Γ_{total} = 1938 rem · cm²/(h · Ci).
Effective photon energies = 0.941 MeV for gammas, 1.4 MeV for bremstrahlung, and 1.22 MeV when γ's and X rays are combined.
The tenth thicknesses are 3.57 cm Pb as combined, or 3.43 cm for gammas only.

Using ^{106}Ru—^{106}Rh data (^{106}Rh controls dose) and Steps 1 through 14 obtain:

Steps 1–4 As an example, a package will be designed for 1000 Ci. Since the specific activity is 3.346×10^3 Ci/g, 1000 Ci = 0.3 g of pure isotope or possibly 1 to 10 g as a practical source (size might be determined elsewhere or by other designers). Thus, size is not important, and SAF = 1.0 is reasonable.

Step 5 The specific gamma-ray constant (Γ_γ) is 1354 rem · cm²/(h · Ci) for gamma rays and 584.2 for bremstrahlung or 1938 total.

Step 6 Assume cask radius = 15 cm (or another radius if other design information is known); applying Eq. (9.3), the dose rate estimate is

$$D_{calc} = \frac{1000\ Ci \cdot 1938(rem \cdot cm^2/(h \cdot Ci)}{(15\ cm)^2} = 8.613 \times 10^3\ rem/h\ .$$

Step 7 Bremstrahlung has been included in Step 5.

Step 8 From earlier information given in this example, the effective photon energy = 0.941 MeV for gamma photons and 1.4 MeV for bremstrahlung; a combined E_{eff} is 1.22 MeV.

Step 9 At 1.22 MeV, μ/ρ for lead is 0.0605 cm²/g, so $\mu = 0.684\ cm^{-1}$ for $\rho = 11.3\ g/cm^3$, the density of lead.

Step 10 The required attenuation factor would be $0.2/8.613 \times 10^3 = 2.322 \times 10^{-5}$, where 0.2 is the surface dose rate limit (in rem/h) specified in the regulations, and 8.613×10^3 is the calculated dose from Step 6. Then $(\mu t)^* = -\ln(1/5 \cdot 2.322 \times 10^{-5}) = 12.28$

Step 11 Buildup factor is 4.314 as evaluated at 1.22 MeV, and $\mu t = 12.28$.

Step 12 Reevaluation of $(\mu t)^*$ yields $(\mu t)^* = \ln 4.314 - \ln (0.2/8.613 \times 10^3) = 12.132$.

Step 13 Then $t^* = 12.132/0.684 = 17.73$ cm Pb.

Step 14 There is a mismatch between thickness and radius; thus, instead of a 15-cm radius, use an estimated adjusted radius of 22 cm and adjust thickness by applying the inverse square rule to result in a correction of $15^2/22^2$ converted to tenth thicknesses or $\log_{10} (15^2/22^2) = -0.333$. The estimated tenth thickness at 1.22 MeV is 3.57 cm, and the adjusted thickness = $17.73 - 0.333 \cdot 3.57 = 16.55$ cm. At $R = 22$ and $t^* = 16.55$, the cavity radius would be 5.45 cm, and the preliminary design is reasonable. (Cavity radius to be established was to be close to 5.0 cm.)

9.4.2 Preliminary Spent Fuel Designs Using CAPSIZE

A simple interactive program called CAPSIZE[32] has been written for personal computers to determine the likely impact that proposed design objectives might have on the size and capacity of spent fuel shipping packaging designed to meet those objectives. Given the burnup of the spent fuel, its cooling time, the thickness of the internal basket walls, the desired external dose rate, and the nominal weight limit of the loaded package, the CAPSIZE program will determine in an approximate fashion the maximum number of PWR fuel assemblies that may be shipped in a lead-, steel-, or uranium-shielded package meeting those objectives. The necessary neutron and gamma shield thicknesses are determined by the program in such a way as to meet the specified external dose rate while simultaneously minimizing the overall weight of the loaded package. The approximations include an assumed fixed initial enrichment of 3.3 wt %, side doses only and PWR fuel only.

The one-group cross sections used in the CAPSIZE program and shown in Tables 9.2 and 9.3 have been distilled from the intermediate results of several hundred 1-D multigroup discrete-ordinates calculations for different types of casks. Neutron and gamma source terms shown in Tables 9.4 and 9.5 are based on ORIGEN-S analyses of PWR fuel assemblies having exposures of 10-, 20-, 30-, 40-, 50-, and 60 gigawatt days per metric ton of initial heavy metal (GWd/MTU). In each case, values have been tabulated at 17 different decay times between 120 days and 25 years. Other features of the CAPSIZE program include a steady-state heat transfer calculation that will minimize the size and weight of external cooling fins, if and when such fins are required.

In the CAPSIZE program, the thicknesses of the neutron and gamma shields are calculated so that when used in conjunction with specified inner and outer steel shells and an outside liner, the combined neutron and gamma dose rates 10 ft from the centerline of the package will be reduced to some prescribed level. The unshielded dose rates with none of these five components present are assumed to vary as

$$D_n^o = (100.0 \text{ mrem}/h) \frac{S_n(B,T)}{S_n(B_o,T_o)} \frac{F_n(N)}{F_n(N_n)} , \qquad (9.7a)$$

Table 9.2. Macroscopic cross sections (in.$^{-1}$) for spatial attenuation of the gamma dose rate in the various components of different casks as a function of the cooling time of the spent fuel[a]

Type of cask	$t_{ref}^{g.shld}$	Time (year)	$\Sigma_{\gamma}^{i.shell}$	$\Sigma_{\gamma,avg}^{g.shld}$	$\Sigma_{\gamma,dif}^{g.shld}$	$\Sigma_{\gamma}^{o.shell}$	$\Sigma_{\gamma}^{n.shld}$	$\Sigma_{\gamma}^{o.liner}$
Pb	4.50 in.	1	1.0993	1.5924	1.3114	1.0993	0.1119	0.7558
		3	1.1146	1.6499	1.4057	1.1146	0.1195	0.8015
		5	1.1255	1.7059	1.4876	1.1255		
		10	1.1502	1.7382	1.5437	1.1502	0.1305	0.8634
Fe	10.00 in.	1	1.0855	0.9951	0.8566	1.0855	0.1237	0.8381
		3	1.1003	1.0207	0.9182	1.1003	0.1327	0.9035
		5	1.1103	1.0434	0.9692	1.1103		
		10	1.1351	1.0623	0.9973	1.1351	0.1452	0.9906
U	2.75 in.	1	1.1029	2.9372	2.3494	1.1029	0.1071	0.7404
		3	1.1181	3.0532	2.5170	1.1181	0.1129	0.7771
		5	1.1285	3.1645	2.6952	1.1285		
		10	1.1540	3.2274	2.7873	1.1540	0.1248	0.8465

[a] $\Sigma_{\gamma,avg}^{g.shld}$ and $\Sigma_{\gamma,dif}^{g.shld}$ are average and differential cross sections, respectively, for the gamma shield, and $t_{ref}^{g.shld}$ is the reference thickness of the gamma shield to be used with these cross sections in calculating the attenuation through a gamma shield of thickness t_{γ}.

Table 9.3. Macroscopic cross sections (in.$^{-1}$) for spatial attenuation of the neutron dose rate in the various components of different casks as a function of the cooling time of the spent fuel[a]

Type of cask	$t_{ref}^{n.shld}$	Time (y)	$\Sigma_{n}^{i.shell}$	$\Sigma_{n}^{g.shld}$	$\Sigma_{n}^{o.shell}$	$\Sigma_{n,avg}^{n.shld}$	$\Sigma_{n,dif}^{n.shld}$	$\Sigma_{n}^{o.liner}$
Pb	4.00 in.	1	0.1530	0.0617	0.1530	0.9266	0.7894	0.1530
		3	0.1546	0.0643	0.1546	0.9140	0.7837	0.1546
		5				0.9019	0.7760	
		10	0.1565	0.0723	0.1565	0.8871	0.7610	0.1565
Fe	3.75 in.	1	0.1530	0.1530	0.1530	1.1125	1.0113	0.1530
		3	0.1546	0.1546	0.1546	1.0828	0.9791	0.1546
		5				1.0636	0.9624	
		10	0.1565	0.1565	0.1565	1.0359	0.9236	0.1565
U	3.25 in.	1	0.1530	0.2735	0.1530	0.9083	0.7765	0.1530
		3	0.1546	0.2781	0.1546	0.9001	0.7774	0.1546
		5				0.8918	0.7729	
		10	0.1565	0.2860	0.1565	0.8802	0.7643	0.1565

[a] $\Sigma_{n,avg}^{n.shld}$ and $\Sigma_{n,dif}^{n.shld}$ are average and differential cross sections, respectively, for the gamma shield, and $t_{ref}^{n.shld}$ is the reference thickness of the gamma shield to be used with these cross sections in calculating the attenuation through a gamma shield of thickness t_{n}.

Table 9.4. Neutron source (n/s·MTU) for PWR spent fuel as a function of burnup and cooling time[a,b]

Cooling time (year)	Burnup (MWd/MTU)					
	10000	20000	30000	40000	50000	60000
0.00	2.113E+07	1.638E+08	5.317E+08	1.241E+09	2.442E+09	4.370E+09
0.33	1.352E+07	1.118E+08	4.004E+08	1.018E+09	2.126E+09	3.953E+09
0.41	1.213E+07	1.022E+08	3.761E+08	9.764E+08	2.066E+09	3.871E+09
1.00	5.972E+06	5.974E+07	2.673E+08	7.865E+08	1.786E+09	3.474E+09
2.00	2.732E+06	3.670E+07	2.051E+08	6.696E+08	1.594E+09	3.158E+09
3.00	2.050E+06	3.109E+07	1.867E+08	6.262E+08	1.505E+09	2.979E+09
4.00	1.910E+06	2.921E+07	1.778E+08	5.992E+08	1.441E+09	2.839E+09
5.00	1.258E+06	2.813E+07	1.710E+08	5.763E+08	1.385E+09	2.715E+09
7.00	1.203E+06	2.645E+07	1.590E+08	5.347E+08	1.281E+09	2.494E+09
8.00	1.180E+06	2.567E+07	1.535E+08	5.152E+08	1.233E+09	2.394E+09
10.00	1.137E+06	2.421E+07	1.429E+08	4.785E+08	1.143E+09	2.211E+09
13.00	1.077E+06	2.221E+07	1.286E+08	4.284E+08	1.022E+09	1.969E+09
15.00	1.042E+06	2.099E+07	1.199E+08	3.981E+08	9.482E+08	1.826E+09
18.00	9.926E+05	1.932E+07	1.080E+08	3.568E+08	8.485E+08	1.632E+09
20.00	9.629E+05	1.831E+07	1.008E+08	3.317E+08	7.882E+08	1.515E+09
23.00	9.224E+05	1.692E+07	9.092E+07	2.976E+08	7.060E+08	1.357E+09
25.00	8.978E+05	1.607E+07	8.495E+07	2.769E+08	6.563E+08	1.261E+09

[a]Each PWR fuel assembly initially contains 0.4614 t of heavy metal with an initial enrichment of 3.3 wt %.

[b]Includes the (α,n) neutron source terms as well as the spontaneous fission source terms.

Table 9.5. Gamma source (p/s·MTU) for PWR spent fuel as a function of burnup and cooling time[a,b]

Cooling time (year)	Burnup (MWd/MTU)					
	10000	20000	30000	40000	50000	60000
0.00	4.5073E+18	8.8470E+18	1.3106E+19	1.7321E+19	2.1546E+19	2.5784E+19
0.33	4.3467E+16	8.9069E+16	1.3669E+17	1.8600E+17	2.3666E+17	2.8809E+17
0.41	3.5746E+16	7.3938E+16	1.1435E+17	1.5657E+17	2.0020E+17	2.4463E+17
1.00	1.4457E+16	3.1805E+16	5.1554E+16	7.3209E+16	9.6274E+16	1.2014E+17
2.00	7.0732E+15	1.6166E+16	2.6952E+16	3.9075E+16	5.2178E+16	6.5832E+16
3.00	4.3753E+15	1.0103E+16	1.6991E+16	2.4790E+16	3.3252E+16	4.2083E+16
4.00	3.1181E+15	7.1509E+15	1.1987E+16	1.7450E+16	2.3361E+16	2.9520E+16
5.00	2.4910E+15	5.6081E+15	9.2891E+15	1.3403E+16	1.7820E+16	2.2403E+16
7.00	1.9507E+15	4.2032E+15	6.7422E+15	9.4906E+15	1.2374E+16	1.5330E+16
8.00	1.8175E+15	3.8460E+15	6.0811E+15	8.4612E+15	1.0929E+16	1.3442E+16
10.00	1.6444E+15	3.3906E+15	5.2448E+15	7.1658E+15	9.1173E+15	1.1081E+16
13.00	1.4768E+15	2.9805E+15	4.5222E+15	6.0768E+15	7.6244E+15	9.1635E+15
15.00	1.3905E+15	2.7846E+15	4.1934E+15	5.5986E+15	6.9861E+15	8.3591E+15
18.00	1.2804E+15	2.5461E+15	3.8075E+15	5.0524E+15	6.2726E+15	7.4743E+15
20.00	1.2155E+15	2.4098E+15	3.5925E+15	4.7542E+15	5.8894E+15	7.0049E+15
23.00	1.1271E+15	2.2278E+15	3.3098E+15	4.3671E+15	5.3970E+15	6.4066E+15
25.00	1.0730E+15	2.1178E+15	3.1409E+15	4.1380E+15	5.1076E+15	6.0569E+15

[a]Each PWR fuel assembly initially contains 0.4614 t of heavy metal, with an initial enrichment of 3.3 wt %.

[b]Includes contributions from all of the fission products, the actinides, and the light elements. Also includes contributions due to activation of the Fe, Ni, and Co in the structural materials comprising the fuel assemblies.

and

$$D_\gamma^o = (1.376E{+}6 \text{ mrem/h}) \frac{S_\gamma(B,T)}{S_\gamma(B_o,T_o)} \frac{F_\gamma(N)}{F_\gamma(N_\gamma)} \frac{\chi(T)}{\chi(T_\gamma)} , \tag{9.7b}$$

where B is the burnup of the spent fuel (MWd/MTU), T is the cooling time in years, N is the number of PWR assemblies in the package, $S_n(B,T)$ is the number of n/s/MTU in the spent fuel as given by the ORIGEN data, $S_\gamma(B,T)$ is the number of p/s·MTU in the spent fuel as given by the ORIGEN data, and $S_n(B_o,T_o)$ and $S_\gamma(B_o,T_o)$ are reference values at $B_o = 33{,}000$ MWd/MTU and $T_o = 10$ years.

$$F_n(N) = N^{0.73643} \tag{9.8a}$$

is a geometric/self-attenuation factor for the neutrons, with $F_n(N_n) = 9.413$ being a reference value based on $N_n = 21$ assemblies.

$$F_\gamma(N) = \sqrt{N} + (0.0919/\sqrt{N}) \tag{9.8b}$$

is a geometric/self-attenuation factor for the gammas, with $F_\gamma(N_\gamma) = 3.491$ being a reference value based on $N_\gamma = 12$ assemblies, and $\chi(T)$ is an empirical factor that accounts for spectral hardening of the gamma source in fuel that has cooled for 5 to 7 years (y). This spectral correction factor is given by

$$\begin{aligned}
\chi(T) &= 1.0 && T < 1 \text{ y} && (9.9a)\\
&= 1.101429 - (0.17533)T + (0.073905)T^2 && 1 \text{ y} < T < 2.5 \text{ y} && (9.9b)\\
&= 0.465568 + (0.3567)T - (0.037173)T^2 && 2.5 \text{ y} < T < 5 \text{ y} && (9.9c)\\
&= 1.405376 - (0.003452)T - (0.002734)T^2 && 5 \text{ y} < T < 10 \text{ y} && (9.9d)\\
&= 1.09744 && T > 10 \text{ y}, && (9.9e)
\end{aligned}$$

with a reference value of $\chi(T_\gamma) = 1.09744$ at $T_\gamma = 10$ years.

The macroscopic cross sections (in.$^{-1}$) shown in Tables 9.2 and 9.3 can then be used for calculating the spatial attenuation of the photon and neutron dose rate in the various components of Pb, Fe, and U-metal packages containing PWR spent fuel that has been out of the reactor for 1, 3, 5, and 10 years. Values at intermediate times are obtained by interpolation, while the values at 1 and 10 years are assumed to apply to fuel that has been out of the reactor for less than 1 year or more than 10 years. The neutron dose rate (and, similarly, the gamma dose rate) on the outer surface of the inner shell may, for example, be calculated as

$$D_{outer}^{n} = D_{inner}^{n} \exp\{- \Sigma_n^{i.shell} \, t^{i.shell}\} , \tag{9.10}$$

where D^{n}_{inner} is the neutron dose rate on the inner surface and $t^{i.shell}$ is the thickness of the inner steel shell in inches. The same basic formula applies for attenuation through the gamma shield, the outer steel shell, and the outside liner. Attenuation through the neutron shield, however, is given by

$$D^{n}_{outer} = D^{n}_{inner} \exp\{-[\Sigma^{n.shld}_{n,avg}\, t^{n.shld}_{ref} + \Sigma^{n.shld}_{n,dif}(t_n - t^{n.shld}_{ref})]\}\ , \qquad (9.11)$$

where D^{n}_{inner} and D^{n}_{outer} are the neutron dose rates on the inner and outer surfaces, t_n is the actual thickness of the neutron shield, and $t^{n,shld}_{ref}$ depends on the type of package as shown in Table 9.3.

These formulas as presented are useful as simple checks or for preliminary design studies for a wide range of spent fuel packages. Their use in the CAPSIZE program allows for a rapid package optimization to be performed based on these tabulated parameters. Typical results from such a study are shown in Tables 9.6 through 9.10. These tables give optimized neutron and gamma shield thicknesses for Pb, Fe, and DU shields based on a variable number of PWR fuel assemblies burned to 33,000 MWd/MTU and cooled for 2, 3, 5, 7, and 10 years and which results in a dose rate of 10 mrem/h at 10 ft from the centerline of the cask. These neutron and gamma shield thicknesses can often provide the basis for more detailed dose computations with more advanced shielding methods.

9.5 CALCULATIONAL TECHNIQUES FOR DOSE EVALUATION

Many calculational techniques as applied to shielding analyses of storage/transport casks are reviewed and discussed extensively in ref. 33. Much of the information presented in this section is extracted from that reference.

9.5.1 Review of Codes and Data

A variety of computer programs have been developed over the years to treat each of the separate aspects of a complete shielding analysis as applied to packages designed to transport radioactive materials. The major source of information on available codes and data libraries associated with radiation shielding is the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory.

Table 9.6. Optimized neutron and gamma shielding requirements[a] for Pb, Fe, and U-metal spent fuel shipping casks as a function of the number of 2-year-old PWR assemblies in the cask[b,c]

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield[d]	G-shield	N-shield[d]	G-shield	N-shield[d]
1	4.47	3.71	10.11	3.81	2.72	3.10
4	4.93	3.98	10.81	3.86	2.98	3.32
8	5.18	4.06	11.18	3.86	3.13	3.49
10	5.28	4.13	11.33	3.89	3.18	3.58
12	5.37	4.25	11.46	3.92	3.23	3.65
15	5.44	4.24	11.56	3.90	3.27	3.67
18	5.49	4.24	11.64	3.93	3.31	3.67
21	5.58	4.38	11.77	3.93	3.36	3.76

[a]Neutron and gamma shield thicknesses are given in inches.

[b]Results in a dose rate of approximately 9 mrem/h gamma and 1 mrem/h neutron at 10 ft from the cask centerline.

[c]Assumes inner steel shell, outer steel shell, and outside steel liner thicknesses of 1.5, 2.0, and 0.75 in. for Pb cask; 0.75, 2.0, and 0.75 in. for U-metal cask; and 0.375, 0.375, and 0.75 in. for Fe cask.

[d]Neutron shield assumed to be a water/ethylene glycol mixture.

Table 9.7. Optimized neutron and gamma shielding requirements[a] for Pb, Fe, and U-metal spent fuel shipping casks as a function of the number of 3-year-old PWR assemblies in the cask[b,c]

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield[d]	G-shield	N-shield[d]	G-shield	N-shield[d]
1	4.06	3.72	9.52	3.68	2.49	2.86
4	4.50	3.97	10.19	3.76	2.74	3.18
8	4.74	4.05	10.55	3.74	2.88	3.30
12	4.92	4.19	10.82	3.79	2.99	3.48
15	4.99	4.21	10.92	3.79	3.02	3.57
18	5.05	4.24	10.99	3.82	3.06	3.58
21	5.12	4.34	11.11	3.87	3.11	3.61

[a]Neutron and gamma shield thicknesses are given in inches.

[b]Results in a dose rate of approximately 9 mrem/h gamma and 1 mrem/h neutron at 10 ft from the cask centerline.

[c]Assumes inner steel shell, outer steel shell, and outside steel liner thicknesses of 1.5, 2.0, and 0.75 in. for Pb cask; 0.75, 2.0, and 0.75 in. for U-metal cask; and 0.375, 0.375, and 0.75 in. for Fe cask.

[d]Neutron shield assumed to be a water/ethylene glycol mixture.

Table 9.8. Optimized neutron and gamma shielding requirements[a] for Pb, Fe, and U-metal spent fuel shipping casks as a function of the number of 5-year-old PWR assemblies in the cask[b,c]

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield[d]	G-shield	N-shield[d]	G-shield	N-shield[d]
1	3.55	3.60	8.79	3.44	2.21	2.54
4	3.97	3.85	9.42	3.48	2.45	2.85
8	4.19	3.95	9.76	3.51	2.58	3.09
12	4.37	4.07	10.02	3.57	2.67	3.30
15	4.43	4.08	10.11	3.56	2.71	3.33
18	4.49	4.07	10.19	3.57	2.74	3.35
21	4.56	4.21	10.30	3.62	2.79	3.45
24	4.60	4.25	10.35	3.65	2.81	3.44

[a]Neutron and gamma shield thicknesses are given in inches.

[b]Results in a dose rate of approximately 9 mrem/h gamma and 1 mrem/h neutron at 10 ft from the cask centerline.

[c]Assumes inner steel shell, outer steel shell, and outside steel liner thicknesses of 1.5, 2.0, and 0.75 in. for Pb cask; 0.75, 2.0, and 0.75 in. for U-metal cask; and 0.375, 0.375, and 0.75 in. for Fe cask.

[d]Neutron shield assumed to be a water/ethylene glycol mixture.

Table 9.9. Optimized neutron and gamma shielding requirements[a] for Pb, Fe, and U-metal spent fuel shipping casks as a function of the number of 7-year-old PWR assemblies in the cask[b,c]

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield[d]	G-shield	N-shield[d]	G-shield	N-shield[d]
1	3.28	3.54	8.35	3.49	2.05	2.67
4	3.69	3.70	8.98	3.50	2.28	2.93
8	3.91	3.82	9.31	3.55	2.40	3.14
12	4.08	3.94	9.56	3.59	2.50	3.34
15	4.15	3.95	9.65	3.59	2.53	3.37
18	4.20	4.01	9.73	3.58	2.56	3.40
21	4.27	4.03	9.84	3.62	2.61	3.49
26	4.35	4.16	9.95	3.68	2.65	3.54

[a]Neutron and gamma shield thicknesses are given in inches.

[b]Results in a dose rate of approximately 9 mrem/h gamma and 1 mrem/h neutron at 10 ft from the cask centerline.

[c]Assumes inner steel shell, outer steel shell, and outside steel liner thicknesses of 1.5, 2.0, and 0.75 in. for Pb cask; 0.75, 2.0, and 0.75 in. for U-metal cask; and 0.375, 0.375, and 0.75 in. for Fe cask.

[d]Neutron shield assumed to be a water/ethylene glycol mixture.

Table 9.10. Optimized neutron and gamma shielding requirements[a] for Pb, Fe, and U-metal spent fuel shipping casks as a function of the number of 10-year-old PWR assemblies in the cask[b,c]

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield[d]	G-shield	N-shield[d]	G-shield	N-shield[d]
1	2.98	3.69	7.90	3.61	1.90	2.61
4	3.39	3.83	8.52	3.66	2.13	2.86
8	3.61	3.93	8.85	3.66	2.25	3.06
12	3.78	4.04	9.10	3.68	2.34	3.26
18	3.90	4.02	9.27	3.68	2.41	3.33
21	3.97	4.13	9.39	3.71	2.45	3.42
26	4.04	4.25	9.49	3.75	2.49	3.47

[a]Neutron and gamma shield thicknesses are given in inches.

[b]Results in a dose rate of about 9 mrem/h gamma and 1 mrem/h neutron at 10 ft from the cask centerline.

[c]Assumes inner steel shell, outer steel shell, and outside steel liner thicknesses of 1.5, 2.0, and 0.75 in. for Pb cask; 0.75, 2.0, and 0.75 in. for U-metal cask; and 0.375, 0.375, and 0.75 in. for Fe cask.

[d]Neutron shield assumed to be a water/ethylene glycol mixture.

Based on the available reference material, the following sections provide an overview description of the well-known codes, cross-section libraries, and a modular code system that illustrates one approach to performing nuclear fuel facility and package design analyses using fixed analytic sequences.

9.5.1.1 Codes for Radiation Source Generation

Table 9.11 provides a list of widely used codes suitable for generating spent fuel radiation source terms. All the codes are available from the RSICC computer code collection.

The most widely known code in the table is the original ORIGEN code,[34] which serves as a basis for several of the other codes: ORIGEN-JR,[35] KORIGEN,[36] ORIGEN2,[37] and ORIGEN-S.[38] These four updated codes all provide significant improvements over the original version. These improvements are well documented in the respective references for each updated code. Of the four updated ORIGEN codes, the U.S. codes ORIGEN2 and ORIGEN-S appear best for applications involving spent fuel HLW isotopic characterization. A recent overview report[39] compares and contrasts these two codes from a standpoint of application to HLW isotopic characterization.

The ORIGEN2 code is the most popular of the updated versions of the ORIGEN codes to be used in a stand-alone fashion. The library data and radiation source-term evaluation offer a significant improvement to the ORIGEN code. ORIGEN2 provides the gamma spectra in an 18-energy-group format that matches the group format of the 22n-21γ FCXSEC cross-section library[40] for all but the last few high-energy groups (typically unimportant for spent fuel). However, only the neutron source strength is provided by ORIGEN2. Thus, the analyst must generate a neutron spectrum in

Table 9.11. Well-known computer programs utilized for generation of radiation source terms

Code (RSICC CCC No.)[a]/ developer	Description and comments	Refs.
EPRI-CINDER (CCC-309) Los Alamos National Laboratory	Point-depletion code for computing actinide and fission-product atom densities. Solution via Bateman equations. Auxiliary codes required for generating radiation source spectra and strengths. Other code versions are CINDER2, CINDER3, CINDER7, and CINDER 10. Data libraries and availability vary between versions.	42, 43
ORIGEN (CCC-217) Chemical Technology Division Oak Ridge National Laboratory	Point-isotope generation and depletion code. Solution by matrix exponential method. Actinide, fission-product, and light element libraries available. Photon source spectra (fixed groups) and neutron source strength generated using original data and/or analytic functions.	34
ORIGEN2 (CCC-371) Chemical Technology Division Oak Ridge National Laboratory	Significant update of the ORIGEN code to remove deficiencies, improve input features, provide new and better data libraries (actinide, fission product, and light elements). Photon source spectra (fixed groups) and neutron source strength improved over ORIGEN code. Well documented and widely used.	37
ORIGEN-S and ORIGEN-ARP (CCC-545) Nuclear Engineering Applications Section Oak Ridge National Laboratory	Significantly updated version of the ORIGEN code developed for the SCALE system. Decay data and photon data same as for ORIGEN2. Radiation source (n and γ) strength and spectra provided in user-specified or default multigroup energy structure. Well documented. Full-screen PC input processor and interpolation on data base of burnup-dependent cross-section libraries provided in ORIGEN-ARP.	38, 59
RIBD-II (CCC-137) Pacific Northwest Laboratories	A subroutine within the ISOSHLD II and III point-kernel codes. Performs a reactor point-depletion analysis to produce gamma source spectra for fission products. Fission-product data libraries available for generic thermal and fast reactors.	45

[a]Codes identified with a RSICC CCC number are available by writing to Radiation Safety Information Computational Center, P.O. Box 2008, Oak Ridge, Tennessee 37831-6362, and identifying the CCC number.

the required group structure when using ORIGEN2. Also, if using a gamma cross-section library having a group structure different from that for which the source is provided, the analyst must adjust or interpolate the ORIGEN2 gamma source spectrum. ORIGEN2 is relatively easy to use and has several built-in data libraries for typical LWR and fast-breeder-reactor (FBR) applications. However, this makes it difficult to use for other fuels such as those used in research reactors.

The ORIGEN2 code was used as the basis of the development of an extensive data base of radioactive waste characteristics entitled the Characteristics Data Base (CDB).[41] The waste characteristics include integral radiation and heat generation sources, as well as detailed isotopic information for light-water reactor (LWR) spent fuel, immobilized HLW, non-LWR spent fuel, and miscellaneous wastes (such as wastes arising from the decommissioning of nuclear reactors and other facilities handling radioactive material).

The ORIGEN-S code was designed to interface with neutronic codes that could be used to prepare appropriate burnup-dependent cross sections (see Appendix E). ORIGEN-S provides complete neutron and gamma source spectra in any multi-energy group format. Thus, the shielding analyst is provided with the flexibility to select a multigroup cross-section library without needing to interpolate from one fixed group to another. ORIGEN-S produces the separate spectrum for (α,n) and spontaneous fission neutrons, and the total neutron spectrum. Stand-alone use of ORIGEN-S has recently been facilitated by the development of ORIGEN-ARP, which provides a full-screen PC input processor and automatic interpolation on a data base of burnup-dependent, cross-section libraries (see Appendix E).

The CINDER series of codes represent the major alternative to the ORIGEN codes in the United States. As with the ORIGEN-type codes, there are several updated versions of the CINDER code[42] that have been developed and are currently in use at Los Alamos National Laboratory (LANL). Of these, EPRI-CINDER[43] is the only one publicly available from RSICC. For the CINDER codes, LANL uses an auxiliary code called SPEC5 (not available from RSICC) to evaluate the spent fuel source spectra. The user may manually calculate or write coding to compute the combination of CINDER nuclide densities and SPEC5 spectra, account for significant bremsstrahlung from UO_2, and add gamma rays from spontaneous fission or (α,*n*) reactions as needed. The neutron spectrum may be produced by using the SOURCES code.[44] Another limitation of EPR-CINDER is that light-element libraries are not available. Thus, potential radiation sources arising from the activation of light elements (e.g., ^{59}Co) must be evaluated in some alternative fashion.

Although older and far more limited than the ORIGEN- or CINDER-type codes, the RIBD-II code[45] has been frequently used to define spent fuel gamma sources because it is interfaced with the point-kernel code ISOSHLD[46] to provide an easy-to-use procedure for gamma-ray source generation and shielding analysis. The RIBD routine is limited to evaluating the gamma source spectra from fission products only and requires another routine called BREMRAD[47] to evaluate the bremsstrahlung source spectra.

In a more complex, yet more complete, fashion, the SAS2 shielding sequence of SCALE (see ref. 48 and Appendix E) uses ORIGEN-S to generate radiation source spectra for subsequent input to a radiation transport module.

9.5.1.2 Codes for Radiation Dose Rate Evaluation

This section provides a discussion of relevant codes that implement the three basic techniques for evaluating dose rates from radiation sources—point-kernel codes, discrete-ordinates codes, and Monte Carlo codes.

9.5.1.2.1 Point-kernel codes. Point-kernel codes provide only approximate evaluations of the primary gamma-ray dose rate from a gamma source. However, these codes are inexpensive, computationally fast, and far less cumbersome or complex relative to discrete-ordinates or Monte Carlo codes. They are well suited for scoping studies.

Table 9.12 provides a list of three of the more popular point-kernel codes. Of these, the QAD family of codes have surely enjoyed the greatest popularity and use. Originally developed at LANL in the 1960s, it has been successfully updated by a variety of users. The latest and best version of QAD is QAD-CGGP[49] which features the flexible, 3-D combinatorial geometry package (CG), the standard buildup factor data of ANS-6.4.3,[50] and the geometric progression (GP) fitting function for the buildup factor data. These latter two features represent a substantial improvement over the basic buildup factor data and interpolation scheme currently used in most other codes. The improvement is most evident for shield materials of low- or very high-Z number and/or low (<0.5 MeV) photons. The CG geometry feature of QAD-CG and QAD-CGGP is attractive because the geometry input description can be easily interchanged for use in combinatorial geometry versions of MORSE (see paragraph on Monte Carlo Codes in this section).

Table 9.12. Computer programs based on point-kernel techniques used for radiation dose evaluation

Code (RSICC CCC No.)[a]/ developer	Description and comments	Refs.
ISOSHLD Pacific Northwest Laboratories	Code uses direct point-kernel techniques to generate gamma doses for common geometric models. Source may be input directly or calculated via the RIBD routine (see Table 9.11). An extremely user-friendly, for-sale version called MICROSHIELD is available for the PC (see ref. 51). A revised version called ISO-PC is available as CCC-636.	46
PATH GA Technologies, Inc.	Proprietary point-kernel code available from General Atomics. Buildup factor specification can vary with source and dose point. Claims to be "fully validated and suitable for nuclear licensing applications."	52
QAD (CCC-48, 307, 346, 396, 401, 448, 493, 645) Original code from Los Alamos National Laboratory	The QAD family of codes makes up seven different code packages in the RSICC code collection. All use direct point-kernel techniques and differ principally in the available geometry package, source integration method, buildup factor interpolation scheme, and ease of input. Latest buildup factor data and attractive geometry in new QAD-CGGP (CCC-493). Most widely used of the point-kernel codes.	49

[a]Codes identified with a RSICC CCC number are available by writing: Radiation Safety Information Computational Center, P.O. Box 2008, Oak Ridge, Tennessee 37831-6362, and identifying the CCC number.

The ISOSHLD code has a nice feature of generating an irradiated fission-product source using the RIBD routine. Also, an extremely user-friendly version of ISOSHLD (called MICROSHIELD) has been developed in a proprietary package (see ref. 51). The other codes in the table also have attractive, but less noticeable, features that distinguish them from QAD-CGGP in terms of the assessment criteria. It might be hypothesized that companies offering for-sale codes such as MICROSHIELD or PATH[52], which see their users as valuable for-profit customers, may have a more systematic approach to quality assurance criteria—maintenance, ease of use, and validation.

Note that the new buildup factor data and GP fitting functions could be rather easily added to any of the point-kernel codes mentioned in this section. Many, if not all, of these codes will probably be updated at some time once the advantage of the new data and fitting function is realized.

9.5.1.2.2 Discrete-ordinates codes. The discrete-ordinates codes provide a numerical solution to the Boltzmann transport equation and, as such, are more appropriate for general applications than point-kernel or other approximate codes. Neutrons and photons (including photons from neutron capture) can be treated simultaneously or separately without any real restrictions. However, the added complexity of these codes requires greater computational resources and user involvement. Table 9.13 lists a number of popular discrete-ordinates transport codes and auxiliary codes that facilitate accurate radiation dose evaluations. The table includes three 1-D discrete-ordinates codes (ANISN,[53] ONEDANT,[54] and XSDRNPM[55]), two 2-D discrete-ordinates codes (DOT[56] series and TWODANT[57]), and three auxiliary codes for use in evaluating doses at point detectors. Geometry requirements and/or level of desired computational effort typically dictate the selection of a 1-D or 2-D code. Often a shield configuration can be reasonably approximated in one dimension (plane, cylinder, or sphere), and the 1-D programs can combine the accuracy of discrete ordinates with the near speed of point-kernel techniques.

The 1-D ANISN code in Table 9.13 is probably the most widely used radiation shielding code (point-kernel or otherwise) both in the United States and abroad. The code has been updated and maintained since its development in the late 1960s. Using the numerical solution techniques of ANISN, the XSDRN code[58] evolved from its initial release in 1969 to the present version, called XSDRNPM-S,[55] with the following added features: (1) solutions using double-precision flux arrays to circumvent potential convergence difficulties; (2) more user-friendly input (availability of parameter default values, automatic generation of appropriate angular quadrature, etc.); (3) increased flexibility in the input/output and processing of multigroup cross-section data; and (4) inclusion within a well-maintained modular code system called SCALE[59] (see Appendix E).

The ONEDANT code[54] is the third 1-D code noted in Table 9.13. The code is much newer than ANISN, XSDRNPM, and the older LANL code called ONETRAN.[57] For deep-penetration problems (common to spent fuel shielding applications), ONEDANT appears to be more restrictive because it uses only the standard flux extrapolation technique with a simple fix-up if nonphysical solutions are encountered. ANISN and XSDRNPM-S include enhanced fix-up techniques which ensure nonphysical solutions are not encountered. Although the various flux models give nearly the same solution in the limit of very small-mesh intervals, the enhanced fix-up method generally retains a better accuracy for larger-mesh intervals than does the standard flux extrapolation technique. Thus, a user interested in using mesh sizes larger than required by standard technique would find ANISN and XSDRNPM to be more attractive. References 56, 58, and 60 present good discussions of various flux models. One advantage of ONEDANT is that it does provide a diffusion acceleration technique not available in ANISN or XSDRNPM. Although this feature

Table 9.13. Computer programs based on discrete-ordinates techniques utilized for radiation dose evaluation

Code (RSICC CCC No.[a])/ developer	Description and comments	Refs.
ANISN (CCC-82,253-255,314,514) Oak Ridge National Laboratory	General 1-D discrete-ordinates coupled neutron-gamma radiation transport code. Most popular version is ANISN-ORNL (CCC-254). Flux or dose rates at a detector site can be evaluated.	53
XSDRNPM (CCC-545) Oak Ridge National Laboratory	1-D discrete-ordinates coupled neutron/gamma-ray transport code based on ANISN. Extends ANISN capabilities to provide user-friendly features, automatic quadrature generation, and flexibility in weighing cross sections. Easily coupled to XSDOSE for doses exterior to shield.	55, 58
ONEDANT (CCC-547) Los Alamos National Laboratory	General 1-D discrete-ordinates coupled neutron-gamma radiation transport code. Modular program developed to be very user friendly. Fluxes and/or dose rates provided at detector points.	54
DORT (CCC-543) Oak Ridge National Laboratory	General 2-D discrete-ordinates coupled neutron-gamma radiation transport code. Earlier versions are obsolete. DOMINO II (PSR-162) couples DOT IV to the Monte Carlo MORSE-CG code (CCC-203). Fluxes and dose rates calculated. Excellent documentation of theory and techniques.	56, 60, 61
TWODANT (CCC-547) Los Alamos National Laboratory	General 2-D discrete-ordinates coupled neutron-gamma radiation transport code. TWODANT is basically the ONEDANT package with the 1-D SOLVER module replaced with a 2-D SOLVER module.	57
SPACETRAN (CCC-120) Oak Ridge National Laboratory	Evaluates dose rate for detectors at various distances from the surface of a cylinder. Integrates ANISN leakage or DOT 3.5 flux data. Eliminates potential ray effects in air or void outside a cylinder. Not accurate for detector points near the cylindrical surface.	63
FALSTF (CCC-650) Oak Ridge National Laboratory	Calculates dose rates exterior to a shield based on DORT calculated fluxes in cylindrical geometry. Doses evaluated as sum of last flight contributions from shield regions. Eliminates potential ray effects in air or void outside cylinder.	62
XSDOSE (CCC-545) Oak Ridge National Laboratory	Used in conjunction with XSDRNPM (or ANISN) to compute the neutron/photon flux and the resulting dose rate at various points outside a finite cylinder, sphere, rectangular slab, or circular disc. Uses direct line-of-sight integration of surface angular flux over the surface. Eliminates potential ray effects from discrete-ordinates outside shield. Extremely easy to use.	64

[a]Codes identified with a RSICC CCC number are available by writing: Radiation Safety Information Computational Center, P.O. Box 2008, Oak Ridge, Tennessee 37831-6362, and identifying the CCC number.

speeds up the convergence process, the potential increased efficiency is of marginal value for 1-D problems.

For problems requiring 2-D discrete-ordinates shielding analyses, the DOT code series have become the international standard. The latest version, DORT,[61] represents a significant advancement in computing efficiency and speed; however, most problems of reasonable size still require substantial computer resources. The DOT codes have been developed primarily for radiation shielding analysis, while TWOTRAN and TWODANT have been developed in a reactor/core physics environment. This difference in emphasis explains why DORT is typically selected where shielding calculations are of prime import. Like ONEDANT, the TWOTRAN and TWODANT codes allow only the standard flux extrapolation technique. The incorporation of a variety of flux model options reflects the DORT emphasis on allowing accurate solution of complex shielding problems. Also, the DORT manual provides an excellent explanation of the basic theory and numerical techniques employed in the code.

Table 9.13 includes three auxiliary codes that were developed to provide an easy means of accurately evaluating the flux or dose at a point exterior to a shield. For problems where doses are required exterior to a shield in a low-scattering medium (air, void, etc.), extension of the discrete-ordinates spatial mesh out into the exterior medium is often unattractive because

1. a penalty in computing time is paid for the extra spatial mesh (typically, a fine spatial mesh and angular quadrature are needed for curvilinear geometries),

2. for 1-D problems, there is no good way of accounting for the finite dimensions of the shield from which the radiation leaks, and

3. ray effects in multidimensional problems are very difficult to alleviate and can yield unreliable results.

To alleviate these problems, FALSTF,[62] SPACETRAN,[63] and XSDOSE[64] have been written for use with the ANISN, DOT, and XSDRNPM codes. Although available for DOT 3.5, versions of FALSTF and SPACETRAN for the DORT code are not available from RSICC. The SPACETRAN code is computationally more efficient than FALSTF, but it is inaccurate close to the shield and can be unreliable if an inappropriate spatial mesh or an insufficient angular quadrature is used. The XSDOSE code has the best numerical techniques of all these codes, but is currently limited to use with the 1-D ANISN or XSDRNPM codes. The techniques used in XSDOSE eliminate the difficulties inherent in the 1-D SPACETRAN method.

9.5.1.2.3. Monte Carlo codes. Table 9.14 lists two popular Monte Carlo codes that are widely used in the United States for performing radiation shielding analyses. Historically, a number of general-purpose Monte Carlo codes were developed in the late 1960s and early 1970s. The SAM-CE[65] code was perhaps the leader in the creation of an attractive, easy-to-use geometry package (combinatorial geometry or CG), handling of point cross-section data, and implementation of various biasing techniques. However, because of its easy-to-use features, accessible ANISN-formatted cross-section data, and ready availability, the MORSE code emerged in the late 1970s as the most widely used Monte Carlo code for radiation shielding. The latest versions of MORSE (CGA[66] and SGC/S[67]) utilize multigroup cross sections, a wide variety of source and particle biasing features, and a CG package with nested array features.[68] The version of SAM-CE in RSICC is old and incomplete with regard to some routines that the developers provide for a fee.

Table 9.14. Computer programs based on Monte Carlo techniques utilized for radiation dose evaluation

Code (RSICC CCC No.[a])/ developer	Description and comments	Refs.
MCNP (CCC-660) Los Alamos National Laboratory	General-purpose Monte Carlo code for coupled neutron/photon particle transport. Capable of handling point energy and discretized cross-section data. New features for automatic generation of importance functions. Flexible geometry capabilities. Source and response estimator specification flexible and easy to use. Well-supported program with constant improvements and updated. Well documented.	70
MORSE (CCC-127,203, 258,261,277,368,394, 545,471,474) Oak Ridge National Laboratory	General-purpose multigroup Monte Carlo code for coupled neutron/photon particle transport. Most widely used of Monte Carlo codes. Latest versions from ORNL in CCC-345 and CCC-474 have a popular, easy-to-use geometry package capable of generating multiple arrays within arrays. Flexibility in specifying source, response estimator, and biasing mode easy via wide variety of user-supplied routines.	66, 67

[a]Codes identified with a RSICC CCC number are available by writing: Radiation Safety Information Computational Center, P.O. Box 2008, Oak Ridge, Tennessee 37831-6362, and identifying the CCC number.

Like SAM-CE, updates of other codes provided to RSICC in the early 1970s are probably available from private companies or consultants.[69]

Although the MORSE codes were once the most widely used Monte Carlo shielding codes, the MCNP code[70] developed at LANL is now the most popular. Once regarded as a highly specialized code that was difficult to use, the MCNP developers have made a concerted effort to retain the sophisticated attributes of the code and still provide an easy-to-use and readily acceptable tool. The main areas of sophistication involve the use of point-energy cross-section data and include the development of user-friendly biasing schemes. The biasing schemes are an attempt to reduce the need for user expertise in analyzing a problem; however, a good deal of user savvy is still required for deep-penetration and streaming problems. Another potential disadvantage with MCNP is its inability to perform adjoint analyses.

9.5.1.3 Cross-Section Data Libraries

This section provides a review of the various types of available cross-section libraries and discusses those currently used for spent fuel shielding analysis and those that have potential for future use. The primary differences between various types of cross-section sets will be clarified. The "best" multigroup data library will vary from application to application, even more so than with radiation transport codes. This section notes libraries that have been widely used and those that need further assessment for spent fuel shielding applications.

Broad-group libraries have traditionally been developed for general uses and, whether generated from a fine-group library or developed directly from evaluated data, are typically application-dependent. Some of the older broad-group libraries generated directly into a discrete-ordinates format are shown in Table 9.15.

Typically these libraries were developed and used successfully for a given project and results obtained with the data were published. The first library, CASK,[71] was developed for depleted uranium shipping casks with a water-filled cavity. The energy grouping was determined based on typical spent fuel spectra. The data source for this library is quite old and the resonance correction for ^{238}U is inadequate if the subcritical multiplication source is important to the dose. However, CASK has been one of the most widely used (for all type applications) ANISN-formatted libraries. The second library, FEWG1,[72] was developed for radiation transport through concrete and air. The work was sponsored by the Defense Nuclear Agency (DNA), and the group structure was developed for applications with source spectra from nuclear weapons. However, the library may be useful for analysis of concrete storage casks. The library also has an extensive selection of kerma response functions for various materials. The CLAW-IV library[73] was developed for shielding analyses related to weapons applications. Although a potential candidate for spent fuel shielding applications, it appears to have too few gamma groups in the energy range of importance to spent fuel applications.

Since the late 1970s, the more attractive procedure for developing a coupled broad-group library has been to process (resonance shielding and temperature correction) and collapse a fine-group, pseudo-application-independent library to create a production, application-dependent library. A collection of these libraries is shown in Table 9.16, all of which are ANISN-formatted. The BUGLE-80[74] and SAILOR[75] libraries are nearly identical. The primary difference is the

Table 9.15. Some broad-group libraries in discrete-ordinates format developed for specific applications

Library	Contributor	Energy groups	Processor	Source	Applications/ (No. elements)
DLC-23/ CASK	ORNL	22n, 18g	SUPERTOG, POPOP4 (Collapse)	ENDF/B-II, POPLIB	Shipping Casks/ (29)
DLC-31/ FEWG1	ORNL/ DNA	37n, 21g	AMPX	ENDF/B-IV, DNALIB	Concrete, Air/ (43)
DLC-36/ CLAW-IV	LANL	30n, 12g	NJOY	ENDF/B-IV	General/(48)
CCC-545[a]/ 27N-18COUPLE	ORNL	27n, 18g	AMPX	ENDF/B-IV	General/(83)

[a]Codes identified with a RSICC CCC number are available by writing: Radiation Safety Information Computational Center, P.O. Box 2008, Oak Ridge, Tennessee 37831-6362, and identifying the CCC number.

Table 9.16. Some broad-group application-dependent libraries developed from fine-group libraries in "flexible" format

Library	Contributor	Energy groups	Master library source	Applications/ (No. elements)
DLC-75/ BUGLE-80	ORNL, ANS-6.1.2	47n,20g	DLC-41/VITAMIN-C	Standard for concrete, LWR shielding/(66)
DLC-185/ BUGLE-96	ORNL	47n,20g	VITAMIN-B6	LWR shielding and pressure vessel dosimetry
DLC-76/ SAILOR	SAI, ORNL	47n,20g	DLC-41/VITAMIN-C	BWR and PWR radiation transport analyses/(58)
DLC-85/ FCXSEC	ORNL	22n,21g	DLC-41/VITAMIN-C	Fuel cycle shielding analyses/(Many)

availability in SAILOR of mixed macroscopic cross sections for PWR and BWR reactor applications.

The BUGLE-80 library and its parent VITAMIN-C are listed in American National Standards Institute (ANSI)/ANS-6.1.2/1989 as suitable cross-section sets for nuclear radiation protection calculations for nuclear power plants. The standard lists the processing procedures and verification efforts required to be included in the standard. The group structure in the library is appropriate for applications involving spent fuel radiation sources. However, testing of the BUGLE-80 library was done primarily for concrete shields, and resonance processing was not done on nonconcrete nuclides. The weighing spectrum used in the collapse from VITAMIN-C was that of a concrete medium. Thus, the library could be applied to the analysis of concrete storage casks for spent fuel, but its use for nonconcrete shielded applications needs further testing.

The BUGLE-96 cross sections[76-77] are intended for use in LWR shielding and pressure vessel dosimetry applications. The multigroup data have been collapsed from the ENDF/B-VI-based VITAMIN-B6 library, and in some cases self-shielded, using flux spectra typical of PWR and BWR reactor models. Flux spectra from five specific locations within these models were used, corresponding to (1) off-center in a BWR core region, (2) off-center in a PWR core region, (3) the downcomer region in a PWR model, (4) within the pressure vessel at a depth of one-fourth the total thickness, and (5) within the concrete shield surrounding a PWR reactor vessel. The concrete-spectrum-weighted cross sections have been shown to be generally applicable to a wide range of shielding problems.

The FCXSEC library[40] was developed for fuel cycle shielding analyses. A generic fusion-fission-1/E-Maxwellian spectrum was used to collapse from the VITAMIN-C group structure. Resonance self-shielding was performed for three background cross sections (composition dependent)—0.1, 1000, and 10^8 barns/atom. Macroscopic cross sections are available with appropriate resonance processing for several mixtures routinely used in spent fuel shielding analyses.

As noted above, the ANISN format or, more generally, the discrete-ordinates format, is a "working" format; that is, the radiation transport codes read these formats directly. No further resonance or temperature correction is possible. A newer approach generates a broad-group library with a selected weighing spectrum, but retains the "flexible" AMPX-like format that provides the neutron resonance information. The shielding libraries provided with the SCALE system were the first to use this approach. The libraries are provided in the AMPX "master" format, and the SCALE sequences use BONAMI and NITAWL modules to generate the resonance and temperature correction for each particular problem and alter the format from a master to an AMPX "working" format. Libraries in SCALE with no resonance information (e.g., CASK 22n-18γ) follow the same procedure, but no actual processing takes place; that is, BONAMI and NITAWL are merely used to change the master format to a working format. Installations that have the capability to carry out criticality analyses have resonance processing codes available because resonance processing is of extreme importance in evaluating an accurate neutron multiplication factor.

To date, there are very few libraries that follow the above approach. Of these, the 27n-18γ library in SCALE is the most prominent. The neutron data were collapsed from the CSRL library, and the gamma data were created directly using various AMPX modules. The library group structure and weighing function were selected to be appropriate for spent fuel shielding applications. The large number (13) of thermal neutron groups can increase the computer processing time of a discrete-ordinates shielding analysis yet be very beneficial if photons from neutron capture or source multiplication from fission are important.

The only other broad-group libraries available in the flexible format are the MATXS libraries from LANL (see Table 9.17). However, the MATXS1 and MATXS5 libraries contain no resonance information and are simply MATXS-formatted versions of CLAW-IV. The only library of potential interest is the DLC-116/MATXS6 library. However, there appear to be too many neutron groups to be used for production work.

Table 9.17. Some libraries in "MATXS" format available from RSICC

Library	Source	Energy groups	Processor/ (format)	Source	Applications/ (No. elements)
DLC-114/ MATXS1	LANL	30n, 12g	NJOY-II (MATXS)	ENDF/B-IV, LANL	MATXS Equivalent of DLC-36/CLAW-IV/(64)
DLC-115/ MATXS5	LANL	30n, 12g	NJOY-II (MATXS)	ENDF/B-V, LANL	ENDF/B-V equivalent of MATSX1/(87)
DLC-116/ MATXS6	LANL	80n, 24g	NJOY-II (MATXS)	ENDF/B-V	Fast reactor shielding, fusion/(91)
DLC-117/ MATXS7	LANL	69n	NJOY-II (MATXS)	ENDF/B-V	EPRI-CPM group structure, PWR studies/(80)

9.5.2 Comparison of Code Capabilities

9.5.2.1 Geometry

Generally speaking, point-kernel codes and Monte Carlo codes can treat 3-D geometry easily with little or no limitation in developing calculational models. On the other hand, discrete-ordinates codes can efficiently treat packages having a 1-D or 2-D geometry. Although 3-D discrete-ordinates codes have been developed and have undergone extensive verification testing, they are generally impractical for application to large spent-fuel-bearing packages. The 2-D geometry models suffer from a lack of flexibility in that they have fixed r-z, x-y, or r-theta geometries and do not allow a mixing of various geometries such as a model that uses x-y geometry inside of cavity and r-theta geometry for a cask body. Additionally, it is difficult to express correctly holes or orifices penetrating the cask body or lid for drainage or ventilation.

For the ends of a cask, 1-D discrete-ordinates codes are not recommended other than in initial survey calculations, unless the 1-D models are validated with more rigorous multidimensional models.

9.5.2.2 Buildup Factors and Cross-Section Data

Point-kernel codes use dose buildup factors that are determined from experiments and a large number of transport calculations using flux-to-dose conversion factors, normally found in ANSI/ANS 6.1.1-1977. If the dose evaluation using new flux-to-dose conversion factors from ANSI/ANS 6.1.1-1991 is required, a reevaluation of the dose buildup factors must be performed. This situation has been addressed[78] by a revised version of QAD-CGGP called QAD-CGGP2.[79]

Discrete-ordinates codes and the Monte Carlo code MORSE use multigroup cross sections. Appropriate weighing functions, adequate group structure, and proper resonance treatment are application-specific considerations that are crucial to preparing multigroup cross sections that give accurate results for a set of applications. Selection of a multigroup cross-section set is discussed in Sect. 9.5.4.3. The popular Monte Carlo code MCNP has largely overcome these problems by the use of point cross sections.

9.5.2.3 Computational Time

Point-kernel and 1-D discrete-ordinates codes, if applicable to a given problem, are the most efficient means of solution. If additional geometric details are needed, 2-D discrete-ordinates codes are generally the next most efficient, followed by Monte Carlo codes. If the case is complicated enough to use fine energy structures and/or a higher angular quadrature set and/or finer space meshes, the difference in efficiency between discrete-ordinates and Monte Carlo codes becomes unclear.

9.5.2.4 Calculated Dose Rates

The point-kernel codes give both energy-dependent and the energy-integrated dose rates. However, only the energy-integrated dose rates should be used since the energy-dependent dose rates are

usually inappropriate. The discrete-ordinates and Monte Carlo codes can present reliable dose rates for practically any spatial location and energy distribution.

The dose rates obtained by Monte Carlo codes have statistical uncertainties. Empirically, the statistical uncertainty should be less than 10% overall, and less than 5% when using point detectors to obtain reliable results.

Even though the statistical uncertainties are reasonably low, Monte Carlo codes sometimes can give poor results, especially when using point detectors. When this happens, it is usually the result of the use of improper bias techniques to accelerate the calculation.

9.5.2.5 Detector Location

Point-kernel codes give results at a point identified by the user. Therefore, it is necessary to determine the detector locations before carrying out the calculation with a limited knowledge of the important regions. Thus, some skill is necessary to select the detector points.

A Monte Carlo code has three kinds of detectors: point, surface, and volume. With respect to the point detector locations, the situation is the same as with point-kernel codes. With surface and volume detectors, it is necessary to judiciously select the detector sizes to obtain results with reasonable statistical uncertainty.

On the other hand, discrete-ordinates codes give the dose rate distribution for the entire geometry automatically. Hence, it is easy to determine the location where the maximum dose rate occurs.

9.5.2.6 Limitations

If a calculational model consists of several layers of shield materials, the selection of buildup factors is important to get reliable results using a point-kernel code. A rule of thumb is to use the buildup factor for the final layer if that layer is several mean-free paths thick, or to use the buildup factor for the dominant shield layer if the outer layers are only a few mean-free paths thick.

The point-kernel code cannot estimate radiation streaming correctly due to the underlying approximations in the method. When holes or orifices penetrating the cask body or lid are present, the point-kernel results should be validated by the use of a more rigorous method. It is recommended that at least one discrete-ordinates or Monte Carlo calculation be performed to establish confidence in the point-kernel code or provide correction factors when the model has multilayer shields and/or holes or orifices between the source and detector.

The presence of ray effects can be a serious problem when using multidimensional discrete-ordinates codes. Special techniques must be employed in order to correctly calculate results at external detector points. One technique is the so-called last-flight estimator, which calculates the dose contribution at the detector due to particles scattering from all spatial meshes in the system (FALSTF). Another technique is to calculate the dose rate at each detector from the angular leakage on the outside surface of the shield (SPACETRAN and XSDOSE).

Evaluation of streaming from narrow and long holes or orifices penetrating the cask body or lid is a difficult task for a discrete-ordinates code. Even though the holes in a cask are correctly expressed by a 2-D model, a higher angular quadrature set or specialized angular quadrature set must be used to evaluate a streaming component correctly.

9.5.2.7 Expertise Needed

The items needing expertise for the various calculational techniques are summarized below:

1. for point-kernel codes the selection of buildup factor and size of source mesh;
2. for discrete-ordinates codes the selection of space mesh, angular quadrature set, and polynomial expansion order and approximation of the actual model; and
3. for Monte Carlo codes the bias technique selection, and bias generation.

Detailed discussions of these items are given in Sect. 9.5.4.2.

9.5.3 Validation of Calculational Techniques

There is a continuing search for means to test the various calculational methods used for radiation shielding analyses. Uncertainties in calculated results arise from several areas, including the properties of the source, the cross sections of materials involved in the calculation, modeling limitations, calculation method approximations, and statistical uncertainty of Monte Carlo results. Two methods are used to assess or verify calculations—experimental benchmark and calculational benchmark.

9.5.3.1 Comparison with Experiment

The best scheme to assess the technical validity of a code is to compare analytic results from various codes and/or data libraries against experimental results designed for the application. In essence, this provides a validation of the software for the intended application. Comparison with experiment can also reveal where further fundamental measurements are required. Experiments are of two types—integral and benchmark.

Integral experiments are so named because they measure integral or macroscopic properties of a system. Benchmark experiments represent an elite class of integral experiments and must satisfy certain criteria in addition to the functional intent of the experiment. To be considered a benchmark, the experiment must have well-defined sources and detector responses, accurately described geometry and material compositions, known sources of uncertainty, and complete documentation.

Few integral-type experiments exist for spent fuel shielding applications,[80-91] and no benchmark experiment exists. Some integral experiments exist for cask-type geometries, but they vary greatly in quality and, for all but one,[83] the source is actual spent fuel. The use of spent fuel for the radiation source adds greatly to the uncertainty of any related analysis because typically the actual source spectra or pertinent isotopes are not measured, but must be calculated via ORIGEN-type

analyses. These experiments also vary widely in their documentation in that an accurate analysis of the experimental model would be difficult based on the information publicly reported. However, these experiments may be a valuable source of information for assessing the total software package (data and code) employed by an analyst.

So far, very few analyses of these types of measurements have been reported. The Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency Committee on Reactor Physics sponsored a working group for intercomparison of computational methods for spent fuel transport casks.[92–94] In the working group, the evaluations of TN-12 cask loading 12 PWR fuel assemblies were performed by several organizations. Reference 95 shows the comparison using several codes such as SAS2H, SAS1 (XSDRNPM), DORT, SAS4 (MORSE-SGC), and MCNP with the measurements of four different types of casks (TN-24, MC-10, CASTOR, VSC).

9.5.3.2 Comparative Analysis

As noted above, there is a distinct lack of high-quality shielding experiments available to assess the software packages. Given the lack of experimental data available to verify the physics, an alternative is to compare the analytic results obtained from analyses of a variety of independently developed codes and data libraries. Although the comparison does not allow a conclusion as to the "correct" answer, it allows analysts to (1) study the effect and/or uncertainty of using various methods, codes, and data for a particular analysis, (2) study the theory or software techniques to determine the source of difficulty with deviant results obtained for an application, and (3) investigate the variance in calculated results caused by difficulties in input and/or approximations used by experienced independent users.

The OECD working group also performed several comparative analyses for some simplified cask models. The U.S. delegate reported the detailed comparisons of QAD-CGGP, SAS1 (XSDRNPM), DORT, SAS4 (MORSE-SGC), and MCNP, including the comparison of multigroup cross-section libraries.[14]

9.5.4 Techniques for Accurate Modeling

9.5.4.1 Source Definition

9.5.4.1.1 Consideration of fission sources. As the effective multiplication factor (k_{eff}) of a system increases toward a value of 1.0, the importance of the neutrons emitted from fission reactions becomes increasingly important to a shielding analyst. The best way to consider this fission source is to calculate it as part of a shielding analysis using the proper cross-section library such as the 27n-18g SCALE library. Usually it is difficult to model the inside of a cask cavity appropriately from a criticality aspect in a shielding calculational model, except with a Monte Carlo method. A second approach consists of modifying the fixed spent fuel neutron source by the factor $1/(1-k_{eff})$. Should this second approach be used, care should be taken to turn off or not allow any fission events in the system.

Table 9.18 shows an example of the differences between the two methods above. As the latter method assumes a flat fission-source distribution, it always overestimates neutron dose, especially

Table 9.18. Neutron dose rate (mrem/h) for OECD problems 1a and 1c

	Fission accounted for directly		Fission accounted for via $1/(1-k_{eff})$	
	CASK[b]	SCALE[b]	CASK[c]	SCALE[c]
Wet cavity (problem 1c)	29.7 (k_{eff} = 0.845)	58.1 (k_{eff} = 0.924)	76.4 (k_{eff} = 0.924)	77.9 (k_{eff} = 0.924)
Dry cavity (problem 1a)	68.7 (k_{eff} = 0.181)	72.2 (k_{eff} = 0.207)	70.9 (k_{eff} = 0.207)	72.3 (k_{eff} = 0.207)

[a]All shielding analyses performed with SAS1 module of SCALE and ANSI standard flux-to-dose conversion factors.

[b]k_{eff} values obtained by the 1-D SAS1 calculation during the shielding analysis.

[c]k_{eff} values obtained by the 1-D SAS1 calculation using SCALE library in shielding analysis.

in a wet system (occurring in various accident scenarios or when a cask is removed from a pool and prior to purging). Note that the 22n-18g CASK library is not adequate for the former method because of the inappropriate resonance cross section of ^{238}U for criticality analysis.

9.5.4.1.2 Radiation source spectra. To evaluate the potential impact of neutron source spectra to dose, four otherwise identical 1-D discrete-ordinates analyses were performed using thermal-fission spectra of ^{235}U, ^{239}Pu, spontaneous fission spectra (mixing of ^{242}Cm and ^{244}Cm), and (α,n) spectra (mixing of ^{242}Cm, ^{244}Cm, and ^{238}Pu). The spontaneous fission spectra and (α,n) spectra are obtained from ORIGEN-S. The total source intensity for each calculation was identical. The radial dose rates for the OECD working group problem 1b were obtained using the SAS1 (XSDRNPM) module of SCALE with the 22n-18g CASK library. As shown in Table 9.19, there is little difference in the estimated doses except for the (α,n) spectra. For a long cooling time or thin neutron shield, the effect of (α,n) contribution to total neutron dose may be small enough to neglect. But if the cooling time is very short, or the neutron shielding is very thick, the effect of (α,n) cannot be neglected. Therefore, it is better to consider the (α,n) spectra at any time.

Many transport codes require the discrete lines for the gamma radiation source to be cast into a multi-energy group format that corresponds to the multigroup cross-section set. For spent fuel, even the point-energy codes such as MCNP and QAD-CGGP generally require the source term be cast into a multigroup format due to the large number of isotopes represented in burned fuel.

Table 9.19. Effect of neutron source spectra[a]

Spectra	Spontaneous fission	^{235}U	^{239}Pu	(α,n)
Neutron	1.0	0.948	0.976	1.120
(n,γ)	1.0	0.974	0.983	1.033
Total	1.0	0.958	0.979	1.086

[a]Normalized to the dose rate obtained by spontaneous fission spectra.

The preferred method for binning discrete gamma lines associated with each energy group is to conserve the total energy of the gammas produced rather than conserve the total number of gammas produced. Even then, this method is not always conservative. The procedure is especially prone to error when using broad energy group structures. For thick heavy shields such as a 30-cm-thick iron cask, the error can become large. For example, Table 9.20 shows the effect of gamma energy replacement from 1.0 to 0.9 or 1.1 MeV for spherical iron shields of 1 cm and 30 cm thickness. Even with the use of energy conservation, the dose rate is underestimated 40% when using 0.9 MeV, and overestimated 60% when using 1.1 MeV. In this case, the combination of the 0.9-MeV and 1.1-MeV values gives better results. When the energy structure of the cross section is different from that obtained by a source intensity calculation code, prudent calculation is necessary to ensure that original discrete energy levels are still in the appropriate energy group of the cross-section set for dominant nuclides.

The validity of the specified source is dependent on the group width, the variation of the cross section within a group, and the attenuation by the shield. It is recommended to use a fine-group structure in the energy range that corresponded to the dominant gamma energies: ^{144}Pr (2.186 MeV) for a short cooling time, ^{134}Cs (1.365 MeV), and ^{154}Eu (1.275 MeV) for longer cooling times, and ^{60}Co (1.17 and 1.33 MeV) for top and bottom direction of the cask.

9.5.4.1.3 Model of source region. A discrete-ordinates code is usually restricted to a homogenized model of the source. The technique of using a homogenized source region equal to the fuel assembly area is often not conservative for radial calculations. It is recommended to use a widely spread homogenized source region, the outer boundary of which covers the entire fuel assembly region. In most cases, spreading the source over the entire cavity region is recommended.

Comparison of the homogeneous and heterogeneous models (i.e., explicit pin by pin) is performed in ref. 33. The results indicated that there is no advantage in using a heterogeneous model. Using a heterogeneous model causes an increase in calculation time.

9.5.4.1.4 Burnup profile. Input of an appropriate axial profile of fuel assemblies should be considered. Because of the uncertainty associated with the profile and the potential for change

Table 9.20. Relative dose rates showing effect of gamma energy replacement[a]

Thickness of shield	1 cm		30 cm	
Gamma energy	Particle conserving	Energy conserving	Particle conserving	Energy conserving
1.0 MeV	1.0[b]	1.0	1.0	1.0
0.9 MeV	0.93	1.03	0.53	0.59
1.1 MeV	1.06	0.97	1.74	1.58
0.9 + 1.1 MeV[c]	1.0	1.0	1.13	1.08

[a]Normalized to the dose rate obtained using 1.0-MeV source energy.
[b]Calculations performed by MCNP.
[c]0.9 + 1.1 MeV means the dose rate is obtained from the average of 0.9 and 1.1 MeV.

between assemblies, it is difficult to use a typical burnup profile itself. Often a flat profile corresponding to the maximum burnup is used to ensure maximum dose. A more realistic method is to assume a maximum burnup for radial calculation and an average burnup for top and bottom calculations of the cask.

9.5.4.2 Code-Dependent Techniques

9.5.4.2.1 Point-kernel method. This section describes two separate effects that should be considered when using the point-kernel method for analysis of typical spent fuel cask models. As described in Sect. 9.5.1.2, selection of buildup data for multilayers can be a difficult task when using the point-kernel method. Table 9.21 gives several examples of the effect of buildup factor data selection for some typical cask models. Calculations were performed by QAD-CGGP for the radial direction of four typical cask models and compared with MCNP calculations. The calculational models used for cases 1 and 2 are the same as the OECD working group problems 1a and 1b, respectively, while cases 3 and 4 are modifications of case 1.

These results clearly show a dose rate overprediction at large detector distances. Additionally for all the cases shown, the iron buildup factors appear to be the best selection.

A second effect that should be considered when utilizing the point-kernel method is the selection of appropriate source mesh intervals. As the center of each source segment represents a source point for the calculation of the distance between a source and a detector, the source segment must

Table 9.21. Effect of buildup data for multilayer cask models

Layer of shield[b]	Buildup data	Dose rates[a]			
		Surface	1 m	2 m	10 m
Solid cast iron (case 1)	Iron	0.976	1.090	1.154	1.503
Cast iron and poly.[c] (case 2)	Iron	0.956	1.111	1.231	1.486
	Water	1.308	1.515	1.680	2.073
Lead with SS shell (case 3)	Lead	0.508	0.546	0.594	0.740
	Iron	1.172	1.260	1.370	1.708
Lead with SS shell and poly. (case 4)	Lead	0.469	0.518	0.541	0.653
	Iron	1.081	1.193	1.248	1.506
	Water	1.305	1.437	1.503	1.824

[a]Normalized to the dose rate obtained by MCNP calculation.
[b]Thickness of shielding (radial direction):
Case 1—cast iron 39 cm
Case 2—cast iron 36 cm and poly. 6 cm
Case 3—SS 4 cm, lead 14 cm, and SS 5 cm
Case 4—SS 4 cm, lead 14 cm, SS 5 cm, and poly. 6 cm.
[c]Poly. means polyethylene.

be small enough to express the self-shielding effect correctly. The effects of source segmentation variations are shown in Table 9.22 for both radial and top directions of OECD problem 1a.

As shown in Table 9.22, the user should be especially aware of source segmentation in an axial calculation. The use of biased source segments is also strongly recommended.

9.5.4.2.2 Discrete-ordinates method. This section discusses the proper selection of angular quadrature sets, cross-section polynomial expansion order, spatial mesh, and auxiliary codes when utilizing discrete-ordinates codes in the analysis of storage/transport radioactive waste packages. As for angular quadrature sets, at least S_8 or above is recommended for cask calculations. For the Legendre expansion order of scattering distributions, P_3 is normally sufficient for neutron shielding problems. For photon problems, a higher expansion order is preferable but is not always used (or available). The validity of using P_3S_8 for cask calculations has been shown in the OECD calculations.[93] Be aware that this recommendation is valid for bulk shielding only. A higher angular quadrature set and Legendre expansion order will be necessary if there is a streaming path in the cask body or lid.

Usually there are no practical limits in the use of a spatial mesh fine enough to obtain a reliable result in a 1-D calculation, but it may be necessary to compromise to some extent in a 2-D calculation. The safest and best approach for deciding the size of space mesh of a 2-D model is to repeat the 1-D calculation for the principal directions with a finer mesh until successive calculations show little or no change in the desired dose results. A rule of thumb for conservative method is to use a mesh size of 0.3 cm in lead or uranium, 0.5 cm in iron, 1.0 cm in light elements, and 3 to 5 cm in a void for a gamma shielding problem. For neutrons, the mesh size

Table 9.22. Effect of source segmentation of OECD problem 1a

	Number of source segments[a]	Surface dose rate[a]			
		Case 1[b]		Case 2[b]	
		Radial	Top	Radial	Top
Equal interval space mesh					
30 × 30 × 20[c]	1.0	0.999	0.928	0.997	0.778
90 × 91 × 90	41.0	1.0	1.0	1.0	1.0
Biased interval space mesh[d]					
10 × 21 × 18	0.21	1.004		1.003	
10 × 20 × 1	0.011		1.001		1.033

[a]Dose rates and number of source segments are relative. CPU time is proportional to the number of source segments.

[b]Cases 1 and 2 are the same as the OECD problem 1a, except the density of case 2 is 3.94 g/cm^3 (twice value of case 1).

[c]Number of source segments in the radial, axial, and azimuthal directions.

[d]Biased interval space mesh means that a finer mesh is used in regions close to a detector and a coarse mesh in regions far from a detector. For the top direction, symmetric geometry is assumed.

is 0.25 cm in a neutron shield such as polyethylene. The final criterion by which to judge the input mesh size is the flux profile. A smoothly varying flux profile is desired, that is, the groupwise fluxes for all important groups should not undergo significant changes (> 50%) from one mesh interval to the next.

As noted in Sect. 9.5.2.1, ray effects are often seen in discrete-ordinates calculations of doses more than a meter or so from a cask. The use of auxiliary codes (FALSTF, SPACETRAN, or XSDOSE) and the proper selection of the spacial mesh and angular quadrature are necessary to alleviate this problem. The use of these auxiliary codes also has the advantage of enhancing efficiency since the discrete-ordinates codes can be used to model the attenuation through the cask and a separate less time-consuming calculation is performed external to the cask.

9.5.4.2.3 Monte Carlo code. Because of the thick shielding requirements of a spent fuel cask design, the use of biasing techniques in Monte Carlo dose rate analyses is a must. The principal danger in the use of bias techniques is the distortion of the generation of the tracks, in particular by path lengths stretching when such distorted tracks are not similar to those which dominate the penetration in the physical situation. Monte Carlo calculations, if applied to spent fuel casks having long cylindrical shapes, will require different biasing objectives for radial and axial external detectors. Energy cutoff, weight cutoff, splitting and Russian roulette, and weight window (MCNP only) are popular biasing techniques. Usually a combination of these biasing techniques gives a good acceleration of the calculation.

The SAS4 module in SCALE (see Appendix E), which has been developed for dose rate analysis of shipping casks, provides an easy-to-use feature that implements the entire procedure for cross-section preparation, automatic generation of Monte Carlo biasing parameters, and a Monte Carlo calculation. In SAS4, source energy biasing, path-length stretching, splitting and Russian roulette, and collision energy biasing are automatically applied using the adjoint flux from an XSDRNPM calculation with a 1-D cask model.

9.5.4.3. Miscellaneous

9.5.4.3.1 Selection of multigroup cross-section library. Erroneous results can be produced if the cross sections employed are not suitable for the physics of the applied problem or if the user does not adequately understand the physics of the problem. A good code using poor cross sections will yield poor results. For deep penetration situations such as the cask body where the flux attenuation is over several orders of magnitude, the results are very sensitive to the cross-section data.

The weighing spectrum used to collapse the underlying point cross-section data is important. Thus, cross sections that are collapsed using a typical steel cask spectrum should be used for the calculation of a steel cask. The improper weighting actually causes cross-section changes of only a few percent, but these changes are magnified to produce large differences in the final dose because of the deep penetration. As noted in Sect. 9.4.5.1, improper energy group structure in important energy ranges can lead to erroneous results, especially for gamma rays.

Reference 33 shows the comparison of results using several cross-section libraries. Judging from this comparison, it can be said that among the publicly available broad-group libraries, the 27n-18g SCALE or 22n-18g CASK libraries are applicable for steel cask problems because the doses using these libraries show good agreement for neutron doses and slightly conservative

values for gamma doses as compared with the results of MCNP calculations using a point cross-section library.

9.5.4.3.2 Model for fin region. Some casks have cooling fins on the outer surface of the cask body to promote removal of spent fuel decay heat. If there are many small fins, it is common to use a homogenized model for the fin region. Generally, the homogenized model of a fin region produces an underestimation of the doses. This is because the homogenized model cannot treat the streaming effect correctly. The effect is especially large when large fins are homogenized.

To validate a homogenized model or introduce a correction factor, it is recommended to perform a comparative analysis of an explicit model and a homogenized model of a fin region. An alternative approach is to neglect fins from the shield to ensure a conservative result. The same is true when neutron shield rods are located inside the gamma shield. Reference 33 surveys the difference between explicit and homogenized neutron shield and fin models for the OECD problems.

9.5.4.3.3 Evaluation of secondary gamma dose. Though thermal neutron groups may be important in calculating secondary gamma doses, most shielding cross-section libraries significantly limit the number of thermal neutron groups (often to one). The contribution of secondary gamma particles to the total dose becomes large in some cases such as thick neutron shields at the outer surface of a cask. In this case, the treatment of thermal neutron groups becomes very important. Table 9.23 shows the comparison of the dose calculated by XSDRNPM using the 27n-18g SCALE library and the 22n-18g CASK library. The SCALE library has 15 thermal neutron groups, and the CASK library has only one thermal group. In this example, calculations using the CASK library underestimate the secondary gamma dose around 10%. Notice that the calculation using the SCALE library with one outer iteration shows a dose rate four times lower than the standard SCALE result.

Table 9.23. Comparison of secondary gamma dose[a] of OECD problem 1b

	Wet cavity	Dry cavity
SCALE library		
Outer iteration = 4	1.0	1.0
Outer iteration = 1	0.25	0.25
CASK library		
Outer iteration = 1	0.91	0.87

[a]Normalized to the dose obtained using SCALE library (outer iteration = 4). Fission sources are not considered in this calculation.

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10. Criticality Safety

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10.1 INTRODUCTION

Criticality safety is the practice of ensuring that adequate protection is provided against an accidental self-sustaining or divergent fission chain reaction. For packages that transport fissile material, this "adequate protection" is provided by using a design and safety-assessment philosophy that effectively eliminates the possibility of a criticality event occurring under any credible scenario. Thus the package design and allowable loading specifications must be such that the safety evaluation can demonstrate, under all credible transport conditions, that more neutrons are lost from the system (single package or array of packages) than are produced; that is, the system must always be subcritical.

Restriction of fissile mass, use of a favorable geometry (to provide enhanced neutron leakage from the package), incorporation of neutron poison materials, and moderator control are potential means of controlling the neutron balance. A detailed consideration of the many parameters that interact to influence the neutron behavior is needed to provide an adequately safe, yet efficient, package design. Neutron poisons added to package materials require special attention because their presence must be ensured under all conditions and because their incorporation may change the mechanical and/or thermal properties of host materials.

Whatever the control mechanism, an adequate margin of subcriticality must be demonstrated for both the single package in isolation and for arrays of packages. Undamaged (normal transport conditions) and damaged (subsequent to accident conditions) packages must be considered using the credible fissile material configuration and the moderator and reflector conditions that provide the maximum reactivity (i.e., highest neutron production-to-loss ratio). The number of fissile packages allowed for transport and the transport index (TI) for criticality control is obtained from the evaluation of package arrays. The evaluation of complex package designs under the prescribed conditions typically requires the use of sophisticated computer codes that incorporate either deterministic or statistical techniques to model neutron transport and predict the effective neutron multiplication factor (k_{eff}). Proper validation of the computer code and nuclear data is necessary to establish calculational biases and uncertainties. The margin of subcriticality (degree to which the calculated k_{eff} is less than 1.0) required for the ensurance of safety should include the effect of these biases and uncertainties, together with design uncertainties and an acceptable safety margin.

The goals are to present and to discuss the issues related to the criticality safety of transportation packages containing fissile material. The issues highlighted above are expanded in the sections that follow. Section 10.2 provides a discussion of the regulatory requirements specific to the

criticality safety of transportation packages. The principles of criticality safety are reviewed in Sect. 10.3. Applicable analysis methods and their validation/use in the criticality safety evaluation of a package are discussed in Sect. 10.4. Computational modeling considerations are presented in Sect. 10.5. The remaining section deals with specific issues that include: design considerations relative to packaging certification, allowance for reactivity loss due to fissile material irradiation (burnup credit), and in-situ measurement techniques for the demonstration of criticality safety.

10.2 REGULATIONS AND REQUIREMENTS

All performance standards and regulatory requirements for U.S. certification of fissile material packages are prepared by the Nuclear Regulatory Commission (NRC) and provided in Title 10 of the *Code of Federal Regulations*, Part 71 (10 CFR Part 71). The regulations related to criticality safety are the same for Type A and Type B packages and require evaluation of the damaged package configuration after the hypothetical accident. The NRC, the Department of Energy (DOE), and the Department of Transportation (DOT) work together to ensure that DOE Orders concerning off-site transportation and the applicable DOT regulations (cf. 49 CFR Part 173) are consistent with the fissile material package requirements of 10 CFR 71. For international shipments, the International Atomic Energy Agency (IAEA) sets forth performance standards and requirements in "Regulations for the Safe Transport of Radioactive Material," 1985 Edition of Safety Series No. 6 (IAEA SS6). Some differences are noted between the performance standards of the IAEA and 10 CFR Part 71 regulations, but the fissile material requirements are essentially the same. Rule changes to 10 CFR Part 71 (see 60 FR 50248, September 28, 1995) took effect April 1, 1996, and make the domestic standards and requirements more closely aligned with international standards and requirements; specifically, the new rule eliminates the three fissile class specifications of the previous regulation and changes the TI calculation to be consistent with the 1985 edition of the IAEA regulations.

10.2.1 Fissile Material Definition and Exemptions

Fissile material is defined in 10 CFR Part 71 and IAEA SS6 as material that contains any amount of the following nuclides: ^{238}Pu, ^{239}Pu, ^{241}Pu, ^{233}U, and ^{235}U. Except for ^{238}Pu, each of these nuclides meets the strict definition[1] of "fissile" in that they are capable of supporting a self-sustaining **thermal** neutron (neutron energies less than approximately 0.3 eV) chain reaction by only the accumulation of sufficient mass. Strictly speaking, ^{238}Pu is "fissionable" rather than "fissile" because it can support only a fast chain reaction under controlled conditions. Many actinides can be made to fission; however, the regulatory definition of fissile material includes only those nuclides that are either known to be available or are likely to become available for transport in significant quantities relative to criticality safety. Reference 2 (ANSI/ANS-8.15) provides guidance on the minimum amount of other actinide materials that could present a concern to criticality safety. Regardless of the regulatory specification, any future shipments with fissile or fissionable material contents greater than the limits of ANSI/ANS-8.15 should be evaluated for criticality safety per the regulations discussed below.

Several exemptions to the requirements for packages containing fissile material are allowed in 10 CFR 71 and IAEA SS6. These exemptions limit the quantities and impose conditions that are intended to ensure that the possibility of criticality during transport is incredible, even when assuming the chance accumulation of multiple packages containing fissile material and/or the destruction of packages in an accident.

Briefly, the package exemptions as listed in 10 CFR 71.53 are the following:

1. packages with 15 g or less of fissile material (as defined by the regulations);
2. packages containing uranium enriched in ^{235}U to a maximum of 1% by weight;
3. select homogeneous solutions (hydrogenous and uranyl nitrate) or mixtures of fissile material with limits specified on combinations of mass, ^{235}U enrichment, hydrogen content, and/or solution concentration;
4. packages that can be demonstrated to have less than 5 g of fissile material in any 10-L volume; and
5. packages with less than 1 kg of plutonium if together the ^{239}Pu and ^{241}Pu are less than 20% of the total plutonium mass.

The shipper is responsible for being able to demonstrate and to ensure that the package meets the fissile exemption specifications. This demonstration and ensurance is not always straightforward, particularly for waste material or material that could potentially alter its form during transport. The explanatory and advisory material planned for issuance with the 1996 Edition of IAEA SS6 will provide some guidance that will hopefully help clarify the intention and use of the exemptions.

10.2.2 Single-Package Requirements

The specific domestic requirements that must be met for a single package are described in 10 CFR 71.55. The domestic and international regulations require that a single, water-flooded package be adequately subcritical in the undamaged or damaged condition. The undamaged package is considered to be the physical condition of the package under the Normal Conditions of Transport (10 CFR 71.71); the damaged package refers to the physical condition of the package following its exposure to the tests for the Hypothetical Accident Conditions (10 CFR 71.73). All internal voided volumes of the package, including the containment system, must be assumed to be filled with water for the single-package evaluation. The term "water" should be understood to include all forms (i.e., full-density, snow, ice, steam, vapor, and sprays). The moderating water and fissile material contents used in the evaluation must be in the most reactive credible condition consistent with the undamaged and damaged package.

An exception to the requirement to evaluate a single package with water ingress is provided by the regulations [see 10 CFR 71.55(c)]. To exclude water from the analysis requires, minimally, a leak-tight double containment with preshipment testing to ensure leaktightness (see IAEA SS6 para. 565). However, from a regulatory standpoint, exceptions are seldom justified and, until all other options are exhausted, the package designer and safety analyst should not consider a request for excluding water from the internals of a single-package evaluation.

For the single-package evaluation, the new 10 CFR 71 regulations state that the portion of the package defined as the containment system be closely reflected by water unless, as indicated by IAEA SS6, the packaging in combination with water provide greater reflection (i.e., a more reactive condition). The NRC regulations state that reflection of the containment system applies only to the undamaged package. Thus if the package and containment system are not the same, then two analyses must be done for the package in its undamaged condition—one with a water-flooded and water-reflected containment system separate from the package and one with a water-flooded and water-reflected package. The result with the highest multiplication factor must be used in the license approval process. The analysis for the damaged package must consider the fissile

material configuration, water moderation, and water reflection that yields the maximum reactivity consistent with the damaged state of the package.

10.2.3 Requirements for Arrays of Fissile Material Packages

Fissile material packages must be designed such that damaged and undamaged packages of the same design could be stacked in any arrangement under optimum conditions for criticality and still remain subcritical. Evaluations with package arrays and application of safety factors lead to a maximum number of N packages that can be safely handled in a single shipment. The TI for criticality control is obtained by dividing the number 50 by N. Then by limiting the cumulative TI of all packages in a nonexclusive use shipment to 50, the packages per shipment will be limited to N. (Note that the TI value assigned to a package based on its external radiation level may be higher than if it were based on criticality considerations, thus permitting fewer packages to be carried in a single shipment. TI values based on external radiation levels are discussed in Chapter 9.)

10.2.3.1 Previous Requirements

The previous 10 CFR Part 71 that was effective prior to April 1, 1996, defined three fissile classes. These fissile classes are important to review here in order to provide a comparison with the new 10 CFR 71 requirements and because previously licensed fissile packages were categorized by these classes. Only Fissile Class II utilizes the concept of TI for criticality control. The definitions of the fissile classes previously used in 10 CFR 71 and the impact of the new rule change are described below.

- For a package to be categorized as Fissile Class I, an infinite number of undamaged packages in an array had to be adequately subcritical and 250 damaged packages had to be adequately subcritical. ("Adequately subcritical" means below the subcritical limit discussed in Sect. 10.3.2.) Evaluation conditions required optimum interspersed moderation for both array systems, and the finite array had to be reflected by water. A TI for criticality control was not assigned for Fissile Class I packages.

- A Fissile Class II package must be controlled by the carrier during transport. The control parameter is the allowable number of packages (N) that can be transported on a vehicle. Arrays of damaged packages and undamaged packages were considered. The undamaged array evaluations had no material (i.e., only air) between packages, but the damaged array evaluations assumed optimum interspersed hydrogenous (water) moderation. Close reflection by water around the exterior boundary of the arrays had to be assumed for all systems.

Separate evaluations had to be done for a Fissile Class II package to determine the number of packages, N_d, that would ensure the damaged package array is adequately subcritical and the number of packages, N_u, that would make the undamaged package array adequately subcritical. The previous 10 CFR 71 regulations specify that the allowable number of packages on a vehicle, N, must be such that $5N < N_u$ **and** $2N < N_d$. Thus $N = \min(N_u/5, N_d/2)$, and, as discussed above, the TI for criticality control is defined as TI = 50/N. The TI is always rounded up to the first decimal place, and the value could not exceed 10 for a single Fissile Class II package. The vehicle TI limit is 50, and packages with different TI values could be placed on a vehicle.

- Packages certified under the 10 CFR 71 regulations applicable prior to April 1, 1996, that do not meet the specific requirements of Fissile Class I or Fissile Class II had to be transported as a Fissile Class III **shipment**. The number of packages were limited such that twice the allowable number of undamaged packages (i.e., two shipments) were adequately subcritical, assuming no interspersed material (i.e., a void) between packages and close reflection by water on the array boundary. This allowable number of packages in a shipment had to be evaluated to ensure adequate subcriticality with optimum interspersed hydrogenous moderation and close reflection by water. Unless excepted by the Commission under 10 CFR 71 41, this latter evaluation was performed with damaged packages. All Fissile Class III shipments were exclusive use.

10.2.3.2 Current Requirements

The current IAEA regulations and the new 10 CFR 71 rule eliminate the fissile classes and incorporate a uniform use of the TI concept. A TI value must be calculated for all packages, using the basic procedure described above for Fissile Class II packages. However, to avoid confusion caused by the fact that packages with different TIs can be on a vehicle, the quantity N is no longer referred to as an "allowable number of packages," but merely a number derived for the purposes of determining the array limitations and the value of the TI. The normal vehicle limit is a cumulative TI of 50, with no individual package TI > 10. For exclusive-use vehicles, the cumulative TI is limited to 100 and individual packages with a TI greater than 10 can be shipped. The minimum value of N is 0.5, and the maximum value is infinity, thus leading to an allowable TI of zero.

10.3 PRINCIPLES OF CRITICALITY SAFETY

Figure 10.1 provides one summary interpretation of the elements that are needed for a complete criticality safety evaluation of a system. The transport regulations were formulated to define normal and accident conditions of transport that typically enable packaging evaluations to be limited to the branch labeled "Identification of potential criticality." Sections 10.4 and 10.5 provide guidance and discussion relative to nuclear analysis for transportation packages. This section presents information on the major parameters that affect the criticality safety of a transportation package, discusses the meaning of subcritical limit, and provides references that define or explain current practices in criticality safety.

10.3.1 Parameters Affecting Criticality Safety

Three fates are possible for a neutron in a fissile material package. The neutron may encounter a fissile nuclide and induce fission, producing additional neutrons to continue the fission chain; the neutron may be removed from the fission chain through absorption by a nuclide constituent of the package or contents or through absorption by a fissile nuclide without fissioning; or the neutron may escape the single-package system by leaking from the package and so be removed from the fission chain. Criticality is achieved when there is a balance between neutron production by fission and neutron loss by absorption in and leakage from the system. The criticality of a system is often discussed in terms of an effective multiplication factor, k_{eff}, which is defined as the ratio of the neutron production rate to the neutron loss rate in the system. For the system to remain subcritical, k_{eff} must be less than unity. The k_{eff} of the system can be maintained by

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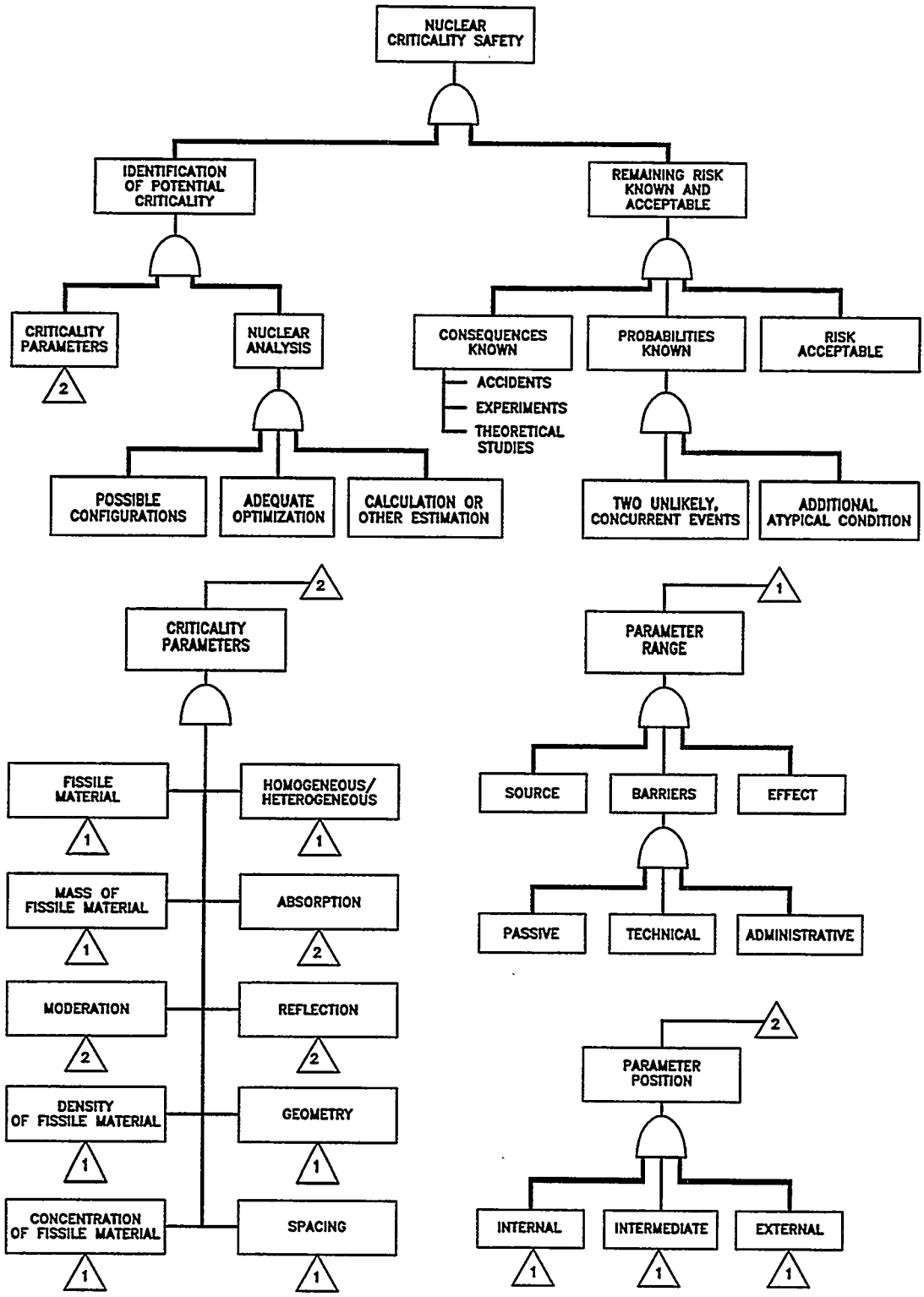

Fig. 10.1. Summary view of nuclear criticality safety.

ensuring adequate control of the parameters that affect the neutron balance. It is important not only to *identify* these parameters, but it is also important to *understand the interrelationship* of the parameters and their effects on k_{eff}.

The principal parameters of concern in controlling the criticality safety of transportation packages are

1. type, mass, and form of the fissile material;
2. moderator-to-fissile material ratio (degree of moderation);
3. amount and distribution of absorber materials;
4. package geometry—internal and external; and
5. reflector effectiveness.

Control of the first, second, and fourth parameters can serve to limit the fission rate either by limiting the amount of fissionable material or by controlling the energy and spatial distribution of the neutrons that cause fission. The second, third, and fourth factors may be adjusted to provide an adequate neutron loss rate through neutron capture, while the last four variables affect the neutron leakage. A major difference between the design of a fissile material package and a nuclear reactor is that a package design should *maximize* neutron loss by absorption and leakage; a reactor design seeks to *minimize* neutron loss. This emphasis on designing a package to maximum neutron loss is necessary for the transport of large quantities of fissile material. In order to both effectively maximize neutron loss and ensure an adequate margin of safety, the range of parameters that is possible in a package must be investigated. The effect of any possible change from the proposed design and/or assumed analysis condition must be known. Rather than designing to a set point (e.g., criticality for a reactor), the package must ensure that no combination of credible parameters (or parameter changes) will lead to criticality. Each of the principal parameters that influence the neutron balance to a package design are discussed below.

10.3.1.1 Fissile Material

The type (^{235}U, ^{239}Pu, etc.), mass, and form (homogeneous, heterogeneous, metal, oxide, etc.) of the fissile material that is carried in a package will affect the neutron production in the package. The effective fission cross section varies both with the type and the form of the fissile material. The mass of ^{235}U is typically defined in terms of the *enrichment*, or weight percent, of ^{235}U in the specified uranium content of the package. For plutonium, which often consists of several different isotopes, the weight percent of the individual isotopes is specified, together with the total mass of plutonium. In some cases, it may be possible to determine whether the package is safe by comparing the amount of fissile material present in a specific form against known safe quantities of fissile materials. Three ANSI Standards that contain such data are ANSI/ANS-8.1,[3] ANSI/ANS-8.7,[4] and ANSI/ANS-8.15.[2]

Moderators are materials that are made up of light nuclei (e.g., hydrogen or carbon) which slow neutrons down (i.e., moderate them) to the energy level where fissile nuclei have their largest cross section. Homogeneous uranium-moderator mixtures above about 7% enrichment are more reactive (produce higher k_{eff}) than the corresponding heterogeneous uranium-moderator mixture with the same mass and degree of moderation. However, heterogeneous systems of high-density uranium (e.g., metal, fuel pins, etc.) containing uranium enriched below about 7% are more reactive because the isolated moderator allows high-energy neutrons that escape from the fissile material to easily scatter to the low energy desired for ^{235}U fission prior to re-entry in the fissile

material. Thus neutron loss is avoided in the mid-energy range, where parasitic ^{238}U neutron capture dominates. As the enrichment of ^{235}U increases, the loss of neutrons due to large ^{238}U cross-section resonances becomes less important. This behavior of homogeneous and heterogeneous systems as a function of ^{235}U enrichment is an example of the knowledge of neutron interactions that needs to be understood in the design and evaluation of a package.

10.3.1.2 Moderator Material

Fast neutrons produced by fission lose energy in collisions with atomic nuclei. The decrease of neutron kinetic energy due to elastic scattering is called moderation. The lower the atomic mass number of the scattering material, the greater the neutron energy that can be lost in a single collision. This increase in potential energy loss with decrease in atomic mass is the reason that hydrogenous materials are the best moderators for efficiently obtaining a thermal neutron system. (A thermal neutron system for criticality purposes is one where the fissions arise predominantly from "thermal" neutrons. Thermal neutrons are low-energy neutrons in thermal equilibrium with their surroundings such that there is equal probability of an increase or decrease in energy after a scattering event.) Water is an excellent moderator because of its high hydrogen content. The common occurrence of water in nature makes it the major moderating material of concern in criticality safety. Carbon is less common but is a more effective moderator than hydrogen because it has a lower absorption cross section than hydrogen. To increase its effectiveness in thermalizing neutrons, a moderator must be in close proximity to the fissile material. Again, because of the high resonance absorption of neutrons in ^{238}U, uranium with enrichments less than about 5% cannot be made critical without the presence of a moderator to scatter high-energy neutrons "around" the intermediate energies of the ^{238}U resonances. Fissile systems moderated by hydrogen are often characterized by the ratio of hydrogen(H)-to-fissile(X) material, often termed the degree of moderation. For thermal systems, there is an H/X ratio that yields the maximum k_{eff} value. Determination of this optimum H/X ratio is an essential component of evaluating the criticality safety of individual packages and arrays of packages. Although water (in its various forms) is the primary moderator material considered in a package evaluation, the effect of other moderating material (e.g., hydrocarbons) in the packaging must also be considered.

10.3.1.3 Absorber Materials

Materials that provide efficient absorption of neutrons without producing new neutrons via fission are typically referred to as absorber materials or neutron poisons. Absorber materials are more effective for low-energy neutrons and so are most efficiently used in combination with a moderating material that will lower the neutron energy prior to encountering the absorber material. Placement of absorber materials between fissile materials can be an effective way to significantly reduce the system k_{eff} by reducing the neutron interaction between separated fissile material. Homogeneous distribution of the neutron poison is important because the absorption effectiveness is strongly dependent on the surface area of the absorber material. Heterogeneous poisoning thus always provides a weaker neutron-absorbing effect than that provided by the same quantity of a homogeneously distributed absorber. The package designer must consider the type, quantity, and placement of absorber material to ensure the presence of the poison, and its effectiveness, under all conceivable conditions. These conditions can include potential loss or redistribution of the absorber in an accident. Boron and cadmium are the primary nuclide absorbers incorporated in the design of packages used to transport significant quantities of fissile materials. Other package materials (principally the iron in structural components) provide some neutron absorption that can

often be considered in the criticality safety evaluation. In addition, isotopes of uranium and plutonium, principally ^{238}U and ^{240}Pu, can be major absorbers in a moderated system.

10.3.1.4 Package Geometry

The internal geometry of a package, together with the characteristics of the fissile material and moderator, provides the most significant effect on the neutron energy and spatial distribution. The geometry of heterogeneous fissile material (e.g., fuel assemblies), the design and placement of absorber materials, and the separation between fissile material are all important to the criticality evaluation of a single package. The surface-to-volume ratio of the package also affects the neutron leakage from the package and thus the neutron interaction between packages. Safe, single package geometries can be obtained by limiting the diameter of the fissile material container (e.g., the cavity diameter), but such packages are often impractical. Since small packages typically have high neutron leakage from their surfaces, the nuclear interaction between like packages (i.e., for the array configuration) becomes very important. Conversely, large packages containing significant amounts of shielding materials allow very little neutron leakage and therefore the neutron interactions related to the single package evaluation typically determine the system k_{eff}.

10.3.1.5 Reflector Effectiveness

A material that facilitates the return of neutrons leaking from the outer fissile material boundary is called a reflector. Large spent fuel packages have very effective reflectors in the form of the shielding material that are part of the packaging. Water is also an effective reflector and, in accordance with the regulations, must be used to evaluate both single packages and package arrays. In general, only about 30 cm of water is needed to ensure a maximum increase in k_{eff} due to the reflection of neutrons. When evaluating an array of packages, any water between individual packages must be such that the neutron interaction is maximized and, for a finite array, a water reflector must be placed at the outer boundary.

A criticality safety evaluation requires that the system of concern be studied to determine the combination of external conditions and parameter variations that yield the most reactive credible configuration. Normal conditions of transport and hypothetical accident conditions are prescribed by the regulations. However, consistent with the normal and accident conditions, realistic variations in parameters must be considered in order to identify the combination that provides for the maximum k_{eff} value. Even though sensitivity to a specific parameter can often be studied with relative ease, the most difficult task is to evaluate the effect of varying one parameter while, at the same time, another parameter is changing. The evaluation is intended to determine if an overall maximum k_{eff} occurs at some point other than the maximum for a single parameter variation. A more thorough discussion of this issue will be presented in Sect. 10.5.

10.3.2 Subcritical Limit

After determining the most reactive single package and array configurations, an assessment must be made to determine if the k_{eff} value provides adequate subcriticality. If, as in the usual case, the quantity of fissile material is greater than known safe values, calculations must be performed to determine the system k_{eff} value. The criteria for "adequate subcriticality" should include:[5]

1. A minimum subcritical margin that does not depend on the accuracy of the code and chosen nuclear data. Thus the margin would persist even if it were possible to calculate the multiplication factor exactly. The value of the margin is a matter for judgment, bearing in mind the sensitivity of the multiplication factor to foreseeable physical changes to the package.

2. An allowance for systematic and random errors associated with the performance of the code and chosen nuclear data library.

Thus if the maximum neutron multiplication value for a subcritical system is k_s, then the system would be adequately subcritical if

$$k_s + 2\sigma < k_c - \Delta k_c - \Delta k_m \ ,$$

where

σ is the standard deviation resulting from a statistical (Monte Carlo) analysis of the system,
k_c is the mean k_{eff} resulting from the calculations of the benchmark experiments,
Δk_c is an allowance for uncertainties in the experiments and calculational technique (if not accounted for in k_c), and
Δk_m is a required additional margin to ensure subcriticality.

The systematic and random errors embodied in k_c and Δk_c should involve allowances for (1) the bias associated with the experimental data used in the validation, (2) the bias associated with any interpolation and/or extrapolation of trends in the calculated experimental data to the conditions of the actual package configuration, (3) any bias associated with the results of the computational method as compared with the experimental data, and (4) the uncertainty in the computational method. Statistical approaches have been used to enable a quantitative measure of the bias and uncertainties to be estimated based on the performance of computational methods against critical experiments. An example of one approach is provided in ref. 6 (see also Appendix C). The upper limit for adequate subcriticality often varies as a function of system-dependent parameters such as the average neutron energy causing fission or the ratio of thermal-neutron absorption to total absorption. Other problem-specific parameters such as hydrogen-to-fissile atom ratio, fissile material enrichment, or fissile material density may be appropriate for particular package designs or payloads. Nonstatistical approaches that utilize the worst-case comparison with applicable critical experiments to estimate $k_c - \Delta k_c$ have been used in safety evaluations, particularly when a very limited number of applicable critical experiments are available. Whether a statistical approach is used or a worst-case comparison, if an insufficient number of critical experiments are available in the range of applicability, then an additional margin of uncertainty needs to be added by the analyst and/or Δk_m needs to be increased.

The value for Δk_m is basically a matter of judgment. A value of 0.05 has typically been used by designers and safety evaluators as a reasonable value that is adequate for a majority of transport package designs. However, criticality safety must depend, not on blind acceptance of an arbitrary value, but instead on an evaluation of realistic circumstances that could eliminate the "arbitrary" margin of safety. Package designs that require only a slight change in the design configuration or model assumptions to eliminate the 0.05 margin are unacceptable. In these cases, or when a paucity of applicable experimental data are available for validation (see Sect. 10.4.4), the safety margin needs to be increased. It is also possible to have packages that are so close to a safe-geometry or safe-mass condition that unrealistic measures would be needed to ever eliminate a

safety margin of even 0.03. However, a package designer must develop a justification for the margin of subcriticality, Δk_m, being used.

10.3.3 Current Standards and Practice

In order to understand and to be capable of evaluating the criticality safety aspects of a transport package, it is helpful to have a working knowledge of the ANSI Standards and established guides that pertain to criticality safety. The Standards that are most applicable to the criticality safety of transportation packages are those developed as a part of the ANS-8 series of Standards sponsored by the American Nuclear Society.

The Standard that outlines the fundamental principles of criticality safety practice is ref. 3: ANSI/ANS-8.1, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors."[3] This Standard discusses administrative practices — responsibilities, procedures, controls, and reviews — and technical practices — control parameters, the double-contingency principle, and validation techniques for computational methods. Validation of the computational method to be used in calculating the system k_{eff} is a very important component of package design because computational methods (as opposed to experimental methods or testing) are the dominant means of demonstrating the criticality safety of transportation packages. Much of the general guidance in Sect. 10.3.2 was derived from ANSI/ANS-8.1 and ANSI/ANS-8.17, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors" (ref. 7). This latter Standard addresses in a more specific manner the procedure to establish a calculated allowable maximum k_{eff} value for the system being evaluated.

As indicated above, the usual practice in assessing the criticality safety of a transport package is to use a computational method that calculates the system k_{eff}. "Reality" checks on the computational results and "rules of thumb" to determine initial design parameters can often be obtained from the Standards and the *Nuclear Safety Guide* published originally as TID-7016 and then subsequently revised in 1978 and 1996.[8] The guide provides a substantial amount of background information and a wealth of data on safe quantities and configurations of fissile material contained in a variety of compounds. Data on fissile system critical dimensions and subcritical limits can also be found in ref. 9 and ANSI/ANS-8.1. For many other fissionable actinides (e.g., ^{237}Np, ^{238}Pu, ^{240}Am, etc.) limits can be found in ANSI/ANS-8.15. Another Standard, ANSI/ANS-8.7, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials,"[4] can also be beneficial for estimating array limits that might also be used in transportation applications. Hand calculation methods for determining safe limits on package arrays in storage and/or transportation are addressed in the *Nuclear Safety Guide* as well as ref. 10.

10.4 COMPUTATIONAL METHODS

Over the last 25 years there has been a continual increase in the use of computer codes to calculate the k_{eff} value of transportation packages containing fissile material. Wide use of sophisticated codes at a reasonable cost is now possible by modern, high-performance personal computers and workstations. This section discusses three aspects of a computational method that need to be considered prior to selecting one for a safety analysis: technical capabilities, availability and usability, and acceptance. Advice on validation issues and other areas of concern that arise from the use of computational methods in a safety analysis will also be presented.

10.4.1 Technical Capabilities

Calculation of an accurate multiplication factor (k_{eff}) for a fissile system requires that the analysis method provide sufficient rigor in the physics of particle kinematics and cross-section data preparation while allowing detailed modeling of the system geometry features. The adequacy of current criticality safety analysis methods relative to these technical areas will be discussed below.

10.4.1.1 Particle Kinematics and Cross-Section Data

The particle kinematics of concern in criticality safety calculations involve neutron-nucleus interactions and subsequent secondary neutron energy-angle distributions. These are neutral particle transport processes, and the appropriate physical laws are well understood. A full description of the neutrons in a system can be obtained (in principle) by the solution of the Boltzmann particle transport equation. The solution of this equation can be obtained using statistical (e.g., Monte Carlo) or deterministic (e.g., discrete-ordinates) methods. The kinematics in methods that solve approximations to the Boltzmann equation, such as the deterministic diffusion theory model, are not sufficiently rigorous for most criticality safety applications. In any solution scheme it is necessary to have an accurate representation of the energy-dependent cross-section structure for each nuclide in a problem. Accurate kinematics modeling requires a combination of good data for the neutron scattering, absorption, and production processes as well as sufficient consideration for the actual particle interaction, including anisotropic scattering. The accepted U.S. database for cross sections used in multigroup or energy-pointwise libraries is the Evaluated Nuclear Data Files currently maintained at Brookhaven National Laboratory. The latest version of these data files is Version 6 (ENDF/B-VI).

All deterministic codes and many Monte Carlo codes solve the Boltzmann transport equation using an energy-groupwise representation. To solve the equation using this multi-energy-group representation requires group-averaged cross sections for each spatial zone. The cross sections are prepared with the criteria that the average reaction rates over each energy group are preserved. Preserving the reaction rates necessitates that a neutron flux spectrum be assumed for the preparation of the group data. Use of a very-fine-group structure minimizes the impact of the assumed flux spectrum because it is the relative change in the flux across the energy group that affects the "weighting" of the cross-section data. Typically the spectrum used to prepare a fine-group library (100 groups or more) is a combination of a fission spectrum joined to a 1/E slowing-down spectrum that, in turn, is tied to a thermal Maxwellian spectrum (see Fig. 10.2). Fine-group libraries can subsequently be collapsed to a "broad" group structure using a flux spectrum that is consistent with a general class of problems. For example, a recently developed 44-group library (see ref. 16 of Appendix E) was collapsed from a 238-group library using a flux spectrum calculated for an infinite lattice of light-water-reactor (LWR) fuel pins. The 44-group library has been demonstrated as valid for a large class of problems where the fission is dominated by thermal energy neutrons; however, its applicability is more limited than the parent 238-group library. Multigroup cross-section libraries with various group structures are available for use, depending on the neutron energy range of interest (e.g., well-moderated thermal systems or dry, fast systems).

The multi-energy-group (or simply multigroup) cross-section sets, such as those discussed above, contain a large amount of information relative to the particle kinematics (e.g., scattering matrices). However, correction of the cross sections for resonance self-shielding (reduced effect of the cross section at an energy point due to the effect of resonances at a higher energy) and Doppler effects

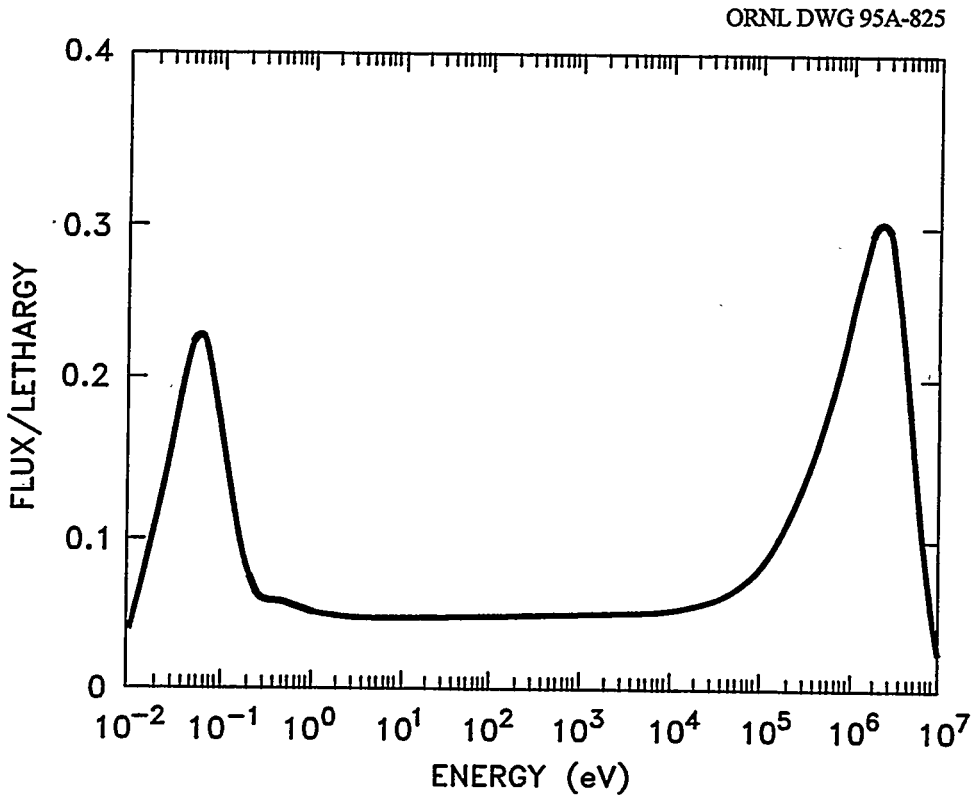

Fig. 10.2. Typical profile of a fission-1/E-thermal flux vs energy, where lethargy is an alternative energy variable defined as $U = \ln(E_0/E)$, with $E_0 = 10$ MeV for this plot.

(broadening of the cross-section resonances caused by the thermal motion of the nucleus) is most accurately performed on a problem-specific basis. Resonance processing depends on the amount of material (fissile and non-fissile) in the system, the relative location of the different materials (moderator, reflector, absorbers, etc.), and the condition of the material (temperature, density, etc.). Resonance effects are often extremely important in the calculation of an accurate k_{eff} value; the typical exception is a system where high-energy neutrons above the resonance range dominate. Stand-alone libraries exist that require the user to select the degree of resonance shielding from a set of preprocessed data. This procedure is limited by the user's ability to define the appropriate level of self-shielding for a specific application. The most well-known library of this type is the Hansen-Roach 16-group library that is based primarily on pre-1960s data. Specific processing based on a user description of the system can be provided by data processing code systems such as AMPX[11] and NJOY.[12] Automation of this process as part of a complete problem analysis is available in the SCALE code system (see Appendix E).

The alternative to using multigroup cross-section data is the use of energy-pointwise data in what are termed "continuous-energy" codes. In these codes the kinematics of the particle interactions are modeled explicitly in the code and the data libraries are preprocessed only to account for temperature effects. Relative to multigroup data libraries, energy-pointwise libraries are very large due to the number of data points required to represent the contour of the cross-section data for each nuclide over the entire energy range. Inaccuracies in analyses with energy-pointwise libraries can result from too few data points or poor processing of the temperature effects.

Computer codes that use the deterministic discrete-ordinates technique to solve the Boltzmann equation are very popular for shielding calculations and small-core (high-neutron leakage) reactor physics calculations. The discrete-ordinates technique solves for the neutron population on a spatial mesh as a function of discrete-energy groups and angular directions. The number of spatial mesh points, energy groups, and discrete angles used to solve for k_{eff} has a direct impact on the computing time and the results. Limiting the groups, quadrature, or mesh to reduce computational time can result in an inaccurate answer for k_{eff}. Angular quadratures, typically referred to as S_n, are based on providing an adequate numerical integration of the angular flux. S_8 and S_{12} quadratures are typical for criticality safety applications. Deterministic codes that solve the integral transport form of the Boltzmann equation can also provide accurate values of k_{eff} for one-dimensional (1-D) geometries or infinite lattices of uniform cells (spheres, cylinders, etc.).

Monte Carlo solution methods involve the use of statistical sampling techniques to "track" neutron histories through scattering and absorption events and then subsequently calculate a k_{eff} value based on the cumulative information from all the histories. Intermediate k_{eff} values are calculated for a batch of neutrons. Early batches of neutrons are typically discarded because they are assumed to be overly biased by the input neutron starting distribution. An estimate of the standard deviation of the k_{eff} value is provided by the code although the k_{eff} value and deviation estimate can be meaningless if the analysis output does not indicate adequate convergence on the predicted k_{eff} value (see Sect. 10.4.5). With the Monte Carlo approach, the neutron kinematics can be treated in either an energy-pointwise or energy-multigroup representation. The burden of ensuring proper particle kinematics shifts in large part from the data processing to the code as one moves from multigroup to continuous-energy analyses.

10.4.1.2 Geometry Modeling

As noted in Sect. 10.3.1, the details of the system geometry and relative material locations therein are very important to predicting an accurate value of k_{eff}. Historically, readily available deterministic discrete-ordinates codes have been limited to one or two dimensions with the spatial mesh restricted to orthogonal geometries. Codes such as ANISN,[13] XSDRNPM,[14] and ONEDANT[15] are limited to 1-D geometries; DORT[16] and TWODANT[17] are limited to two-dimensional (2-D) geometries. The integral transport code CASMO[18] can provide rapid and accurate analysis of a lattice of 1-D cells. Large, 2-D, discrete-ordinates models can require extensive computing resources. Until the advent of high-performance computers with large amounts of memory and storage, three-dimensional (3-D) codes such as THREEDANT[17] and TORT[19] were impractical. Even with advanced computing capabilities, the 2- and 3-D codes are still more often than not very impractical for a packaging analysis because of the amount of specific geometry detail that must be modeled to ensure an accurate representation of the neutron interactions and the subsequent effect on the system k_{eff} value. Codes like TWODANT/GQ[17] allow nonorthogonal meshing and thus extend the range of problems that can be modeled accurately, but only slightly relative to the range of real-world packaging problems.

Thus the role of deterministic codes for criticality safety analyses has been significantly restricted due to geometry model limitations. For package analyses, the role of deterministic codes has been limited to the calculation of bounding values (based on conservative model simplifications) and investigation of the sensitivity of k_{eff} to changes in system parameters. These sensitivity analyses often use a model of a specific region of the full problem (e.g., a fuel pin or homogenized fissile material unit surrounded by a detailed basket model) to ensure or demonstrate that the full package model has utilized conservative assumptions relative to the calculation of the system k_{eff}

value. One distinct advantage of deterministic solution schemes is that accurate details of energy and spatial quantities such as fluxes, reaction rates, etc., are available, together with the calculation of the multiplication factor. This detailed neutronics information can often help the analyst better understand the physics of the problem being evaluated.

The vast majority of criticality safety analyses for transport packages are performed using Monte Carlo codes because of the detailed geometry modeling capabilities they provide. Monte Carlo codes track individual neutrons from collision site to collision site as they move through the geometry model. Based on the number of fissions, parasitic absorptions, and leakage from the system the code can determine the system k_{eff} value. Monte Carlo techniques are most efficient for obtaining average quantities over space and energy; thus local (space or energy) information on fluxes, reaction rates, etc., are more difficult to estimate accurately. The major concern in particle-tracking algorithms is the determination of the boundary location along the track that separates two regions of material. Theoretically, Monte Carlo codes are limited geometrically only in their ability to specify the boundaries between regions.

The KENO series of codes have been the predominant codes of choice for criticality safety analyses of transportation packages over the past 25 years. All publicly released versions of KENO to date use multigroup cross-section data. The initial version of KENO was developed in the late 1960s, and since that time the KENO versions have been used extensively for evaluating the criticality safety of transportation packages. Most KENO versions have utilized a geometry package that allows the user to create complex models using predefined geometric shapes (spheres, cylinders, boxes, etc.). Intersections and rotated bodies have not been allowed. An exception to these restrictions was given in the generalized geometry package provided as an option with the KENO-IV version release.[20] The KENO-V.a code,[21] made available in the early 1980s as a module in the SCALE code system (see Appendix E), provided a significant enhancement for transportation packaging analyses in its "array-of-array" and "hole" features. The array-of-array feature enables any unit in an array to also be defined as an array. The holes feature enables a geometric unit to be placed at any specified location within a larger unit. Computer-drawn 2-D plots of a KENO-V.a model of a spent fuel transport package are shown in Figs. 10.3(a) and 10.3(b). The particular plotting scales used here do not allow the detailed modeling of each fuel pin to be seen, but each pin in each assembly is modeled in explicit detail (fuel, gap, clad). A change of scale could enable the detail to be seen. The new KENO-VI[22] code released with the SCALE system (see Appendix E) provides a geometry package that is more extensive and easier to use than the generalized geometry of KENO-IV. The KENO-VI code will allow rotated arrays, intersections of unit boundaries, and definition of arbitrary bodies surfaces. However, it is anticipated that the more computationally efficient KENO-V.a geometry package will continue to provide the modeling capabilities needed for the vast majority of transportation packages.

The other principal Monte Carlo code for criticality safety analysis is the general-purpose radiation transport code MCNP.[23] Since the mid-1980s the developers of the continuous-energy MCNP code have made an intensified effort to provide the geometry features and pointwise cross-section data needed to perform transportation packaging analyses. The principal geometry addition in this time frame was the capability to model repeated structures, thus providing the means to more easily model multiple arrays such as that shown in Fig. 10.3(a). Another well-established Monte Carlo code used in criticality safety is the MONK[24] continuous-energy code developed in the United Kingdom. Like MCNP and KENO, this code has the geometric capabilities needed for an accurate analysis of a transportation package but has not been widely distributed or used in the United States.

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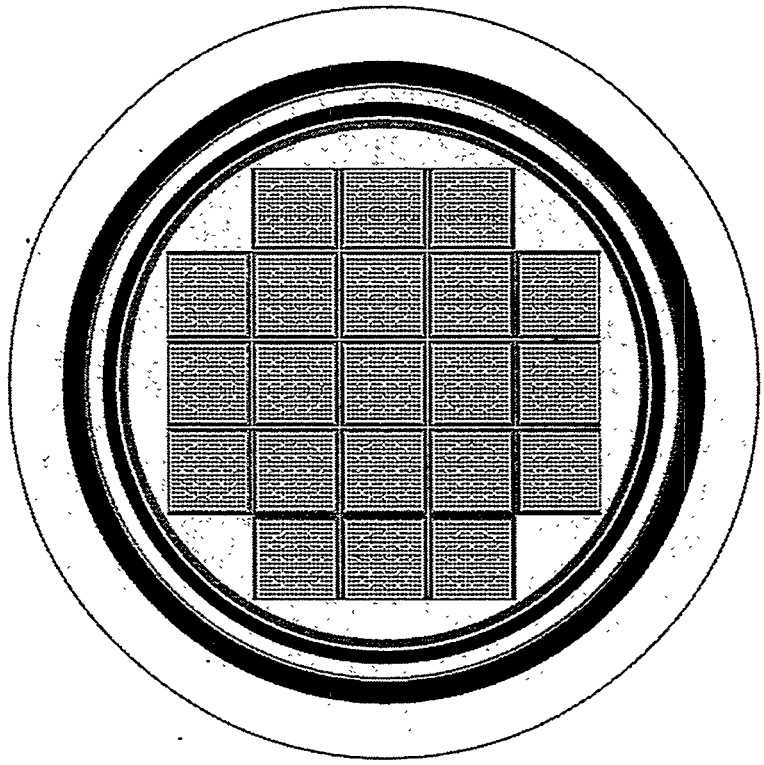

Fig. 10.3(a). Radial slice of a KENO-V.a model for a typical spent fuel transportation package.

10.4.2 Availability and Usability

Although a package designer must first and foremost ascertain that a code used in a safety analysis has the needed technical capabilities, the designer should, to a lesser extent, consider the availability and usability of the code and data. Availability as used here encompasses areas such as ease in obtaining the code, nuclear data, and documentation needed for use, and the access to expert technical assistance. Usability as defined here encompasses the degree of familiarity that staff have with the software, the ease in training existing and/or new staff to use the code, the flexibility of the software for a wide range of packaging problems, and the portability of the software to existing and/or potential future computing platforms.

The KENO V.a multigroup code with cross-section processing as provided within the SCALE code system, and the MCNP and MONK codes with their available point libraries, are the computational tools that are the most well known and used by criticality safety specialists throughout the world. For historical reasons there is a wider base of user experience (criticality safety applications) with the SCALE/KENO package, but the number of MCNP users continues to increase. Both of these code packages are well documented and easily obtained from the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory. The MONK code is also well documented, but its base of user experience is small outside the United Kingdom, and its availability is limited to commercial distribution. Other Monte Carlo codes can be applied to the criticality safety analysis of transportation packages; however, they

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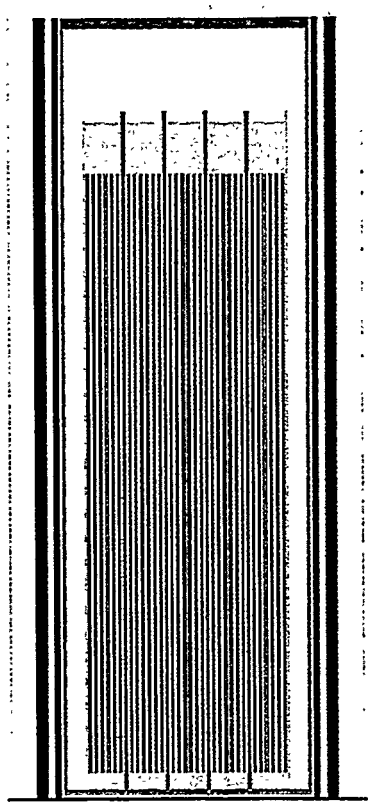

Fig. 10.3(b). Axial slice of a KENO-V.a model for a typical spent fuel transportation package.

are either very limited in use or their availability is such that they would probably not be good choices for designers seeking a new code to use.

Numerous deterministic codes have been used on a limited basis for parameter studies or bounding analyses of package designs. ANISN, XSDRNPM, ONEDANT, TWODANT, and DORT are all publicly available from RSICC and have been used in various aspects of packaging design. Within their geometry limitations the codes are all recognized as acceptable computational tools. Note, however, that only XSDRNPM, ONEDANT, and TWODANT are being actively supported by the developers for criticality safety applications. The greatest concern is ensuring that properly processed multigroup cross-section libraries are used with the codes. The Hansen-Roach 16-group library is a well-known, readily accessible library that is often used with these codes, but the library is based on pre-ENDF data and is highly dependent on the selection of an appropriate self-shielding factor for the problem of concern. As noted in Sect. 10.4.1.1, the SCALE code system provides an automated resonance processing procedure based on a user description of the problem and is readily interfaced with XSDRNPM or DORT; however, there is no readily available interface with ONEDANT and TWODANT. The CASMO code has a more limited area of applicability and is only available via commercial license from the developer.

All Monte Carlo codes mentioned above are available on, or can be easily ported to, a wide variety of computing platforms. In each case the developers are making concerted efforts to keep up with the ever-changing facets of software compilers and operating systems and to provide graphical user interfaces and model visualization packages to enhance the ease of use and improve capabilities for model development and identification of user errors. The applicability of these codes to the vast majority of packaging-type problems is well established. Thus in the area of usability, the deciding factor often lies with the degree of familiarity that users have with an analysis tool and/or the ease with which training or expert assistance can be provided. The designer is responsible for being able to demonstrate that the analyst has the level of understanding necessary to correctly use the code and explain any assumptions required in the analysis. In addition, the ability to understand and to use multiple codes and data is a desirable feature for an individual or organization because it allows independent confirmation that a difficult problem has been analyzed correctly.

10.4.3 Acceptance

For the purposes of this section, the acceptance of a code refers to the extent to which the code and data are recognized as applicable to the problem at hand and/or are understood by the criticality safety and packaging design community, including the regulatory reviewers. Each of the Monte Carlo codes cited above is well established in the area of criticality safety and has been used numerous times in the safety analysis of transportation packages. However, if a code used in the safety analysis is unfamiliar to the reviewer, then additional questions may be generated during the certification process unless adequate explanation is provided with the analyses. For example, most technical reviewers in the United States currently have less familiarity with the MONK code than with the KENO or MCNP codes.

The infrequent use of deterministic codes in criticality analyses, and packaging analyses in particular, means that more explanation regarding the adequacy of the code of choice may need to be provided with the safety analysis. The limitations on model geometry and the scarcity of adequate cross-section data for all the codes add to the reasons that the deterministic codes are not as readily acceptable for general packaging analysis as the Monte Carlo codes. As noted above, however, deterministic codes must be used where analyses to investigate reasons for small changes in k_{eff} are desired.

10.4.4 Validation Issues

The calculational method used to establish the criticality safety of a transport package should be validated against measured data that can be shown to be applicable to the package design under consideration. The validation process should provide a basis for the reliability of the calculational method and should enable designers and regulatory staff to have assurance that the k_{eff} for the package will not exceed the subcritical limit (see Sect. 10.3.2). Thus as designers seek to optimize packages for higher fissile loadings and/or increase their flexibility in defining the package contents, the k_{eff} value for package safety analyses are often closer to the subcritical limit and cause the validation process (and the basis of the subcritical limit value) to come under increasing scrutiny. Unfortunately, the availability of experimental data and the lack of guidance on determining the applicable range for a validation study can be major problems for designers seeking to design a package very near the subcritical limit. Although this section will not provide

definitive answers to all these problems, this section will try to identify issues that the analyst must carefully consider in performing a validation study for a safety analysis report.

The basic criteria for validation of a calculational method is provided in ANSI/ANS-8.1 (ref. 3). This Standard is endorsed by the NRC Regulatory Guide 3.4 (ref. 25). The Standard calls for the

1. establishment of bias and uncertainties through comparison with critical experiments that are applicable to the package design;
2. establishment of a range of applicability for the validation based on the range of parameter variation in the experiments;
3. extension of the range of applicability beyond the experimental parameter field based on trends in the bias as a function of the parameters and use of independent calculational methods;
4. establishment of a subcritical limit that considers the bias and uncertainties;
5. use of nuclear data, such as cross sections, that are consistent with experimental measurements of these properties; and
6. thorough documentation of the validation calculations with justification of the subcritical limit.

This guidance from the Standard has been interpreted and applied by many organizations responsible for criticality safety in the handling, transport, storage, or processing of fissile material (see, for example, refs. 26 and 27 for a good discussion of the validation process). In addition, validation studies have been performed on specific codes and data to demonstrate the general adequacy of the code and data (see, for example, refs. 26 and 27 and Appendix E, ref. 21). Although these general validation studies are excellent reference material for the package designer, the criticality safety analyst must still demonstrate that the specific (e.g., code version, cross-section library, and computer platform) computational method applied in the packaging design is validated in accordance with the above criteria. The following subsections provide a discussion of these elements of the validation process with emphasis on packaging design applications.

10.4.4.1 Establishment of Bias and Uncertainty

The first phase in the validation process is to select properly characterized critical experiments that have the parameters (e.g., materials, geometry, etc.) that are characteristic of the package design. The selection of critical experiments can be a time-consuming process. Several compendiums of experiments have been reported in the literature (see, for example, refs. 28 through 31). Reference 29 discusses experiments that typically were performed to address the materials and geometry characteristics found in packages designed to transport LWR fuel. References 28 and 30 provide data on all types of critical experiments. The work of ref. 30 is an ongoing international effort that seeks to provide evaluated critical experiment data for analysts to use in validation studies. Currently this database is rather limited, but work continues to add to the list of experiments. Other valuable sources of information on critical experiments are validation study reports (e.g., ref. 32 and Appendix E, ref. 21). Often these validation reports provide the necessary data on the critical experiments that are modeled.

With a suitable database of experiments in hand, analysts must still select a range of critical experiments that cover the compositions and arrangements of fissile, moderator, absorber, and structural materials that are present under normal and accident conditions of transport **for their specific package design.** As will be discussed later, it is not easy (nor even possible) many times to find sufficient critical experiments with all the characteristics used in the packaging design.

However, analysts must make every effort to demonstrate the adequacy of the computational method in important areas that often dominate the reactivity of a package: enrichment of ^{235}U, hydrogenous moderation consistent with optimum conditions in and between packages, leakage from and/or reflection of the package, and effectiveness of neutron poisons needed for criticality control. The analyst must remember that the combination of different characteristic parameters found in the package design are very important to criticality safety, and experiments with these similar combinations of parameters are of prime importance in the validation process.

Unfortunately, nearly all critical experiments available for the validation of transport packages tend to emulate only the contents of a single, undamaged package under water-flooding conditions. A package evaluation will require calculations for a single package, as well as arrays of packages for both undamaged and damaged conditions under optimum moderation. Other problems are associated with code validation: (1) no critical experiments are available that emulate large arrays of packages, and (2) the particular combination of important package parameters may seldom appear in one set of experiments. Because of these concerns, it may be necessary to model a wide variety of benchmark experiments to adequately assess the validity of the calculational technique used in the evaluation over the full parameter space of interest.

Section 10.3.2 provides a discussion of what must be considered in determining the bias and uncertainty of a calculational method. Uncertainties in the validation calculations come from three general sources. The first source is from the critical experiment and the experimenter, which may include uncertainties in the material and fabrication tolerance of the experimental hardware and fuel (compositions, assays, masses, densities, dimensions, etc.), the experimenter's manipulation and/or adjustments to the reported data, an inadequate description of the experimental layout and surroundings, etc. Reference 30 reports known experimental uncertainties that can be studied to determine their impact on the calculated k_{eff}.

The second source of uncertainty is from the computational technique itself, which may include uncertainties in the mathematical equations solved, calculational approximations utilized in solving the mathematical equations, the convergence criteria, the cross-section data and the manipulation of cross-section data, limitations of the computer hardware, etc. The third source of uncertainty is from the analyst and the calculational models developed to emulate the experiment, which may include uncertainties because of material and dimensional modeling approximations, the selection of various code options, individual modeling/coding techniques, interpretation of the calculated results, etc.

In practice, the code validator can estimate the total uncertainty through the application of any valid statistical treatment of the data. The total uncertainty determined usually appears as the bias and a variability in the bias, depending upon the statistical analysis applied. The combination of the bias and uncertainty in the bias is deduced from the mean k_{eff} to establish a maximum k_{eff} value. This maximum value, and any larger k_{eff} values, is taken to be critical within the confidence limits applied to the statistical technique to determine the uncertainty. Appendix C provides one statistical approach that enables the bias and uncertainty to be evaluated as a function of changes in a particular characteristic parameter. Note that in a statistical manner, the method of Appendix C provides an estimate of the uncertainty that arises from the computational method, the experiments, and the specific analyst.

10.4.4.2 Establishment of Range of Applicability

An integral part of a code validation effort is to define the range of applicability for which the validation is applicable. The range of applicability is intended to describe generically the type of system by identifying the important parameters and/or characteristics for which the code was (or was not) validated. The ANSI/ANS-8.1 Standard does not provide guidance on how to establish that a validation study using select experiments is applicable to the package design and arrangements of concern. Traditionally, this decision has been left to the experience and judgment of the criticality safety analyst. The method of Appendix C indicates the range of applicability as the limits (upper and lower) of the parameter or characteristic used to correlate the bias and uncertainties. For example, the range of applicability may be defined in terms of the moderating ratio (e.g., H/X = 10 to 500), or in terms of the average energy group causing fission (e.g., AEG = 6.5 to 21.5), or in terms of the ratio of total fissions to thermal fissions (e.g., F/F_{th} = 1.0 to 5.0), etc. For subsequent use of a validation study, the user should justify that the parameters and characteristics of the problem being calculated fall within the range of applicability defined during the validation study. Reference 33 presents an alternative scheme for defining the range of applicability by seeking similarity in important physical and neutronic categories.

The open question of course is what parameter or parameters should be used in defining the applicability of a system to a set of experiments. And can the parameters be considered individually or must they be considered in combination. The system of interest may have certain parameters that lie within the range of the experiments, but other important parameters may lie outside the range established by the experiments.

A draft criteria for establishing the range of applicability was conceived by an ad hoc group of criticality safety specialists from various DOE installations and published as Appendix E of ref. 34. According to this ad hoc group, three conditions must be satisfied to ensure that safety analyses for a real system lie within the range of applicability for the validation performed: similarity in materials, geometry, and neutron energy spectrum. The subcategories that can be used to define similarity are listed in Table 10.1. Appendix E of ref. 34 even proposed conservative tolerance bands (e.g., variation in enrichment or material specification) that specified the degree of similarity desired between the experiment database and the system of concern. The safety analyst with a knowledge of available critical experiments will quickly find that matching all the conditions of Table 10.1 can be a difficult, if not impossible, task for some applications. In fact, members of the ad hoc group that tried to "retrofit" the criteria of ref. 34 to existing situations quickly realized that the criteria and conservative tolerance bands that had been specified were far too restrictive for the existing experimental database. Even though work continues to provide guidance on establishing the range of applicability, the conditions, such as those listed in Table 10.1, provide safety analysts with a list of parameters that should be considered when searching for experiments to use in a validation study.

10.4.4.3 Extension of Range of Applicability

The ANSI/ANS-8.1 Standard indicates that extension beyond the range of applicability is justified based on trends in the bias as a function of system parameters and, if the extension is large, confirmed by independent calculational methods. However, as scientists continue to discover, extrapolation can lead to a poor prediction of actual behavior. Even interpolation over large ranges with no experimental data can be misleading. An example of this latter situation has been demonstrated in ref. 35, where a selected cross-section library performs well for high-energy and

Table 10.1. Factors affecting the range of applicability of criticality experiments

Factors	Subcategories used to establish similarities
Materials	Material types
	Fissionable
	Absorber
	Moderator
	Scatterer
	Criteria (applicable to all four material types)
	Element
	Isotopic composition
	Physical form (metal, solution, compound)
	Ratio to fissionable material
Geometry	Homogeneous and heterogeneous
	Shape
	Reflection
	Layering-ordering
	Relative material thickness
	Array criteria
	Mixed or same type units
	Number of units
	Shape of units
	Lattic pattern and spacing
	Interstitial material
	Reflection
	Coupling
	Layer-ordering
Neutron energy spectrum	Neutron density vs energy
	Leakage, absorption, production
	Flux

low-energy (thermal) systems, but has a substantial bias at intermediate energies, where ^{235}U resonance data become increasingly important. A comparison with other computational methods can illuminate the deficiency with a data library or code; however, given discrepant results from independent methods, it is not always a simple matter to determine which result is "correct" in the absence of experimental data (see ref. 36 for an illustration).

The Standard does not provide any guidance on what constitutes a "large" extension, nor does it provide any guidance on how to extend trends in the bias. In fact it is not just the trend in the bias that needs to be considered but the trend in the uncertainties and bias. The paucity of experimental data near one end of a parameter range may cause the uncertainty to be larger in that region. (Note: Any extension of the uncertainty using the method of Appendix C should consider the functional behavior of the uncertainty as a function of the parameter, not just the maximum value of the uncertainty.) To date, extension beyond the range of applicability has depended largely on the experience and judgment of the criticality safety specialist. To aid this judgment it is obviously beneficial to determine **and understand** the trends in the bias and uncertainty. One problem, however, is that there may not be sufficient data to determine a trend, much less understand the physics that contributes to the trend. A good example of a study where judgment and trend evaluations were used to extend the range of applicability is provided in ref. 37.

Confirmation of results with independent methods (code and data) are also very beneficial when a system analysis is beyond the range of applicability. The Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA) has sponsored working groups that have developed standard problem exercises related to transportation package systems for intercomparison of criticality safety analysis methods in regions where there is little or no experimental data.[38,39] The problems of ref. 39 may be particularly beneficial to help demonstrate the validity of a methodology for the analysis of large arrays (for which there are very little experimental data). As noted above, the intercomparison of two independent methods can only aid the analyst in determining how to extend the bias uncertainty; it is not a replacement for comparison with a critical experiment with the needed parameter values.

10.4.4.4 Establishment of a Subcritical Limit

Section 10.3.2 provides a discussion of the subcritical limit and the factors that should be considered in establishing the subcritical value.

10.4.4.5 Use of Nuclear Data

The computational methodology consists of both the computer code and the nuclear data (most prominently, the neutron cross sections). The validation process should be performed with the same cross-section library and processing scheme as used in the safety analysis for the transport package. Furthermore, the ANSI/ANS-8.1 Standard specifies that the nuclear data should be consistent with experimental measurements of these properties. This statement has been interpreted to have different meanings. However, the important issue is that the neutron cross sections used in the analysis should ideally be derived from actual measurements of the neutron capture, fission, and scatter cross sections. This is only an ideal that holds to varying degrees for the most common nuclides (uranium, plutonium, hydrogen, oxygen, iron, etc.) found in typical applications. Much of the nuclide data found in the Evaluated Nuclear Data Files (ENDF) are derived from known principles of nuclear physics or with guidance from measurements made on nuclides with similar

nuclear characteristics. Data included in ENDF are carefully evaluated and tested by experimentalists and specialists in nuclear data. The ENDF files do have deficiencies for certain nuclides, although often only in certain energy ranges. These deficiencies, however, point up the need for critical experiments that have similar material, geometry, and neutronic characteristics of the system being analyzed. However, in all the cases safety analysts should be cognizant of the source of the nuclear data used in the validation process and investigate the pedigree of nuclear data for materials that are not commonly used in criticality safety analyses.

10.4.4.6 Documentation of the Validation Process

The validation process is an excellent demonstration of the capability of the analyst to properly use the computational method on experimental configurations similar to the package design. Documentation of the validation process should provide the results that establish the bias and uncertainty, discuss the establishment of the range of applicability and any needed extension of the range, and clearly specify the basis and assumptions used for the subcritical limit. However, the documentation should also review the capabilities and known limitations of the computational method(s) that were used. The analyst that can demonstrate an understanding of the pitfalls that can occur in a validation process and can clearly address the issues (and defend assumptions) relative to the specific package design of concern will typically use good judgment in specifying the subcritical limit.

10.4.5 Use and Abuse

As indicated in the introduction to this section, computational methods have evolved to the point that detailed geometry models can be analyzed using rigorous solutions of the Boltzmann equation that define radiation transport phenomena. With the inexpensive computing capacity available today, it is now possible for the criticality analyst to produce estimates of k_{eff} for a very wide range of credible conditions and/or parameter variations that can be conceived by the designer. **However, computational methods must be viewed as a tool to aid the criticality assessment.** Criticality safety must be based on reasoned, prudent judgement, not blind adherence to k_{eff} estimates generated for a calculational model. On the other hand, this does not mean that k_{eff} results from a validated code should simply be discarded. Instead, the criticality safety analyst must have a knowledge of the pertinent physics as well as a knowledge of the limitations and capabilities of the particular computational methodology selected for use. With this information the criticality safety analyst is better equipped to determine potential inaccuracies in a calculated k_{eff} value and also determine when an unexpected calculated value is demonstrating an important factor of the design or model that was overlooked by the analyst.

As noted above, a strong dependence is placed on Monte Carlo codes to provide detailed models of a transport package and accurate estimates of k_{eff}. Calculations made using Monte Carlo methods inherently contain uncertainties because of the statistical nature of the processes that are being simulated. Every k_{eff} value from a Monte Carlo code should be reported with a standard deviation, σ. Nearly all Monte Carlo codes provide an estimate of σ. A better value for the standard deviation, however, is obtained by repeating the calculation with different valid random numbers and using this set (five to ten calculations) of k_{eff} values to predict σ.

Theoretically, as the number of particles tracked increases, the size of the standard deviation will decrease regardless of how close the result is to the "correct" answer. Identifying error (the

deviation of the calculated result from the correct answer) is different from determining whether a calculation has adequately converged around "some" answer. Different random number sequences, various neutron starting distributions, and other methods can be used to investigate whether the answer given by a calculation is reproducible and therefore more likely to be near the "correct" answer. But the correct answer is almost always unknown, so one must look rather exclusively at calculational convergence as the criterion for sufficiency.

For most Monte Carlo codes, several pieces of information are given in the output which are useful when determining calculational convergence, including among others:

1. k-effective by generation run,
2. plot of average k-effective by generation run,
3. final k-effective edit table by generation skipped,
4. plot of k-effective by generations skipped, and
5. frequency distribution bar graph (see example of Fig. 10.4).

Conditions that may cause the questioning of convergence include the following:

1. trends (upward or downward) in k_{eff} by generation run over the last half of generations run,
2. trends in k_{eff} by generation for the first half of generations skipped,
3. sudden changes of greater than 1 standard deviation in either k_{eff} plot,
4. abnormally high- or low-generation k_{eff} (±20% of calculated mean), and
5. a calculated result that is not consistent with expectations.

Calculational convergence may be improved through various means: running more histories, starting with an initial neutron distribution in the most reactive region of the model, or using biasing techniques. References 40 and 41 provide excellent practical discussions on the uncertainties associated with Monte Carlo codes for calculating k_{eff} and offer advice on output features and trends that should be observed by the user.

The statistical nature of Monte Carlo codes typically makes the computational method impractical for accurately determining small changes in k_{eff} due to changes in variables. A k_{eff} change that is statistically significant is typically needed to actually indicate a true increase or decrease in k_{eff}. Thus analysts must often use deterministic codes to investigate small changes in k_{eff} due to small parameter changes. As noted in Sect. 10.4.1, deterministic codes are also often used for bounding analyses where simpler geometries can be demonstrated as adequate. The user of deterministic codes must ensure that the mesh spacing and angular quadrature are adequate for the case under consideration. The easiest way of ensuring proper space and angular discretization is to increase the discretization and check for an impact in the k_{eff} result.

10.5 CALCULATIONAL MODELS AND RESULTS

10.5.1 Analysis Approach

The purpose of the criticality safety evaluation is to demonstrate the subcriticality of a single package and an array of packages subject to Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) and to determine the TI for criticality control purposes. Modeling an exact representation of the shipping packaging and its contents is usually impossible and unnecessary. The calculational models that are developed, however, must be of sufficient

Frequency for generations (inadequate)

Frequency for generations (adequate)

Fig. 10.4. Examples of an inadequate (top) and adequate (bottom) expected distribution (normal) for the k_{eff} values by generation in a Monte Carlo generation.

detail to describe explicitly the physical features that are important to the nuclear criticality calculations. Bounding models (that yield the maximum k_{eff}) for the single package and array of packages subject to the specified regulatory conditions (see Sect. 10.2) must be developed. To demonstrate the bounding nature of the model, several calculational models are typically needed to allow consideration of conditions that might occur during normal and accident conditions of transport. Examples of conditions that should be considered are

1. the movement of package contents, including fissile material, neutron absorbers, and structural elements;
2. the increased package interaction due to the loss of packaging material during the accident;
3. the potential for a change in form of any of the package contents or packaging materials; and
4. the effects of temperature changes on the package material and/or the neutron interaction properties.

The actual calculations necessary will be governed by the various parameter changes and conditions that must be considered and will be influenced by the packaging design and features, the contents, their susceptibility to damage, etc.

The nuclear criticality evaluation is required to address NCT and HAC with specific requirements for both single packages and for an array of packages. By judiciously developing the calculational models representing the as-shipped configuration, one can minimize the number of different

models necessary to analyze normal and accident conditions. If no significant dimensional changes are caused by accident conditions, material changes (such as water flooding and reflection) and the number of packages in an array may be the only variables that require investigation. If significant dimensional changes (e.g., contents shifting in the inner cavity of the packaging or external deformation affecting spacing) are necessary, additional calculational models and figures may be needed.

Dimensional tolerances of the packaging and contents should be addressed. When the calculational models are being designed, tolerances that tend to add conservatism (i.e., produce higher neutron multiplication factors) should be included. For some situations, it may be necessary to evaluate the effects of including the maximum, as opposed to the minimum, tolerances. For example, the steel wall of a drum-type container will function as a neutron reflector for a single package but will function as a neutron absorber in an array of packages. Adding the plus tolerance to the nominal wall thickness may be conservative for single-package calculations but may be nonconservative for array calculations.

The following discussion presents a logical, generic approach to the calculational effort based on the regulatory requirements discussed in Sect. 10.2. Two series of calculational cases should be performed: a series of single-unit cases and a series of array cases. Subsets of the array series for different size arrays may also be necessary. Each array series should include calculations for NCT (i.e., undamaged packages in an unmoderated array) and HAC (i.e., damaged packages in a moderated array).

10.5.1.1 Single-Unit Series

The single-unit (i.e, a single package) series of calculations are necessary not only to demonstrate the evaluation requirements of 10 CFR 71.55 but also to provide valuable reference points for subsequent calculations involving variations of certain parameters and contingency conditions that will help characterize the package for the array evaluation.

The single-unit series should start with an undamaged, as-shipped package. The remaining single-unit cases should systematically and progressively reflect and flood the package to represent certain normal and accident conditions. If HACs cause damage to the package or movement of the contents or packaging material, then damaged configurations must also be considered. If a package has multiple containment boundaries, flooding each boundary consecutively should be considered. As noted in Sect. 10.2.1, for NCT the containment system must be reflected unless it is demonstrated that inclusion of the packaging material surrounding the containment provides for a higher k_{eff} value. The containment system as defined in the regulations is "the assembly of components of the packaging intended to retain the radioactive material during transport." It is possible to conceive a package design where the containment designed to prevent leakage (an inner sealed unit) may be only one of many in a package and the criticality control is based on separation of units or interspersed neutron absorber material. In such situations, the wording of the regulations may be confusing. Although the intent of the transportation regulations is not to govern the safe handling of a containment system outside the package, the analyst and designer must consider safety issues during loading and unloading.

The final case of the single-unit series will represent a package (or the containment if demonstrated to be optimum for NCT) completely flooded and reflected. Variations in the flooding sequence may be necessary, such as partial flooding, considering the package in the

horizontal and vertical orientations, flooding (moderating) at less than full-density water, progressively flooding regions from the inside out, etc. The primary objective of the single-unit cases is to show that a single package is subcritical under normal and accidental conditions, under conditions specified by 10 CFR 71.55, and to identify the specific conditions that produce the highest neutron multiplication factor. Packaging designed for different fissile material loading configurations (including partial load configurations) will require a similar approach for each different loading, unless the contents model was developed to encompass the different loadings.

The systematic, progressive order to parametric variation of the single-unit cases, combined with the review of certain calculated nuclear characteristics (e.g., the neutron leakage fractions, the average neutron energy group causing fission, etc.) will greatly enhance the understanding of the neutron physics and interaction potential of the package to varying conditions. The results of the single-unit calculations can greatly influence the approach to and the number of calculations required for the array series calculations. This approach is especially important if there are different content loading configurations. By thoroughly evaluating each different loading configuration and determining the most reactive conditions for each loading, it may be possible to minimize the number of array cases needed. Demonstration that the contents model provides a k_{eff} value that bounds all the different possible package loadings is required and may be a difficult task for packages designed for a variety of fissile material loadings.

It must be understood that 10 CFR 71.55, "General Requirements for all Fissile Material Packages," has three different performance standards for the single package. Paragraph 71.55 (b) requires consideration of water in-leakage into a single package to the point of the most reactive credible configuration, regardless of the water-tightness of the containment boundary(s). Paragraph 71.55 (d), for NCT, presumes that there will be no in-leakage into the containment system (unless it has been assumed that the packaging is not watertight and moderation is already present to cause maximum reactivity). Paragraph 71.55 (e), for HAC, requires consideration of water moderation to be consistent with the damaged conditions. Therefore, packages with a containment boundary that remains water-tight under normal and accident conditions need not be evaluated with internal moderation from water leakage into the containment system for the array calculations. Even though undamaged and damaged conditions must be specifically addressed per 10 CFR 71.55 (d) and (e), the requirements of 10 CFR 71.55 (b) usually result in a higher k_{eff} for the single unit, unless the accident conditions alter the fissile material configuration. Consequently, the most reactive conditions from the single-unit calculations may not be the most appropriate for the array evaluations. Thus an extensive investigation of the single-unit nuclear characteristics is necessary before starting the array calculations.

10.5.1.2 Array Series

A primary concern in the series of array calculations is the interstitial package conditions that provide for the highest k_{eff} value. In a package criticality evaluation, numerous conditions exist for which the effects of moderation must be investigated, such as (1) moderation from hydrogenous packing materials, which are usually inside the primary containment boundary, (2) moderation from hydrogenous materials of construction (e.g., thermal insulation and neutron shielding), and (3) the region between the packages in an array external to the package. The treatment of array moderation can be very easy or very complex, depending upon the type of placement of the materials of construction and their susceptibility to damage from hypothetical accident testing.

In the array calculations for NCT, only the hydrogenous moderator present in the package need be considered [items (1) and (2) above]; moderation between packages [item (3) above] from rain, snow, flooding, etc., is not required per 10 CFR 71.59(a)(1). In the array calculations for HAC, it is necessary to consider all three moderator conditions, including how each form of moderation can change. For a package with thermally degradable insulation, the analyst will need to evaluate both arrays with the undamaged insulation and no additional moderation for NCT and evaluate the effects of reduced moderation from the thermal tests as well as increased moderation from water submersion tests, with water moderation between the packages for HAC. If the inner containment vessel is not a high-integrity, water-tight container, varying degrees of moderation in that region must also be evaluated. For all such conditions and combinations of conditions, the optimum degree of moderation must be found and shown to be adequately subcritical.

Ideally, the first series of array calculations should be for an infinite array. If the infinite array is adequately subcritical under normal and accidental conditions, then no additional array calculations are necessary. If either of the normal or accidental conditions are shown to be critical, or if the maximum k_{eff} exceeds the acceptable upper safety limit, a large finite array should be selected (e.g., a $12 \times 12 \times 12$ array) and all cases recalculated. Successively smaller finite arrays (e.g., $10 \times 10 \times 10$, $8 \times 8 \times 8$, $6 \times 6 \times 6$, etc.) may be required until the array sizes for normal and accidental conditions are found to be adequately subcritical. As an alternative, an applicant may start with any array size (for example, one that is based upon the number of packages planned to be shipped on a vehicle) and demonstrate that this array is adequately subcritical. If this number is significantly less than the maximum that might be permissible, had the iterative process just described been performed, then the TI for criticality control may be unnecessarily high and this additional conservatism could penalize future, unplanned operations.

Each array series should start with the undamaged packages in an unmoderated array. Varying amounts of interstitial hydrogenous moderation should be added in all void regions within and between the packages by varying the density of water in these regions. The water density should be varied from zero (unmoderated) to full density (flooded) in increments such that the optimum moderating density is determined, if an optimum actually exists. Usually eight to ten calculations will be sufficient to determine the optimum. By graphically representing the multiplication factor, k_{eff}, as a function of the moderator density, the response or trend of the plot may be determined. If k_{eff} remains constant or continuously decreases as moderator density increases, no optimum exists, and maximum reactivity occurs when it is unmoderated. If k_{eff} increases as moderator density increases and then begins to decrease (or remains constant at a plateau) as moderator density continues to increase, the conditions of optimum moderation occur at the maximum k_{eff}. However, if k_{eff} continuously increases as the moderator density increases and does not peak or reach a plateau before full-density moderator has been achieved, the optimum moderating conditions may *not* have been achieved.

For unmoderated infinite array calculations, the spacing of the packages is unimportant and does not affect k_{eff}. However, as an interstitial moderator is added to the region between packages, the spacing may become very important because of the amount of moderator that may be present. For this reason, it is sometimes convenient to model an infinite array of packages using an array unit cell consisting of the individual package and a tight-fitting repeating boundary. If the k_{eff} response to increasing interspersed moderator density for this array with the units in contact has an upward trend (positive slope) at full-density moderation, the applicant should consider increasing the size of the unit cell and recalculating k_{eff} as a function of moderation density. Increasing the size of the unit cell provides an increased edge-to-edge spacing between packages and makes more volume available for the interspersed moderator.

To illustrate this situation, consider a cylindrical shipping package with a diameter of one unit and a height (or length) of two units. With a tight-fitting cuboid around the cylinder, 21.5% of the cuboid's volume is outside the package and is available for a moderator. By increasing the cuboid's dimensions such that the edge-to-edge spacing between the packages in all directions is 10% of the package diameter, 38.2% of the cuboid's volume is outside the package and is available for the moderator. This small increase in edge-to-edge spacing corresponds to a 126.0% increase in volume available for the moderator. Therefore, if maximum or optimum k_{eff} has not been achieved with the packages in contact, increasing the packaging spacing to permit additional moderation will be necessary.

Figure 10.5 depicts some typical plots of k_{eff} vs interspersed water moderator density, illustrating the moderation, absorption, and reflection characteristics that may be encountered in packaging safety evaluations. These curves represent changes in array moderation for a fixed package spacing. Curves A, B, and C represent arrays for which an array of packages at the selected spacing is overmoderated, and increasing water moderation only lowers (curves B and C) or has no effect (curve A) on the k_{eff} value. Curves D, E, and F represent arrays for which the array is undermoderated at zero water density, and increasing the moderator density causes the k_{eff} value to increase. Then as the water density increases further, neutron absorption comes into effect, neutron interaction between packages decreases, and the k_{eff} value levels out (curve D) or decreases (curves E and F). Peaking effects, such as seen in curves E and F, frequently occur at very low moderator density (e.g., 0.001 to 0.1 fraction of full density). Therefore, care should be exercised when selecting the values of interspersed moderator density to calculate in the search for the maximum k_{eff} value.

As indicated above, optimum moderation conditions for the array of packages represented by curves D, E, and F have been obtained; and the only mechanism that could make the k_{eff} value of curve D, E, and F rise above the value at full-density water is increased reactivity due to increased reflection provided by more interspersed water (i.e., additional spacing between packages). If the array k_{eff} at full-density moderation is less than the k_{eff} of the flooded and reflected single unit, the edge-to-edge spacing of the packages is not sufficient to permit full reflection.

However, for responses such as those illustrated in curves D, E, and F, there is no need to increase spacing and recalculate the array k_{eff} because the maximum k_{eff} of the array will be that of the reflected single unit, or the k_{eff} of the optimally moderated array (i.e., the first local maxima of curves D, E, and F), whichever is larger. For curves A through F, the packages in the array are essentially isolated at full-density moderation and the corresponding k_{eff} will typically be the same (within statistical limits) as the flooded and reflected single-unit case.

Curve G represents an array in which the optimum array moderator density has not been achieved even with full-density water, and the maximum k_{eff} has not been determined. For this situation, the center-to-center spacing of the packages in the array must be increased and all cases should be recalculated. The center-to-center spacing must be sufficiently large for the curve to reach a plateau (like curve D) or to peak and then decrease (like curves E and F).

Many package designs have enough cylindrical shape to their bodies such that collections of cylindrical packages may be arranged either intentionally or by accident conditions into a triangular-pitch configuration. The array density for a triangular-pitch arrangement is about 15.5% greater than that for a square-pitch configuration. Even though some computer codes permit modeling triangular-pitch lattices, the geometry input can be unwieldy, and it is difficult to make simple changes necessary for parametric variations. To avoid the difficulties of modeling

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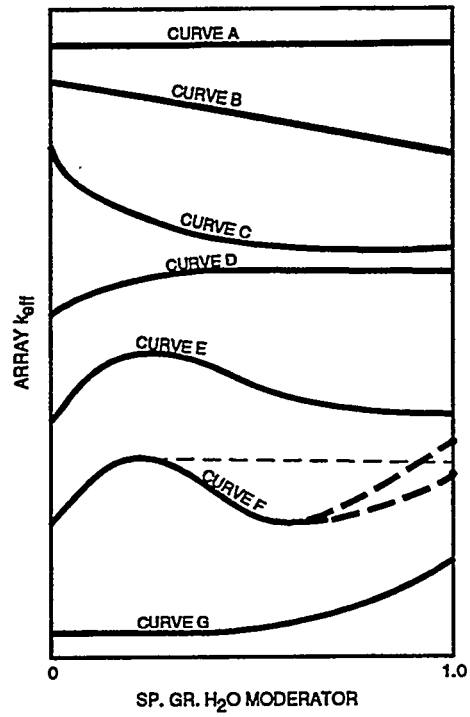

Fig. 10.5. Typical plots of array k_{eff} vs specific gravity of water moderation.

triangular-pitch arrangements of packages, the outside diameter of the single package model can be reduced by 7% to create the array package model. This 7% reduction in diameter produces an array density for the square-pitch lattice equal to the array density of a triangular-pitch lattice of packages with the original diameter.[42]

The diameter reduction will reduce the volume, and consequently the mass, of material in each region where the diameter is altered. Diameter reduction should not be applied to either the contents model or the inner container model. The regional volume reduction of the array package model may be addressed by either of two methods. The first method is to select a region for which the reduction in mass will be conservative for the array calculations. A thick radiation shielding region would be an example. The evaluation must also demonstrate that this regional mass reduction is conservative compared with the single-package model calculations. The second method is to select a region that contains a less-than-full-density material, such as a region of low-density thermal insulation, and to increase the density in the calculations to conserve the actual mass.

The diameter-reduction technique or triangular-pitch modeling is not necessary for shipping packages that exhibit little or no neutron interaction. This condition is often observed for heavily shielded spent fuel casks. If the neutron leakage fraction from a single, unreflected package is less than about 0.25, neutron interaction between packages in an array will usually be insignificant. The best measure of neutron interaction is to compare the calculated k_{eff} of a single package with that of an infinite array of packages. If the infinite medium multiplication factor, k_∞, is within a few percentage points of the single package, k_{eff}, neutron interaction is not a concern and one may conclude that package spacing is not a sensitive parameter. For such packages, use of the full-diameter, single-package model is justifiable for all array calculations, and there is no need to use

either triangular-pitch modeling or the diameter-reduction technique to account for triangular-pitch configurations of packages.

10.5.2 Description of Model and Results

Simplified, dimensioned figures depicting the physical features modeled in the calculations should be provided. Figures drawn specifically for the various portions of the model are preferable to providing the engineering drawings. It is generally simpler and clearer to limit the dimensional features on each figure and to provide multiple figures, with each figure building on the preceding figure. Multiple figures may be required for each calculational model to show adequately, yet simply, the necessary detail. Multiple figures may be necessary for the contents model to show different loading options and for the array packaging model to depict different types of damaged conditions. The dimensions provided on each figure should be the values used in the geometry input for the calculations.

Associated with each figure should be a subsection describing that portion of the model. Differences between the calculational models and the as-shipped configuration should be identified and discussed. The discussion should state why there is a difference, how the different value was determined, and the impact of the calculational results (i.e., conservative or nonconservative).

Each figure showing a portion of the calculational model should have a corresponding discussion that details the material compositions and densities of each material region identified in the figure. If standard or code default densities are used, the values should be stated. If densities other than normal are used, the discussion should state why the density is different and how the altered density was determined. For example, if dimensional differences between a region in the calculational model and the as-shipped configuration result in a significant volume difference in that region, it may be necessary to adjust the material density to conserve the actual mass. Composition differences should also be discussed and justified. For example, a ligneous fiber thermal insulating material may be compositionally represented as cellulose at the same density.

The calculated results should be presented as tables, and at a minimum should provide the case number, a brief description of the conditions, and the calculated results. Providing in the table additional information that supports and/or simplifies the verbal description in the text may also be helpful. No specific format for the tables is required. The format used should be the one that most clearly presents the results and permits easy cross-referencing between the table and the text.

Each calculational model should include a table identifying the material in each region of the model. It is useful to provide in this table additional information, such as the density, the region mass represented by the model, and the actual mass of the region. The dimensions, materials, and masses provided in the figures should be comparable to and consistent with the corresponding items in the engineering drawings and should be the same numerical values used in the input of the calculational method. Frequently, the reference drawings and specifications for the packaging use the English system of measurement (inches, feet, and pounds) rather than the metric system (centimeters and grams), typically required as input to neutronic computer codes. When both systems of measurement are encountered, both values should be included when discussing them in the text, whereas only the metric values as used in the computer code input are needed in tables and figures.

10.6 DESIGN AND SPECIAL ISSUES

Over the last decade the commercial and government sectors involved in handling fissile material for fabrication, processing, storage, and transportation have increased their attention to criticality safety issues. The reasons for this increased attention are nearly universal among the applications:

1. The desire to optimize operations has often raised design k_{eff} values to near the subcritical limits or changed the design basis on which criticality safety has traditionally been predicated.
2. Demands have been increased to provide detailed, defensible documentation for all nuclear safety assessments, of which criticality safety is a principal area.
3. New schemes for use and/or disposal of fissile material (including legacy wastes and excess fissile material from weapons manufacture) have sometimes challenged the demonstrated capabilities of the experimental and computational areas that support criticality safety.

Designers seeking to reduce conservatism in the criticality safety aspects of transportation packages must consider carefully criticality safety issues throughout the entire design process. The difficulty in reducing the bounding conservatism traditionally used in criticality safety often arises in confirming the performance of the package under accident conditions and demonstrating the effect that this performance would have on criticality safety. Interaction with members of the design team responsible for structural, material, and containment aspects of the package design is essential in order for the criticality safety analyst to obtain the knowledge required for making defensible assumptions for the calculational model.

The purpose of this section is to review common design approaches that can provide reduced reactivity and discuss issues associated with their implementation. These design approaches utilize the parameters discussed in Sect. 10.3.1. Typically, some combinations of these control techniques are used in any package design.

10.6.1 Mass and Geometry Control

Limiting the fissile mass of a specific material type to a safe subcritical value under optimum moderation for a selected geometric shape is an excellent control mechanism. The goal is to provide a shape that maximizes neutron leakage. However, the designer must demonstrate that the controlling geometric shape will be maintained under accident conditions and that the fissile mass will not escape or conform to a less limiting geometric shape due to physical or chemical processes. If these conditions cannot be demonstrated, then the criticality safety analyst must assume the worst credible geometric shape for the fissile material under optimum moderation. In addition, the designer must provide measures that ensure the form, mass, and type of fissile material loaded in the package is in conformance with the safety analysis report.

10.6.2 Moderation Control

As noted in Sect. 10.2.2, the package designer should not consider excluding water from the internals of a single package evaluation as a viable design option. However, in many designs it may be appropriate to provide a design that reduces the volume space available for water ingress. This concept of moderator displacement has been used to lower the reactivity of the fissile material contents of a package [i.e., consolidation of fuel pins from an assembly and insertion of steel or aluminum rods in the guide tubes of pressurized-water-reactor (PWR) assemblies].[43] The

obvious problem with these examples is that the contents have to be altered from their normal configuration, and so additional operational and quality assurance procedures must be in place to ensure proper loading.

Water ingress in the array evaluation can be ignored if the single package is shown to prevent water ingress during normal and accident conditions. A demonstration that water ingress can be prevented may be difficult, and the designer must assess this issue prior to incorporating this assumption in the criticality safety evaluation. Also, the moderation aspects of packaging components may have an influence on the reactivity of the package, and it may be beneficial to limit such moderating materials (e.g., plastics, wrapper material, etc.).

10.6.3 Separation of Fissile Material

Criticality control through separation is a viable control strategy that is being considered less and less because of requirements to optimize fissile material loading. Again, a demonstration that the separation can be maintained under accident conditions must be provided in the safety analysis report. Any material that is in the separation region must be ascertained to be present in all potential package conditions, or optimum moderation conditions (caused by water ingress) must be assumed. Potential changes to any spacer material due to chemical or physical processes must be considered. Uncertainty limits that bound any movement of package contents and components must be considered in the criticality safety analysis.

Criticality control via the use of separations designed to enhance neutron absorption were at one time a predominant feature of transportation and storage packages designed for reactor fuel assemblies. The concept, called a neutron flux trap, requires that the walls surrounding the assembly contain a thermal neutron absorbing material and that the walls be separated by a minimum spacing. When the potential for criticality is increased due to water ingress, the concept assumes that the spacing between the assemblies would fill with water and that the effectiveness of the absorber material would be increased due to the thermalization of high-energy neutrons in the water gap. The effectiveness of the flux-trap concept to reduce reactivity is demonstrated in Fig. 10.6 and discussed in refs. 43 and 44. Effectiveness of the concept relies on maintaining the design spacing, uniform water ingress (i.e., no preferential ingress in the assembly voids without ingress in the space between assemblies), and presence of the neutron absorber material.

10.6.4 Uses of Neutron Poisons

All materials exhibit some degree of neutron absorption characteristics. Criticality calculations must take credit for the neutron absorption in fissile material to demonstrate subcriticality. Traditionally, neutron-absorbing materials are divided into two categories: materials of construction and neutron poisons.

Materials of construction include all materials normally present, such as the steel in the inner and outer containers, the thermal insulation, the packing material, etc., and also include the fissile material itself. These materials are usually guaranteed always to be present by virtue of their function. Neutron poisons, on the other hand, are intentionally added, specifically for the purpose of absorbing neutrons to reduce neutron reactivity or to limit neutron reactivity increases during abnormal conditions. The principal concern with relying on neutron absorption by poisons (as opposed to relying on neutron absorption by the materials of construction) is ensuring its presence.

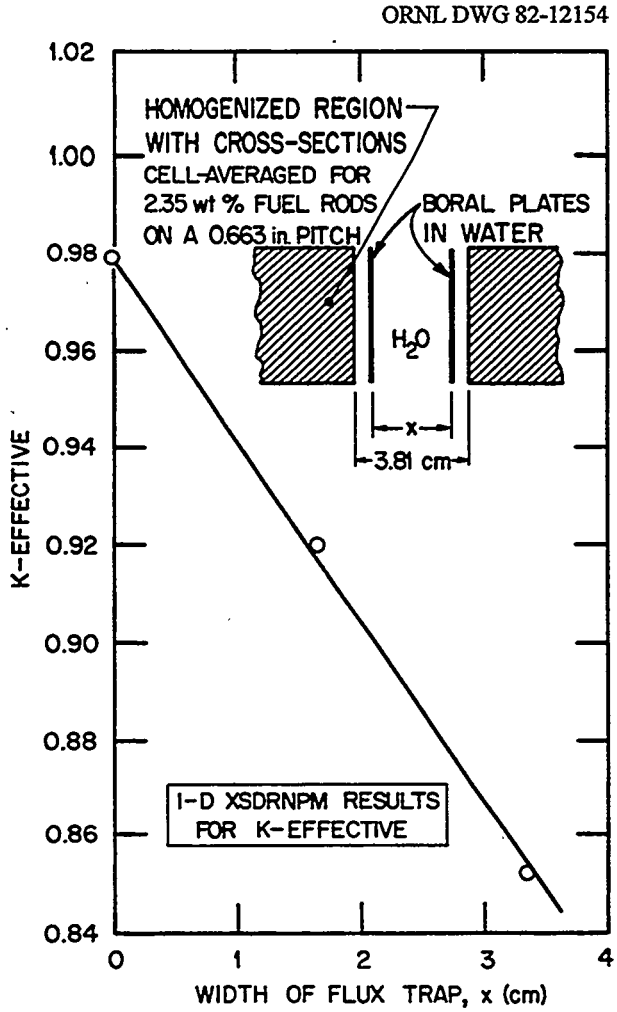

Fig. 10.6. Reactivity change produced by a 1-D neutron flux trap, as a function of the amount of moderator between the borated plates. *Source*: ref. 44.

Therefore, special attention is always required to guarantee both its presence and the proper distribution of the neutron-absorbing material over the assumed life of the package.
Physical, chemical, and corrosive mechanisms must be considered as potential mechanisms for absorber loss. Loss of absorber material through direct neutron absorption (and, so transmutation to a nonabsorbing isotope) is inconsequential because any measurable depletion would take on the order of millions of years of routine operation due to extremely low flux levels in a subcritical system (less than 10^{13} neutrons/year in a loaded PWR package at $k_{eff} = 0.95$).

A thorough criticality evaluation will obviously address the effectiveness of the neutron absorption properties of **all** the materials under both normal and accident conditions (e.g., changes in neutron absorption cross sections as a function of temperature, neutron energy spectrum, the distribution of the absorbing materials for different accident conditions, etc.). The issue of accidental omission of a required neutron poison that is added during package loading may be a credible contingent condition not addressed in the evaluation because it is a container loading issue and not a transportation hypothetical accident condition. If the criticality evaluation determines that a neutron poison is required to ensure subcriticality, the analyst must ensure that the appropriate requirements are included in the packaging operations, acceptance testing and maintenance, and quality assurance chapters of the safety analysis report.

The most common neutron poison used in packaging design is boron. Naturally occurring boron has nearly 20% ^{10}B, which has an extremely high thermal neutron absorption cross section of more than 3500 barns. Figure 10.7 shows a plot of k_{eff} as a function of the natural boron surface density in the 1-cm-thick basket walls of a conceptual 21-assembly PWR transport package (no flux traps). As evidenced from this plot, the benefit of adding boron to the basket becomes marginal as the boron amount is increased. The use of boron enriched in ^{10}B provides an increased efficiency of neutron absorption with an increased cost due to the enrichment process. For a number of years the NRC staff has required package designs to take credit for only 75% of the poison provided in the design. The basis for this number is established in work done to demonstrate that boron carbide granules embedded in aluminum permit channeling of a beam of neutrons between the grains and reduce the effectiveness for neutron absorption. The experimental work of refs. 45 and 46 show that for a monoenergetic neutron beam, the granulated boron carbide areal density of 0.040 g/cm^2 ^{10}B is equivalent to a homogeneous areal density of 0.033 g/cm^2 ^{10}B. The efficiency of boron as a neutron absorber allows credit for only 75% of the poison to be a manageable value for most transportation package designs. However, this value is conservative for several reasons: (1) many neutron poisons are distributed homogeneously through a component of the packaging and are not distributed in a granular fashion, and (2) the experimental work is based on the use of a monodirectional beam of neutrons, while in most package designs an isotropic source of neutrons will be impinging on the wall (thus reducing the potential for intragranular transmission).

When neutron poisons are necessary, it is advisable to incorporate them as intrinsically as possible into the normal materials of construction and verify their presence by a measurement. For example, boron fixed in an aluminum or steel matrix could be used for the inner container (basket) to reduce the neutron interaction between packages (provided it is structurally/thermally acceptable), or cadmium could be plated on the inside surface of the inner container. However, verifying (and reverifying at some frequency) that the absorbers are indeed present, in the prescribed quantity and distribution, could present operational and quality assurance problems that should be addressed in the safety analysis report.

When highly effective neutron-absorbing materials are an integral part of the contents to be shipped, they generally are not considered to be neutron poisons (since by previous logic, they are part of the materials of construction). If subcriticality of the shipment is dependent upon the presence of these materials, the burden of proof that the materials are actually present in the contents (e.g., fissile waste with known absorbers or control rods in a fuel assembly) and will remain present during all normal and accidental conditions is an evaluation issue (structural, containment, and watertightness), rather than an operational/quality assurance issue. An example of the use of poison rods being inserted in the guide tubes of PWR assemblies is discussed in refs. 43 and 47.

10.6.5 Credit for Irradiation History (Burnup Credit)

The principal mandate for fissile material packages is to ensure subcriticality. Thus for packages where thermal, structural, weight, containment, or radiation protection are the design-limiting issues, there is every incentive to keep the assumptions used in the design basis analysis as simple and as bounding as possible as long as the package design is constrained by other technical issues. For the transport of spent nuclear fuel (SNF), the traditional design basis has been to use the isotopic compositions of the fresh, unirradiated fuel in the criticality safety evaluation. This approach is straightforward, easy to defend, and provides a conservative margin that typically

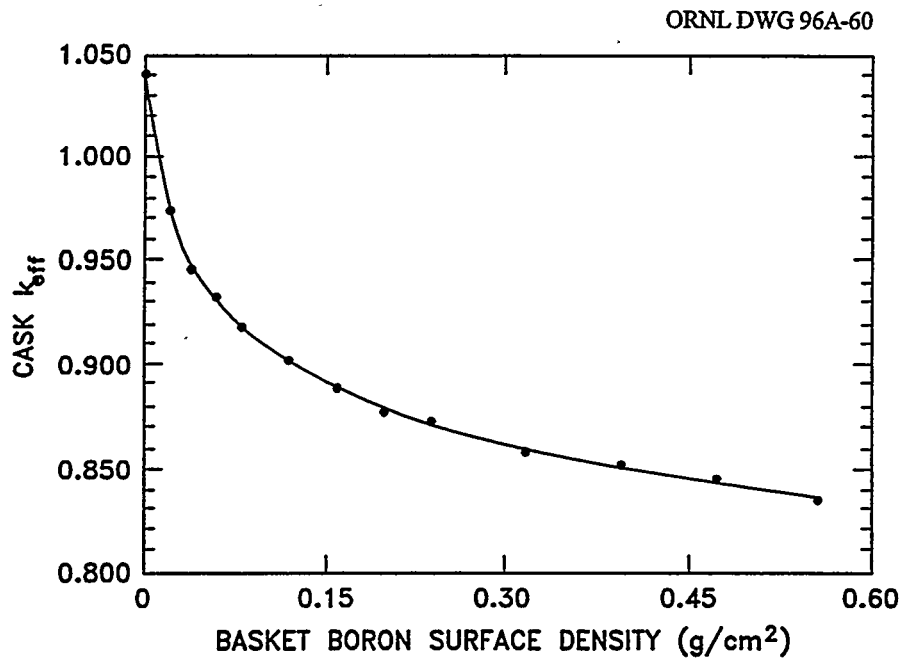

Fig. 10.7. Effect of basket wall boron content on k_{eff}.

precludes any concern about misloading events. With this fresh fuel assumption, the use of neutron poisons and flux traps have historically provided adequate criticality safety control mechanisms without significantly impacting the design criteria.

However, over the next few decades a large inventory of the world's SNF will need to be transported from reactor sites to a permanent disposal facility, an interim storage facility, or a reprocessing plant. A straightforward method for reducing the number of shipments (and so, the associated costs and overall transportation-related risk) is to increase the capacity of SNF in each transportation package while maintaining weight restrictions mandated by existing reactor facilities. In the United States the extended cooling time (minimum of 5 years) now being considered for SNF storage prior to transport reduces the thermal and radiation source per assembly and causes criticality safety control designed for fresh fuel isotopics to become a limiting design consideration as capacities are increased. Even without the move to longer cooling times, the increase in initial enrichment levels causes the fresh fuel assumption to impact package capacity (i.e., existing packages may need to be transported with less than a full load to meet criticality safety requirements). Thus to handle increased SNF capacity in new designs or to enable higher initial enrichments in existing packages, the concept of taking credit for the reduced reactivity caused by the irradiation or burnup of the SNF becomes an attractive design alternative to the fresh fuel assumption. The concept of considering the change in fuel inventory, and thus a reduction in reactivity due to SNF burnup, is referred to as "burnup credit."

Depending on the burnup and initial enrichment, the use of the burnup credit assumption can provide a 20 to 40% decrease in the reactivity of spent PWR fuel compared with that of the fresh fuel. Although the fact that SNF has a decreased reactivity is not questioned, several issues must be addressed and resolved prior to using spent fuel isotopics in the design basis analyses for the criticality safety evaluation. These issues include:

1. validation of analysis tools and associated nuclear data to demonstrate their applicability in the area of burnup credit;
2. specification of design-basis analyses that ensures prediction of a bounding value of k_{eff}; and
3. operational and administrative controls that ensure the SNF loaded into a package has been verified to meet the loading requirements specified for that package design.

The following subsections will provide a brief discussion of these issues with specific references and illustrations to the application with PWR spent fuel. To date, burnup credit in transportation packages has considered only PWR fuel. The reason for this is that the size and reactivity of fresh BWR fuel is such that it appears possible to provide sufficient criticality control using only neutron poisons in the basket wall without impacting the package capacity.[48]

10.6.5.1 Validation of Computational Methods

Section 10.4.4 discussed validation issues related to computational methods for the prediction of k_{eff} of a system. The use of SNF isotopics in the criticality safety analysis means that the computational methods used to predict the isotopics must be validated against measured data. Since there are hundreds of nuclides in spent fuel, it is beneficial to limit the nuclides considered in the safety analyses and the validation effort. The reduced reactivity in SNF is caused by the decrease in fissile inventory and the increase in parasitic, neutron-absorbing nuclides (nonfissile actinides and fission products) that build up during burnup. References 49 and 50 provide information to help identify the important nuclides that affect the reactivity of PWR spent fuel. The SNF nuclides that can be omitted from a safety analysis are the parasitic absorbers that can only decrease k_{eff} further if included in the analysis. Neutron absorbers that are not intrinsic to the fuel material matrix (gases, etc.) must also be eliminated.

After selection of the nuclides to use in the safety analysis, the validation process must begin. Compendiums of measured isotopic data have been produced,[51–53] and efforts have been made to validate computational methods using data selected from these compendiums.[53–55] The measured isotopic data that are available for validation are limited and do not span the full range of burnup, enrichment, cooling time, and assembly locations. Concerns with inconsistency between the cooling times of the measured data and the cooling times of the SNF to be transported are minor because isotopic decay mechanisms are well understood and have undergone extensive verification and validation in numerous nuclear applications. However, paucity of measured data for high enrichment and high-burnup fuel have been concerns raised by technical reviewers. Of further concern is the fact that the database of fission product measurements is a small subset of the actinide measurements. In addition, the cross-section data for fission product nuclides have had much less scrutiny over broad energy ranges than most actinides of importance in SNF. Fission products can provide 20 to 30% of the negative reactivity from burnup, yet the uncertainties in their cross-section data and isotopic predictions reduced their effectiveness in safety evaluations with burnup credit.

The use of SNF isotopics has also raised validation issues relative to the performance of computational methods to predict k_{eff}. The concerns originate from the fact that no critical experiments using spent fuel have been performed in a transport package environment. Experimental data using actual spent fuel are desired in order to demonstrate that the SNF cross sections not occurring in fresh fuel are adequate for the prediction of k_{eff}, the variation in isotopic composition and its influence on k_{eff} can be adequately modeled, and the physics of particle interaction in SNF is handled adequately by the analysis methodology. To address these issues

several efforts are proceeding to analyze existing reactor critical data, as well as fresh fuel criticals that include nuclides found in SNF.[56] These efforts are also aimed at obtaining new experimental data[57–60] that can be used to validate (1) the adequacy of cross-section data (e.g., fission products and actinides with increased importance in SNF) and (2) the analysis methods for burnup credit applications. These efforts are viewed as important steps forward in providing an experimental basis for the validation of calculational methods applied in the safety analysis of a package using burnup credit as a design-basis assumption.

A project associated with validation is that of an international working group sponsored by the NEA of the OECD. This group is using carefully defined calculational benchmarks and independent computational methods to investigate reasons for discrepancies (e.g., differences in fission product cross-section evaluations) between analysis tools and data.[61] These exercises provide simple benchmark problems that aid analysts in understanding potential causes for differences between predicted and measured data.

10.6.5.2 Analysis Uncertainties

The understanding of modeling and parameter uncertainties, together with proper incorporation of these uncertainties in the analysis assumptions, is necessary so that a bounding value of k_{eff} is calculated for a packaging safety analysis that applies burnup credit. Many of these uncertainties can and should be examined as part of the validation process. For example, ref. 62 discusses a procedure to incorporate the variability in the analysis of measured isotopic data and the number of data points to provide a "correction" factor that adjusts the SNF isotopics such that a conservative estimate of k_{eff} can be calculated.

The isotopic composition of a particular fuel assembly in a reactor is dependent, to varying degrees, on the initial isotopic abundance, the specific power, the reactor operating history (including moderator temperature, soluble boron, and assembly location in the reactor), the presence of burnable poisons or control rods, and the cooling time after discharge. Seldom, if ever, are all the irradiation parameters known to the safety analyst; typically the analyst will have to demonstrate the criticality safety of a package for a specified initial enrichment, burnup, cooling time, and assembly type. Data on the specific power, operating history, axial burnup distribution, and presence of burnable poisons must be selected to ensure that the calculated SNF compositions will produce conservative estimates of k_{eff}. Identification of important reactor history parameters and their effect on SNF reactivity have been discussed in refs. 63 and 64. Similarly, ref. 63 discusses the effect of the uncertainty in the axial burnup profile and presents information on the detail required in both the axial isotopic distribution and the numerical input parameters (e.g., number of neutron histories, etc.) in order to predict a reliable value of k_{eff}. A database[65] of axial burnup profiles is being developed and should aid safety analysts in the selection of a realistic, yet conservative profile for the criticality analysis.

10.6.5.3 Operational and Administrative Controls

The use of bounding uncertainties in the validation process and the analysis assumptions should provide assurance that the safety analysis is conservative for the range of initial enrichment, burnup, cooling time and assembly type. For a given assembly type and minimum cooling time (reactivity decreases with cooling time for the first 100 years or so), the safety analysis should provide a loading curve (see Fig. 10.8) that indicates the region of burnup/initial enrichment that

ensures subcriticality. However, operational and administrative controls are needed to establish that the SNF being loaded in the package is within the characteristics used to perform the safety evaluation. Obviously, verification against reactor records is needed to ensure the initial enrichment, burnup, and cooling time of the SNF. Furthermore, current criticality safety practice typically requires demonstration by measurement that a neutron poison (such as the fission products and parasitic actinide aborbers in SNF) is present in the system. Similarly, the regulations call for the safety evaluation to be performed using the "most reactive credible configuration consistent with the chemical and physical form of the material." To satisfy these concerns, the 1996 Edition of IAEA Regulations specifies that a measurement must be performed to demonstrate that the conditions of the SNF are such that the isotopic composition and distribution used in the evaluation have provided a conservative estimate of the package k_{eff} after consideration of all uncertainties and biases. The measurement technique and the resulting detail and accuracy (or sensitivity) provided by the measurement should depend on the amount of available subcritical margin from burnup that is actually used in the safety evaluation.

An example of variability in measurement technique is provided by the French, who currently specify the use of a simple gamma detector measurement to verify burnup credit allowances for less than 5600 MWd/MTU but more direct measurement of fuel burnup for allowance of higher irradiation.[66] For this second measurement, the French rely on two instruments that verify the reactor burnup records based on active and passive neutron measurements. In the United States a measurement device similar to one used by the French has been developed and demonstrated to be a practical method for determining if an assembly is within the "acceptable fuel region" (see Fig. 10.8).[67,68] If the axial burnup profile is identified as an important characteristic of the SNF that is relied upon in the safety analysis, then these measurement devices could also potentially be used to ascertain that the profile is within defined limits.

10.6.5.4 Application of Burnup Credit

As indicated above, France is the only country that has implemented the burnup credit assumption in transportation packages.[66] However, work continues in France and other countries to develop the technical bases and implementation strategy that will enable optimum use of burnup credit

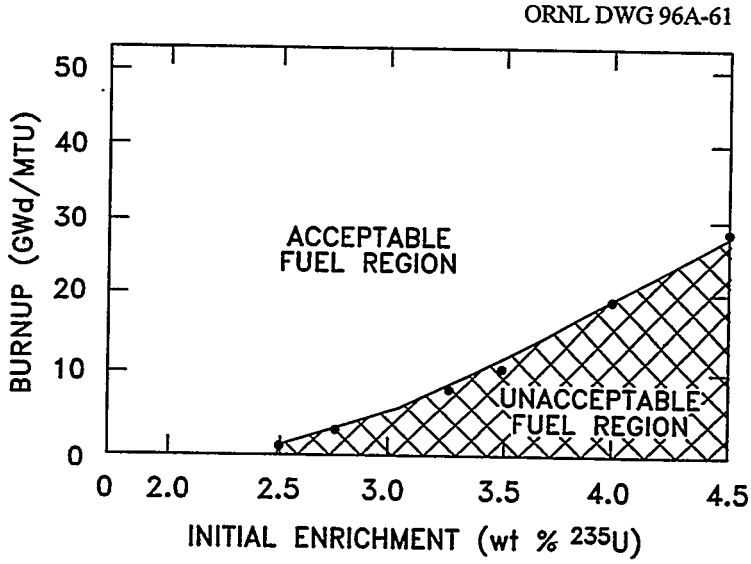

Fig. 10.8. Hypothetical loading curve.

within the existing regulatory criteria for packaging safety. Work to date is directed almost solely at PWR spent fuel. The limitations of burnup credit use in France, together with the experience of the French is documented in ref. 66. Reference 64 provides an overview of the strategy being considered by package designers in the United Kingdom. Similarly, the U.S. Department of Energy has completed a topical report[69] that outlines the technical basis and strategy for the proposed implementation of burnup credit, assuming allowance only for limited actinide isotopics present in the SNF.

Needless to say, the application of burnup credit to an SNF package design is a major change in the safety analysis approach used throughout the history of the nuclear power industry. The experimental database necessary for consistency with other away-from-reactor criticality safety analyses is still under development; however, substantial information and experience related to the application of burnup credit in package design has been gained. No precedent exists in the United States for applying the burnup credit assumption to the criticality safety analysis of a low-enriched SNF assembly. For high-enriched assemblies, limited package loadings have been made using burnup credit in conjunction with neutron count rate measurements that ensure the package load is limited to 75% of the predicted critical mass (see ref. 43). Thus it should be no surprise that the move towards burnup credit has been slow, both in the United States and throughout the world.

10.6.6 Subcritical Measurement Techniques

Direct experimental methods of reactivity determination represent a potential source of excellent verification of criticality safety. The advantage of these methods is that they provide measured evidence of the safety of the actual system during operation. In the case of a fissile material package, the measurement technique would be used during loading operations to measure the k_{eff} of the package as the loading proceeds. Potential measurement techniques[70-73] are the multiplication technique, the Rossi-alpha technique, the ^{252}Cf noise analysis technique, and the source-jerk technique. A comparative review of these techniques is provided in refs. 74 and 75. Each of these techniques has its own set of advantages and disadvantages. Unfortunately, none of the systems works for all systems, under all conditions. Problems of measurement (e.g., in high-gamma radiation fields) and interpretation of data (e.g., necessity to transfer from frequency domain and sensitivity of measurement to detector location) accompany each of the subcritical measurement techniques.

Research work continues to make a practical, reliable subcritical measurement technique. The ^{252}Cf noise technique continues to be studied and applied to a number of systems where information on criticality is desired. Several papers have been written on the potential for this method to be used to ensure subcriticality during loading of an SNF package[76] or to provide a means to perform subcritical spent fuel measurements suitable for use in validation.[77,78]

However, subcritical methods have not been developed to the point of being practical devices for ensuring safe loading of all types of fissile material in all types of packages, nor is there sufficient confidence in the methods to readily apply them in code validation. However, their potential for the ensurance of safety, particularly as package designs move towards higher values of k_{eff}, should be reviewed routinely for application where they may benefit the package designer.

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11. Quality Assurance

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11.1 INTRODUCTION

The U.S. Department of Energy (DOE) is required by 49 CFR 173.7(d) and DOE Order 460.1A to evaluate, approve, and certify packages according to standards equivalent to those used by the U.S. Nuclear Regulatory Commission (NRC). However, DOE has established a slightly different approach for evaluating quality assurance (QA) requirements for packages than the procedure used by the NRC.

The DOE requires the *Safety Analysis Report for Packaging* (SARP) to include packaging-specific QA requirements in accordance with 10 CFR 71.37. DOE Order 5700.6C specifies that a contractor shall have a QA plan established and implemented. The combination of these two regulations have been interpreted by the DOE certifying official to mean that an additional QA Chapter (9) must be included in the SARP. According to NRC *Regulatory Guide 7.9,* the NRC does not require the SARP to include QA requirements, but instead requires each applicant and user to have an NRC-approved QA program meeting the requirements of Subpart H of 10 CFR Part 71 using the guidance provided in NRC *Regulatory Guide 7.10* before an application for packaging certification or use is accepted. The QA requirements for the package are not evaluated directly during the NRC review and approval of each QA program. However, the NRC does perform on-site inspections to verify implementation of QA plans for specific packagings that were developed from the organizations QA program.

In April 1994, 10 CFR Part 830 was passed, identifying QA requirements for DOE managing and operating (M&O) contractors and other organizations that manage DOE nuclear facilities. However, DOE recognized a potential dichotomy between this rule and 10 CFR Part 71, Subpart H. It therefore specifically requested that transportation of radioactive materials be excluded from the definition of nonreactor nuclear facilities, primarily because most transportation activities take place offsite and are governed and regulated by either the U.S. Department of Transportation or NRC (ref. 1). Another primary reason is the statement in 49 CFR 173.7 (d) which indicates that packagings made by or under the direction of DOE may be used to ship Class 7 (radioactive) materials when they are evaluated, approved, and certified by DOE according to packaging standards equivalent to those specified in 10 CFR 71; this includes Subpart H of 10 CFR Part 71. The final rule supported this request, and, as a result, those who design radioactive materials packagings must apply appropriate QA requirements of 10 CFR 71 Subpart H.

The requirements to be included in the QA chapter of a DOE SARP should demonstrate compliance with all applicable DOE Orders and federal regulatory QA requirements during the design, procurement, fabrication, acceptance testing, operation, maintenance, modification, and repair of the packaging. Additionally, they should allow all users of the packaging to develop an effective QA plan meeting all of these applicable requirements. In order to understand the differences between a QA program, a QA plan, and the SARP QA chapter, and the concept of packaging-specific QA requirements, a brief explanation of each is included herein.

11.1.1 QA Program

DOE requires all contractors and subcontractors to have and implement a QA program. For packaging, DOE Order 460.1A requires DOE-approved packaging designers and users (organizations or contractors) to ensure that a QA program is established and that the program addresses the requirements of Subpart H of 10 CFR Part 71 as a minimum. A QA program is usually considered those generic QA requirements that apply to all activities of a specific organization or contractor.

An organization QA program usually consists of a QA manual that details the general QA requirements to be applied to a project as needed and includes site (organization or contractor)-specific quality implementing procedures. This QA program must meet all applicable DOE and regulatory requirements and must be approved by the management of the organization. Most QA programs direct projects and activities to develop QA plans that implement only those QA requirements necessary to achieve project goals and objectives with minimal loss of output.

11.1.2 QA Plan

A QA plan is developed to meet specific project requirements. The establishment of a QA plan for a packaging implies that all important safety activities applicable to the design, fabrication, inspection, testing, purchase, use, maintenance, repair, and modification of the package are implemented with written procedures approved by the management of the organization performing these functions. The QA plan is intended to provide control of the various activities with a graded approach. A graded approach means that the QA requirements should be consistent with the importance to safety of the item or activity. The graded approach will be more fully discussed in Sect. 11.1.5.

A QA plan is unique to a specific organization, such as a company, project or department, and will contain many references to what are referred to as site-specific documents and procedures. These site-specific methods of accomplishing tasks and implementing quality cannot be easily translated from organization to organization, contractor to contractor, or site to site. Therefore, the QA requirements imbedded in these site-specific documents and procedures must be extracted and placed in the SARP so that other organizations can write their own QA plans for the use or modification of the package and be able to meet the QA requirements necessary for the correct use and operation of the package.

In order to have a well-coordinated design, fabrication, and testing program and also to be able to develop an effective QA chapter for the SARP, the package design project (PDP) should develop and implement a packaging-specific QA plan at the earliest possible stage of the design

phase. After developing an effective packaging-specific QA plan, the PDP can easily transfer the QA requirements into the QA chapter of the SARP at any time.

The following is a recommended outline for a QA plan to be used by the PDP and users of the packaging:

I. Introduction/general scope of work

II. List of applicable documents

III. List of acronyms/definitions

IV. Discussion of all quality program elements and requirements to be implemented as part of the QA plan based on the results of the graded approach

11.1.3 SARP QA Chapter

The SARP QA chapter contains only the packaging-specific QA requirements. One significant difference between the QA chapter of the SARP and the organization QA plan concerns packaging-specific versus site-specific information. The packaging-specific QA requirements in the QA chapter of the SARP apply to the certificate holder, packaging custodian, and all users of the packaging. Since site-specific information, with associated internal implementing procedures, is unique to each organization, the QA chapter cannot include site-specific packaging information. A detailed leak-test procedure is an example of site-specific information. Only the important performance requirements are included in the SARP.

The SARP QA chapter establishes QA requirements that the user of the package must impose when developing the QA plan for the operation, maintenance, modification, and repair of the package. Therefore, when developing the SARP QA chapter, the PDP should ensure that all quality requirements are included. QA verification points in procedures are an example of the kinds of information to be included in the QA chapter of the SARP.

11.1.4 Packaging-Specific Information

The understanding of the difference between packaging-specific information and site-specific information is extremely important in the development of the QA plan and QA chapter. The PDP QA plan is developed during the initial stages of the packaging design, while the QA chapter may be written later and then only the packaging-specific information must be extracted from the QA plan and included in the QA chapter of the SARP.

Both the PDP QA plan and QA chapter must include the necessary packaging-specific QA requirements to ensure adequate control over all items and activities that are important to safety as applied to the design, procurement, fabrication, acceptance testing, operation, maintenance, modification, and repair of the packaging. The governing federal regulatory document is Subpart H of 10 CFR Part 71. However, the PDP QA plan may also have to conform to any additional QA requirements invoked by site-specific applicable documents. In order to determine packaging-specific QA requirements, the following documents provide requirements and excellent guidance for developing a QA plan and SARP QA chapter that meets the DOE requirements. In

order to develop an effective QA plan and the QA chapter of the SARP, the PDP should have a working knowledge of all of the following documents and their respective subjects.

Document	Subject
DOE Order 460.1A	Packaging and Transportation Safety
49 CFR Part 173	Equivalency of DOE Certification and QC Requirements for Packaging
49 CFR 173.474	Quality Control for Construction of Packaging
49 CFR 173.475	Quality Control Requirements Prior to each Shipment of Radioactive Material
10 CFR 71.37	Quality Assurance
10 CFR Part 71, Subpart H (10 CFR 71.107-71.137)	QA Requirements for Packaging
ASME-NQA-1	QA Guidance for Nuclear Facilities
NRC *Regulatory Guide 7.10*	QA Guidance for Packaging-Specific Requirements
NUREG/CR-3854	Guidance for Packaging Fabrication
NUREG/CR-3019	Guidance for Welding Criteria
NUREG/CR-5717	Guidance for Packaging Supplier Inspection

11.1.5 Graded Approach

As stated earlier, Subpart H of 10 CFR Part 71 recommends using a graded approach to establish the quality requirements. In other words, the QA requirements applied to important-to-safety items and activities will be determined by the category assigned the item or activity by the design criteria. In addition, the quality requirements should be consistent with the designated packaging category that is based on the amount of radioactive contents being transported.

The packaging category is determined by the PDP during the initial design phase by determining the appropriate package category. A category level of 1, 2, or 3 is based on the curie content of the package from *NRC Regulatory Guide 7.11* or *7.12*. After the curie content is determined, the appropriate category is found in Table 1 in NUREG/CR-3019, which is given below as Table 11.1.

The PDP should first develop a working knowledge of all of the regulatory and guidance documents relative to the necessary quality requirements to satisfy the packaging design requirements. Then the PDP should prepare a QA plan outline that identifies what is to be accomplished, but not necessarily the specific steps of the process. In order to prepare the detailed

Table 11.1. Container safety category

Component type	Container contents safety category		
	Category I	Category II	Category III
Containment	Sect. III NB (Class 1)	Sect. III ND (Class 3)	Sect. VIII Div. 1
Criticality	Sect. III NG (Core Supports)		
Other safety components	Sect. VIII Div. 1 (Vessels) or Sect. III NF (Supports), as appropriate		

QA plan, the PDP must now use the necessary requirements based upon the package category previously determined during the initial design phase by the PDP. The quality requirements are then developed by using a graded approach based on the three-step process relative to the safety and performance function of each item, component, part, or activity as described in Appendix A of *NRC Regulatory Guide 7.10.* A detailed explanation concerning the applications of the graded approach is included in Sect. 11.2.2. The graded approach is discussed at this point to be consistent with the format in Subpart H of 10 CFR Part 71.

11.2 QA ELEMENTS

Each of the QA elements in this section includes the QA requirements from Subpart H of 10 CFR Part 71 under the paragraph entitled "General," while the remaining paragraphs in each section provide guidance from the American Society of Mechanical Engineers (ASME)-NQA-1 and NRC *Regulatory Guide 7.10.* This information provides QA requirements that demonstrate implementation of the requirements of Subpart H of 10 CFR Part 71. The selection of QA requirements from ASME-NQA-1 and NRC *Regulatory Guide 7.10* is based upon the graded approach. The lack of a paragraph entitled "General" in any section indicates that 10 CFR Part 71 does not directly address this element. The 18 QA elements that are discussed herein are in the same sequence as they appear in Subpart H of 10 CFR Part 71, ASME-NQA-1, and in the NRC *Regulatory Guide 7.10.*

11.2.1 Organization

11.2.1.1 General

The initial PDP QA plan that contained all of the required package- and site-specific information and the QA chapter that will contain only the packaging-specific information for both the PDP and all users of the packaging should satisfy all organization QA requirements of Subpart H of 10 CFR 71.103 in the following manner. The PDP, as well as all users of the packaging, are responsible for the establishment and implementation of a QA plan. Any organization may delegate to others, such as subcontractors, the work of establishing the QA plan(s), or any part of the QA plan(s), but the organization must retain responsibility for the QA plan(s). The organization should establish and delineate in writing the authority and duties of persons and other

organizations performing activities affecting the important-to-safety functions of structures, systems, components, and activities. These activities include performing the functions associated with attaining quality objectives and satisfying DOE and federal packaging regulations.

Organization(s) performing packaging design should be structured to ensure that a QA program has been established and an independence exists between the PDP and any quality-verifying functions. The QA functions are to ensure that a QA plan is established and effectively implemented and to verify, by procedures such as test verification, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing QA functions should have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions.

The persons and organizations performing QA functions should report to a management level that ensures that the required authority and organizational freedom is provided, including sufficient independence from cost and schedule. Because of the many variables involved, the types of activities being performed, and the location or locations where activities are performed, the structure of the organization implementing a QA plan can take various forms if the persons and organizations assigned the QA functions have the required authority and organizational freedom. Regardless of the organizational structure, the individual(s) assigned the responsibility for ensuring that the QA plan is being effectively implemented should have direct access to the levels of management necessary to perform this function.

11.2.1.2 Structure and Interface Control

The structure of the organization and the assignment of responsibility for each line function should ensure that (1) the specified quality requirements are achieved and maintained by those who have been assigned responsibility for performing the work and (2) conformance to established requirements is verified by individuals and groups not directly responsible for performing the work. When more than one organization including subcontractors are involved in the implementation of activities important to safety, the authority of each organization should be established. The QA and quality control (QC) functions retained by the QA organization or delegated to other organizations should be identified to ensure that all applicable elements of Subpart H of 10 CFR Part 71 will be implemented.

A formal organizational structure should be established and documented as a part of the QA plan. Organization charts should be used to identify each organizational element that functions under the QA plan (e.g., engineering, procurement, fabrication, handling, shipping, receiving, storage, cleaning, installation, test, inspection, maintenance, repair, modification, and quality assurance). The interface relationships and QA responsibilities of each organizational element, including those of subcontractors, should be identified to demonstrate assignment of responsibilities that satisfy Subpart H requirements. Procedures should be provided for resolving potential conflicts at interfaces between overlapping areas of responsibility. Project-specific organizational charts should not be included in the SARP QA chapter. Only the requirements that define the required organizational charts should be included.

All QA plans pertaining to both the design and use of the packaging and the SARP QA chapter should include requirements to ensure that the QA function has the responsibility and authority to stop unsatisfactory work which includes processing, delivery, or installation of nonconforming

material. This authority should be delineated in writing. All management should be continually involved in quality matters in order to implement an effective QA plan.

11.2.2 QA Program

11.2.2.1 General

Any organization involved with the transport of radioactive material should have an established and DOE-approved QA program. This QA program, however, may not necessarily meet all of the requirements of Subpart H of 10 CFR Part 71. The PDP should therefore establish, at the earliest practicable time, consistent with the schedule for accomplishing the activities, a QA plan that complies with all of the requirements of Subpart H of 10 CFR Part 71. QA plans should be controlled documents. The PDP should implement the QA plan by written procedures and/or instructions throughout the period during which the packaging is being designed, fabricated, tested, and used. The PDP should identify the material, components, and activities to be covered by the QA plan, the major organizations participating, and the designated functions of these organizations.

The PDP, through the QA plan, should provide control over activities affecting the quality of the identified materials to an extent consistent with their importance to safety, and as necessary to assure conformance to the approved design of each package used for the shipment of radioactive material. The PDP should assure that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of equipment and suitable environmental conditions for accomplishing the activity, such as adequate cleanliness, and assurance that all prerequisites for the given activity have been satisfied. The PDP should take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test.

The PDP should base the requirements and procedures in the QA plan on control and verification needs, and on the following considerations concerning the complexity and proposed use of the packaging and components:

1. impact of malfunction or failure of the item on safety,
2. design and fabrication complexity or uniqueness of the item,
3. need for special controls and surveillance over processes and equipment,
4. degree to which functional compliance can be demonstrated by inspection or test,
5. quality history and degree of standardization of the item, and
6. difficulty of repair or replacement of the item.

The PDP should provide for indoctrination and training of personnel performing activities affecting quality, as necessary, to ensure that suitable proficiency is achieved and maintained. The PDP should review the status and adequacy of the QA plan at established intervals. The management of other subcontractors participating in the QA plan should regularly review the status and adequacy of that part of the QA plan which the organization is executing.

The QA plan should include (1) the items, structures, systems and activities to be covered by the plan; and (2) should ensure that the approach used for verifying that the stated items, structures, systems, and activities meet design goals. The QA plan and the SARP QA chapter should include requirements for an orderly evolution from design definition all the way through packaging

decommissioning. As part of this process, timely planning should determine what is to be accomplished; who is to accomplish it; and how, and when, the activities are to be accomplished. The planning should consider the following:

1. quality requirements;
2. requirements for special procedures, work instructions, controls, processes, equipment, tools, or skills;
3. evaluation of specifications, drawings, and other working documents to verify that prerequisites have been met and that the activity affecting quality can be accomplished as specified;
4. documentation required;
5. assignment of responsibility for each task; and
6. methods used to verify conformance to quality requirements.

11.2.2.2 Documentation

The QA plan should ensure that activities important to safety (e.g., nondestructive examination, welding, and testing) that are applicable to the design, purchase, fabrication, and testing of the packaging are described by written procedures and instructions and will be in place prior to engaging in these activities. The documentation should include identification of the material and components covered by the QA plan, the major organizations participating in the plan, and the designated functions of each organization.

To demonstrate that a documented QA plan has been fully implemented by written procedures, a master index of QA procedures related to all activities important to safety and a matrix of those QA procedures that implement each criterion of Subpart H should be established and maintained to reflect the current status of the QA plan. A brief description of the content of the procedures should be included. The following table shows an example of a format for listing QA procedures to demonstrate implementation of a documented QA plan. A similar matrix could be included in the appropriate sections of the QA plan to demonstrate implementation of approved documented activities. Examples are design reviews, procurement document control, and test procedures.

11.2.2.3 Personnel Qualification

The QA plan should provide procedures and requirements for ensuring that personnel performing activities important to safety receive indoctrination and training commensurate with the skill levels required and that qualified personnel within the organization be assigned to determine that functions delegated to principal subcontractors are properly accomplished. Personnel to be trained and indoctrinated should be identified. Indoctrination of personnel for a particular function should cover the technical objectives; the general criteria, including applicable codes, standards, and organizational procedures; applicable QA program elements; and job responsibilities and authority. The required training should be accomplished before the personnel engage in activities affecting important-to-safety items and activities. The extent of the training program should consider the scope, complexity, and objective of the activity and the education, experience and proficiency of the personnel being trained. The proficiency of the personnel should be maintained by retraining, reexamining, and recertifying. Indoctrination and training should be adapted to changes in technology, methods, and/or job responsibilities.

Table 11.2. Format for listing of implementing procedures

Implementing document	Title	10 CFR 71 Subpart H	Description
Quality assurance manual (QAM) Quality procedure (QP) 1	Organization	71.103	Identifies organizations and their relationships in the performance of activities affecting quality
QAM, QP 2	QA program	71.105	Describes basic methods for establishing a documented QA program that implements requirements of Subpart H
QAM, QP 3	Design control	71.107	Describes design control measures established for structures, systems and components
QAM, QP 4	Procurement document control	71.109	Describes procedure for ensuring that applicable regulatory requirements, design bases, and other requirements necessary to ensure adequate quality are suitably included or referenced in documents for the procurement of material, equipment and services
QAM, QP 18[a]	Audits	71.137	Describes internal and external audit programs applicable to both in-house and subcontractors

[a]The information requested for all 18 elements would be listed.

Personnel performing activities important to safety should be qualified based on the skills gained through education, training, and experience. Records of persons performing activities important to safety should include the bases on which an individual is qualified to perform a required function. These records should include formal education certificates, employment history, attendance sheets, training logs, and training records that document training history. The implementation of indoctrination and training should also be documented. In addition, for personnel performing special processes, such as nondestructive examinations or welding, procedures and requirements should be established for obtaining proof of their certification to perform the process, the period their certification remains in effect, and the conditions under which recertification would be required. Qualification and certification of nondestructive testing personnel should be accomplished based on guidelines established by such recognized authorities as the American Society for Nondestructive Testing (ASNT), ASME, American National Standards Institute (ANSI), or American Welding Society (AWS).

For important-to-safety activities that require the use of qualified inspection and test personnel, the minimum educational, physical, and training requirements for these personnel should be

specified. Also, written procedures should be established for the qualification of these personnel and for the ensurance that only qualified personnel are permitted to perform inspection and test activities. Additional requirements and guidance for inspection and test personnel and auditors are given in Supplements 2S-1, 2S-2, and 2S-3 and Appendices 2A-1 and 2A-3 of ASME-NQA-1.

11.2.2.4 Controlled Conditions: Important-to-Safety Items and Activities

Controlled conditions should include, in addition to those listed in Sect. 11.2.2.1, the use of proper equipment, applicable codes and standards, and proper work instructions. The assignment of personnel responsible for each task and methods used to verify conformance to these controlled conditions should be documented.

11.2.2.5 Graded Approach

A balanced QA plan, in which the application of QA requirements is commensurate with the importance to safety of each item or activity based upon the design function, should be developed. The PDP should perform a systematic analysis of each item, activity, structure, and system of the package to assess the consequence of malfunction or failure. In particular, the analysis should address the impact of a failure on safety and the environment. The systematic analysis and QA plan development should be initiated as early in the design process as practicable and should be in accordance with approved procedures. Establishment of an engineering basis for the formulation of a QA plan early in the design process enables a uniform, consistent application of QA requirements during design, fabrication, use, and maintenance of the packaging.

Applying realistic QA requirements to packaging items, components, parts, and activities is necessary to ensure consistency with their importance-to-safety function relative to design and performance requirements. An acceptable method to the DOE Certifying Official for establishing realistic QA requirements is as follows:

1. Classify each item, component, part, and activity relative to whether it is important to safety based on containment, shielding, and subcriticality requirements in 10 CFR Part 71.
2. Group items that are classified as important to safety into quality categories.
3. Specify QA requirements for each quality category.

A logical sequence in establishing QA requirements is a "graded approach" based on categorizing each item, structure, system, or activity as important-to-safety or not important-to-safety ("Q" or "non-Q"), grouping the "Q" items into quality categories, and specifying QA requirements applicable to each quality category.

The important-to-safety Q items are subject to the requirements of Subpart H of 10 CFR Part 71. Those items with a critical impact on safety can be assigned a "1" category, items with a major impact on safety a "2" category, and items having a minor impact on safety to a "3" category. Quality categories are defined as follows.

1. QA Level 1 (Critical) Category
   Items or quality-affecting activities whose failure or malfunction would directly result in an unacceptable condition of containment, shielding, or subcriticality.

2. QA Level 2 (Major)
   Items or quality-affecting activities whose failure or malfunction could indirectly result in an unacceptable condition of containment, shielding, or subcriticality. An unsafe condition could result only if the failure of a QA Level 2 item occurred in conjunction with the failure of another QA Level 2 item.

3. QA Level 3 (Minor)
   Items whose failure or malfunction will not reduce the packaging effectiveness and will not result in an unacceptable condition of containment, shielding, or subcriticality.

After determining the QA category for each important-to-safety item and activity, the QA requirements for design, procurement, fabrication, testing, operation, maintenance, modification, and repair activities are determined from the 18 QA elements identified in Subpart H of 10 CFR Part 71 and ASME-NQA-1. All 18 elements need to be addressed in the SARP and include organization; quality assurance program; design control; procurement document control; instructions, procedures, and drawings; document control; control of purchased material, equipment, and services; identification and control of material, parts, and components; control of special processes; inspection control; test control; control of measuring and test equipment; handling, shipping and storage control; inspection, test and operating status; control of nonconforming materials, parts, or components; corrective action; QA records; and QA audits.

The objective of using the graded approach is to lessen the number of QA requirements from the most stringent QA requirements for QA Level 1 items and activities to the least stringent QA requirements for QA Level 3 items and activities. Refer to Appendix A of NRC Regulatory Guide 7.10.

11.2.3 Design Control

11.2.3.1 General

Design control is based on the following three principles:

1. The design is based on satisfying regulatory and performance requirements by using proven codes and standards.
2. The design is proven by design reviews, alternative calculations, or qualification testing.
3. Design control prevents unauthorized, unanalyzed, and untested changes to the design.

In order to implement design control properly, the PDP should establish the necessary design control requirements in the initial QA plan and later transfer the packaging-specific QA requirements into the SARP QA chapter.

In order to properly implement design control, the PDP should establish procedures to ensure that applicable regulatory requirements and the package design, materials, and components are correctly translated into specifications, drawings, procedures, and instructions. These procedures and requirements should include criteria to ensure that quality standards are specified and included in design documents and that deviations from these quality standards are controlled. Procedures and requirements should be established for the selection, review, and suitability of materials, parts, equipment, and processes that are essential to the important-to-safety functions of the packaging.

The PDP should establish procedures and requirements for the identification and control of design interfaces and coordination among participating design organizations. These requirements should include the establishment of written procedures for design organizations participating in the review, approval, release, distribution, and revision of documents involving design interfaces. The design control requirements should provide for verifying or checking the adequacy of design, by methods such as design reviews, alternative or simplified calculational methods, or by a suitable testing program. For the verifying or checking process, the PDP should designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. In this manner, independence of the design process is demonstrated. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, the PDP should include suitable qualification testing of a prototype or sample unit under the worst-case design conditions. The PDP should apply design control procedures and requirements to such items as the following:

1. criticality physics, radiation shielding, stress, thermal, hydraulic, and accident analyses;
2. compatibility of materials;
3. accessibility for in-service inspection;
4. maintenance and repair;
5. features to facilitate decontamination; and
6. delineation of acceptance criteria for inspections and tests.

The PDP should subject design changes, including field changes, to design control procedures and requirements commensurate with those applied to the original design. Changes in the packaging service conditions, configuration, or contents specified in the package Certificate of Compliance require supplementary approval by the DOE Certifying Official.

11.2.3.2 Design Input

Procedures should be established to ensure that design bases, performance requirements, regulatory requirements, codes, and standards are used in the design of the packaging. In the absence of such codes and standards for formulation of the design activities, alternative approaches should be identified. Procedures should also be instituted to ensure that all design parameters (e.g., criticality physics, containment, and radiation shielding) have been properly considered, reviewed, and approved by the responsible design organization. Procedures should ensure that the parameters are in accordance with the applicable performance codes, standards, and regulatory requirements and also that maintenance, repair, in-service inspection, handling, storage, and cleaning requirements are specified in design documents. Packaging performance requirements include factors such as capacity, radioactivity source term, and thermal load; delineation of interfaces among components and assemblies; materials criteria such as compatibility and corrosion resistance; handling, cleaning, and shipping requirements; and QA requirements.

11.2.3.3 Design Process

Procedures and requirements such as "classification of characteristics" should be established to ensure that packaging designs are reviewed to emphasize critical parameters that can be controlled by inspections or tests and to identify test and inspection criteria and quality standards. Recognized engineering practices; review and approval requirements; issuance and distribution requirements, including revisions to them; maintaining current "as-built" drawings; and storage

and control of original and master copies should be established to control the preparation of drawings and specifications. Design activities should be specified and documented by the responsible design organization in a timely manner and in detail sufficient to allow the design effort to proceed properly and to verify that the design satisfies in-service requirements. The design activity documents should be adequate to enable packaging design, fabrication, and use.

Changes from accepted quality standards, including justification, should be specified, approved, documented, and controlled. Design methods, materials, parts, equipment, and processes that are necessary to assure packaging performance should be selected and reviewed for the correctness of application. Any documents containing applicable information based on experience should be made available to assigned design personnel. The requirements for the approval of final design documents should allow ready verification that the design input has been adhered to and that the name of each item, component, and assembly is included in the design. Whenever an item, component, or assembly is modified or specially selected "commercial grade" or "off-the-shelf," the distinction should be documented together with details of the modification or special selection.

Design analyses should be planned, controlled, and legibly documented for reproduction, filing, and retrieval. The documents should provide purposes or objectives; methods; results of literature surveys; assumptions including those requiring verification; design inputs including sources; references; units; and review and approval. Also, computer calculations should be identified, including the computer model, computer program name and revision, inputs, outputs, proof of program validation and verification, and justification for program use. The documents should provide enough detail to enable a qualified person to verify the validity of the results without contacting the originator. To facilitate retrieval, the documents should be marked with such identifiers as the name of an item, component, or assembly; originator; reviewer; and date.

Each computer program that is employed in the analysis of the package design should be verified to prove that valid results are obtained from the encoded mathematical model within a specified envelope for each parameter used in the range of analysis. The encoded mathematical model should be validated to prove that correct answers to the physical problem addressed by the analysis are achieved. Computer programs should be controlled to ensure that changes are not made without documentation and approval by authorized personnel. Reverification of controlled programs should be performed after each change, before the change is approved, and before final approval of the change. For a more comprehensive discussion of computer software QA, refer to Sect. 11.3.

11.2.3.4 Design Verification

Design verification should be accomplished by design reviews, alternative calculations, or qualification testing, including the use of computer programs. The PDP QA plan and the SARP QA chapter should specify which design verification method will be used for each item, activity, or component based on importance to safety. Computer programs can be considered for use in verifying the adequacy of design. The methods used for design verification, the results of the verification, and the name of the verifier should be described in writing by the responsible design organization. The immediate supervisor of the designer may perform the verification provided (1) the supervisor is the only technically qualified person capable of verifying the design, (2) the need is documented and approved in advance by line management, and (3) QA audits confirm the effectiveness of the use of supervisors as design verifiers to prevent abuse of using supervisors in this way.

The scope of design verification depends on the graded approach based on importance to safety, design intricacy, amount of standardization involved, state of the art, and likeness to designs already accepted. The verification process set forth herein does not have to be repeated for duplicate designs, but the suitability of the process to meet the specified design inputs should be verified before reuse. The documents needed for reuse of an already accepted design should include references to the accepted design specifications and the design verification process used.

Design reviews should ensure that the design inputs are appropriate and embodied suitably into the design, the assumptions used are based on good engineering principles, the design methods are correct, and the required design input and verification for interfacing organizations are clearly documented.

Alternative calculations (hand or computer-based program) that are used for verifying design analyses should be evaluated for suitability of assumptions, input data used, and method of calculation.

Design interfaces should be designated and controlled by, for example, assigning responsibility and instituting procedures among participating design groups for the review, approval, release, distribution, and revision of documents related to design interfaces. Correspondence relating to design interfaces should be controlled, should identify the status of the design data or document, and should specify incomplete items requiring further review and approval.

Records that show that the design and design verification were performed as specified should be assembled, stored, and maintained using written procedures. The records should include drawings, specifications, revisions, and information that displays the major design steps, including sources of design inputs and results of design reviews.

When a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, suitable qualification testing of a prototype or sample unit under the worst-case design conditions should be included, and a test plan should be written and approved. The tests to be performed should be identified, and the test configuration should be described explicitly in writing. Test results should be documented, reviewed, and approved to ensure that the test requirements have been met. When testing demonstrates that design improvements are necessary, the improvements should be documented and verified to be adequate by, for example, additional testing. Testing performed on scale models or mockups should be justified by the use of both verified scaling laws and error analysis of the results before incorporation in final design activities.

During the sequence of design verification, changes in the final design may result. Consequently, procedures and requirements should be established for ensuring that drawing and specification changes are reviewed and approved by the same responsible functions and organizations that reviewed and approved the original documents. Design changes, including field changes, should be subjected to design control procedures and requirements commensurate with those applied to the original design. These procedures and requirements should ensure that the design analyses for unchanged portions of the packaging remain applicable. After a design has been approved, changes should be minimized and made only when absolutely necessary. All changes should be subject to design control procedures. When an erroneous design requires a substantive change, the design process and verification methods should be evaluated and corrected.

Design verification, if other than by qualification testing of a prototype or preproduction unit, should be completed satisfactorily before the release of the design for procurement or fabrication or its release to other organizations for use in design activities. Design verification may be deferred, provided the justification for this action is documented and the unverified portion of the design output documents are appropriately identified and controlled.

As-built drawings should be used to record critical dimensions, when required. The status of the design should always be readily available to both designers and users.

11.2.4 Procurement Document Control

11.2.4.1 General

The PDP QA plan and the SARP QA chapter should specify technical and QA requirements in order to ensure that adequate specifications and information are included in procurement documents for items, activities, and services. The PDP should establish procedures and requirements to ensure adequate specifications and information are included in the documents generated for procurement of material, equipment, and services, whether purchased by the PDP or by subcontractors. The PDP should require subcontractors to implement a QA plan consistent with the same criteria as the PDP.

11.2.4.2 Preparation and Issuance of Procurement Documents

Procedures and requirements should be established to control the preparation, reviews, concurrences, and approvals of procurement documents. Relevant design bases and other requisites for ensuring acceptable quality should be attached to or referenced in procurement documents. These documents should also specify that suppliers have QA plans that meet the same criteria as the PDP.

While preparing procurement documents, requirements should be based on the complexity and the proposed use of the item or service as follows:

1. importance to safety of a malfunction or failure of the item,
2. complexity or uniqueness of the item,
3. need for special controls and surveillance over processes and equipment,
4. degree to which functional compliance can be demonstrated by inspection and test, and
5. quality history and degree of standardization of the item.

Details of this approach, which can reduce the QA requirements, are given in Appendix 4A-1 of ASME-NQA-1.

11.2.4.3 Content of Procurement Documents

Procedures and requirements should be established to ensure that procurement documents include the following information:

1. A clearly worded statement on the scope of the work.
2. The technical requirements, regulatory requirements, material and component identification requirements, drawings, specifications, codes and standards, special processes, test and inspection requirements.
3. The applicable requirements of Subpart H of 10 CFR Part 71 and the implementation methods described in the supplier QA plan. The supplier QA plan is to be reviewed and concurred with by the purchaser prior to initiation of any activities. Also, when subcontractors are involved, the QA criteria for these procurements are to be specified.
4. The requirements for access to the supplier or subcontractors plant facilities and records for inspection or audit.
5. The identification of the documentation, for example, drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, results of chemical and physical tests on material, to be prepared, maintained, and submitted to the purchaser for approval.
6. The identification of records to be retained, controlled, and maintained by the supplier and of records delivered to the purchaser prior to installation of hardware.
7. The requirements for reporting and approving disposition of nonconformances.
8. The requirements for spare and replacement parts and the technical and QA requirements needed for procuring these parts.

11.2.4.4 Review and Revisions to Procurement Documents

Procedures and requirements should be established to ensure that the review and approval of procurement documents are recorded prior to release and that changes and revisions to these documents are subject to at least the same review and approval as were the original documents. All reviews and revisions of procurement documents should ensure that the documents provided to the supplier result in the delivery of goods and services that comply with the stated requirements.

Revisions that result from bid evaluations or precontract negotiations should be merged into the procurement documents and reviewed prior to contract award to verify (1) that the requirements of Sect. 11.2.4.3 are met; whether any additional or modified design criteria are involved; and, if so, whether the criteria are acceptable and (2) that the effects on the intent of the procurement documents or quality of the goods or services to be procured are allowable.

Document reviewers should have access to relevant data and sufficient knowledge of the requirements and intent of the procurement documents.

11.2.4.5 Replacement Part Procurement

The same controls applied to the procurement documents for original packaging components and assemblies should also be applied to the procurement documents for replacement parts.

11.2.5 Instructions, Procedures, and Drawings

11.2.5.1 General

The PDP QA plan and the SARP QA chapter should include requirements for instructions, procedures, and drawings. The PDP should prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to their importance to safety and should require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings should include quantitative and qualitative acceptance criteria for determining that important-to-safety activities have been satisfactorily accomplished.

11.2.5.2 QA Plan Procedures

Procedures and requirements should be established to ensure that activities which are important to safety are prescribed and accomplished in accordance with documented instructions, procedures, or drawings. It is important that methods for complying with each of the applicable 18 criteria of ASME-NQA-1 and Subpart H of 10 CFR Part 71 be specified in instructions, procedures, and drawings; and that instructions, procedures, and drawings include quantitative acceptance criteria (e.g., dimensions, tolerances, and operating limits) and qualitative acceptance criteria (e.g., workmanship) to ensure that activities important to safety have been satisfactorily accomplished.

Procedures and requirements should also be established to ensure that the QA function reviews and concurs in inspection plans; test, calibration, and special process procedures; specifications; and changes to these activities. Prior to the fabrication of an item, manufacturing plans should be reviewed to obtain the concurrence of the QA function of scheduled witness and hold points during fabrication.

11.2.6 Document Control

11.2.6.1 General

The PDP should establish procedures and requirements to effectively control the issuance and change of documents, such as procedures, and drawings, as well as information documented therein (e.g., specifications and instructions). These procedures and requirements should ensure that documents are reviewed for adequacy, approved for release by authorized personnel, distributed through the use of a controlled distribution list, and used at the location where the prescribed activity is performed. These procedures and requirements should ensure that changes to documents are reviewed and approved for use.

11.2.6.2 Controlled Documents

Each of the controlled documents, identified by the QA plan, should be kept current. As a minimum, control should be exercised over the following items:

1. QA plan;
2. QA and QC manuals;
3. design documents—drawings, specifications, and computer codes;

4. procurement documents;
5. operating, maintenance, and modification procedures;
6. inspection and test procedures;
7. nonconformance reports;
8. design change requests; and
9. corrective action reports.

The use of a master list should be included to ensure that current applicable documents are available at the location where the activity is being performed. Packagings affected by design changes should be reviewed to verify that the package is in accordance with the appropriate revision. Persons or groups responsible for reviewing, approving, and issuing documents and revisions should be identified.

11.2.6.3 Control of Document Changes

Procedures and requirements should be established to ensure that changes to documents are reviewed and approved by the same organization that performed the original review and approval and that the changes are in accordance with configuration control procedures.

11.2.7 Control of Purchased Material, Equipment, and Services

11.2.7.1 General

The PDP should establish procedures and requirements to ensure that purchased material, equipment, and services, whether purchased directly or through subcontractors, conform to the procurement documents. These procedures and requirements should include criteria for source evaluation and selection, objective evidence of quality furnished by the subcontractor, inspection at the subcontractor source, and examination of products upon delivery.

The PDP should have available documentary evidence that material and equipment conform to the procurement specifications prior to installation or use of the material and equipment. The PDP should retain and have available this documentary evidence for the life of the package plus 5 years to which it applies. The PDP should ensure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.

The PDP should assess the effectiveness of quality control by subcontractors at intervals consistent with the packages' importance to safety, complexity, and quantity of the product or services.

Procurement planning procedures should be established in writing to describe each procurement step leading to a contract award for, and receipt of, acceptable items and services. Which procedures are to be accomplished and how and when the procedures are to be performed should be defined. Responsible organizations for each procurement step should be identified.

This planning should be completed as early as possible—and definitely before commencement of the first controlled procurement procedure—to ensure consistency in the procurement effort and at the interfaces. The order of events and the milestones marking event completion should be specified, and the activities described in the rest of this section should be integrated.

11.2.7.2 Supplier Selection

The PDP should establish procedures and requirements for evaluating and selecting procurement sources, including the extent to which QA requirements are applied and the required engineering involvement. Criteria that should be considered include the capability of the supplier to comply with applicable criteria of Subpart H of 10 CFR Part 71, results of the survey of the supplier's facility and QA program, and review of the supplier's previous records and performance. Appendix 7A-1 of ASME-NQA-1 provides additional guidance regarding supplier selection.

11.2.7.3 Bid Evaluation and Award

The PDP should establish procedures and requirements to ensure that designated individuals or organizations evaluate proposed suppliers based on the following criteria:

1. technical considerations,
2. conformance to QA requirements,
3. production capability,
4. past performance, and
5. personnel qualification.

Prior to a contract award, all unacceptable conditions identified during the bid evaluation should be resolved. If any unacceptable condition cannot be resolved before a contract award, a commitment in writing from the supplier should be obtained indicating that resolution will be made at a mutually agreeable date during the contract period.

11.2.7.4 Supplier Performance Control

The PDP should establish procedures and requirements for pre- and postcontract award activities, such as meetings and other communications, to ensure that the supplier understands procurement requirements, including preestablished "hold" or inspection points in the manufacturing process that require inspection approval and release by the QA function before further processing during manufacturing and testing and before shipment. Appendix 7A-1 of ASME-NQA-1 contains further guidance with respect to purchaser-supplier communications.

Additional procedures and requirements should be instituted to ensure that the supplier specifies the methods:

1. to be used for meeting procurement document requirements;
2. to control, process, and approve supplier-generated documents;
3. to ensure that the documents are submitted as specified in the procurement document requirements;
4. to provide receiving and processing of technical, inspection, and test data and to record comparison against acceptance criteria;
5. to specify and process needed change information;
6. to ensure that the methods used to control procurement document changes are defined, implemented, and documented; and
7. to specify the procedure for exchanging documents with the supplier.

11.2.7.5 Verification Activities

The extent to which source surveillance during fabrication, inspection, testing, and shipment is performed should be established to ensure conformance with the procurement document requirements. The procedures and requirements should include the instructions specifying characteristics or processes to be witnessed, inspected, or verified; the documentation required; and the identification of those responsible for implementing source surveillance. Verification documentation should be reviewed to ensure that the supplier QA plan is adequate to meet all of the imposed requirements.

The extent to which receipt inspection of supplier-furnished hardware is performed to ensure that items are properly identified and correspond with procurement documentation should be established. When acceptance of an item is contingent on tests after installation in the package, the acceptance documentation should be mutually established with suppliers of the item prior to its use. Receipt inspection should confirm objectively the correctness of configuration, identification, dimensional and physical features, protection from shipping damage, cleanliness, and supplier documentation.

Acceptance of services should be based on such evidence as technical confirmation of data delivered, observation of the work in progress, and adherence to the procurement document requirements.

Additional guidance concerning product acceptance is given in Appendix 7A-1 of ASME-NQA-1.

11.2.7.6 Nonconformance Control

The PDP should establish procedures and requirements to ensure the identification and the proper disposition of items or services that do not meet procurement requirements. These procedures and requirements should also include an evaluation of nonconforming items that have been categorized by the supplier on a deviation form, along with technical justification, recommended disposition (e.g., "use-as-is", "repair", or scrap), and verification that the recommended disposition was implemented.

11.2.7.7 Records

The PDP should establish procedures and requirements to ensure that the supplier furnishes to the purchaser the following records as a minimum:

1. documentation that identifies material or equipment and the specific procurement requirements such as codes, standards, and specifications met by the items; or
2. documentation that identifies any procurement requirements that have not been met, along with a description of those nonconformances designated "use-as-is" or "repair."

This documentation should be maintained by the purchaser for the life of the package plus 5 years.

11.2.8 Identification and Control of Materials, Parts, and Components

11.2.8.1 General

The PDP should establish procedures and requirements for the identification and control of materials, parts, and components. These procedures and requirements should ensure that identification of the item is maintained by heat number, part number, or other means, either on the item or on records traceable to the item. These identification and control procedures and requirements should be designed to prevent the use of incorrect or defective materials, parts, and components.

The established procedures and requirements should also include partially fabricated assemblies, and the criteria to facilitate continued processing, when required inspections or tests have not been completed, in order to maintain physical identity and control over the affected material.

11.2.8.2 Material Traceability

Identification of items should be associated with a specific design or other relevant designating document. Physical identification is preferable when practical and adequate. Otherwise, such means as physical separation or procedural control should be used.

11.2.8.3 Limited-life Items

When replacement of a limited-life item is specified, the PDP should establish procedures and requirements to preclude use of items whose shelf life or prescribed operation time has expired.

11.2.8.4 Markings

The established procedures and requirements should provide the means for physical identification (such as stamping, tags, labels, or lot-follower cards) and traceability to documentation (such as drawings, specifications, or mill reports) throughout fabrication, installation, and use. Identification markings should be easily readable, yet not affect the operation or useful life of the item. When a marked item is subdivided, each resulting part should receive the same marking. Surface treatment or coatings should not be allowed to obscure markings until alternative identification is first provided.

Action should be taken to ensure that identification of stored items is not lost because of handling, aging, corrosion, or mildewing of either markings or records.

11.2.9 Control of Special Processes

11.2.9.1 General

The PDP should establish procedures and requirements to ensure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified

personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

The established procedures and requirements should ensure that the special processes are controlled in accordance with the following criteria.

1. The operations are accomplished in accordance with recorded evidence of verification.
2. The process control is accomplished by use of written procedures, drawings, checklists, and lot-follower cards, to ensure process parameters, including environmental conditions, are satisfied. The documentation should include numerical values of the process parameters, and calibration and equipment criteria.

11.2.9.2 Personnel Qualification

The established procedures and requirements should ensure that personnel are qualified in accordance with applicable codes, standards, and specifications; the specific qualification requirements are included in the process documentation; and the qualification records are established, filed, and kept current.

11.2.9.3 Procedure, Process, and Equipment Qualification

The established procedures and requirements should ensure that procedures, processes, and equipment are qualified in accordance with applicable codes, standards, and specifications; the specific qualification requirements are included in the process documentation; and the qualification records are established, filed, and kept current.

11.2.10 Inspection Control

11.2.10.1 General

The PDP should establish and execute a program for independent inspection of activities affecting quality by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activities. The inspection should be performed by individuals other than those who performed the activity being inspected. Examination, measurements, or tests of materials or products processed should be performed for each work operation where necessary to ensure quality. If direct inspection of processed material or products is not carried out, indirect control by monitoring processing methods, equipment, and personnel should be provided. Both inspection and process monitoring should be made available when quality control is inadequate without both. If mandatory hold points, which require witnessing or inspecting by the PDP-designated representative (and beyond which work should not proceed without the consent of the designated representative) are required, the specific hold points should be indicated in relevant technical documents.

Approved design documents and other relevant technical documents are the main sources for inspection and acceptance requirements. During preprocurement planning, requirements should be established to ensure that inspection procedures, instructions, or checklists include:

1. identification of characteristics and activities to be inspected;
2. acceptance and rejection criteria;
3. identification of the individuals or groups responsible for performing the inspection operation;
4. recording of objective evidence of inspection results;
5. identification of hold or witness points;
6. written approval of data by the supervisor to ensure that all inspection requirements have been satisfied; and
7. prerequisites to be satisfied prior to inspection, including operator qualification and equipment calibration.

Inspection records should identify, as a minimum, the item inspected, inspection date, inspector, inspection description, result, and disposition of nonconformance. When sampling is used to verify acceptability of a group of items, the standard used as the basis for acceptance should be identified.

The PDP should establish procedures and requirements to ensure that inspectors are qualified in accordance with applicable codes, standards, and company training programs; that such qualifications and certifications are kept current; and that inspection personnel are independent from individuals performing the activity being inspected and their supervisors. Inspector trainees should not be allowed to verify conformance until qualified.

11.2.10.2 Receipt Inspection

The PDP should establish procedures and requirements to ensure that items important to safety received by the purchaser meet the requirements specified in the procurement documents. Criteria for the control of accepted items until they are placed in stock or released for use and criteria for the proper disposition of rejected items should also be established.

11.2.10.3 In-process Inspection

The PDP should establish procedures and requirements to ensure that process specifications and their supporting documentation provide for indirect control by monitoring processing methods, equipment, and personnel if direct inspection is impractical. Documented mandatory "hold" points, which halt work until continuation is approved by the designated authority, should not be bypassed.

11.2.10.4 Final Inspection

The PDP should establish procedures and requirements to ensure that final inspection provides for the resolution of nonconformances identified in earlier inspections; that the inspected item is identifiable and traceable to specific records and is adequately protected from physical or environmental damage; and that supervisors review inspection records to verify that all inspection requirements have been satisfied.

Acceptances should be documented and approved by designated authorities.

11.2.10.5 Maintenance/Repair Inspection

All maintenance and repair work performed on a packaging following the final inspection should be reinspected to ensure that the item remains acceptable.

11.2.11 Test Control

11.2.11.1 General

The PDP should establish a test program to ensure that all testing required to demonstrate that the packaging components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements of the QA plan and the requirements and acceptance limits contained in the design criteria. The test procedures should include criteria for ensuring that all prerequisites for the given test are met, that adequate test information is available and used, and that the test is performed under suitable environmental conditions. The PDP should document and evaluate the test results to ensure that test requirements have been satisfied.

Computer software test control is discussed in Sect. 11.3.

11.2.11.2 Requirements

Applicable test programs include prototype qualification tests, production tests, proof tests, and operational tests. Modifications, repairs, and replacements should be tested in accordance with the original design and testing requirements.

Test prerequisites should be properly translated into test procedures that were identified in the design documents, such as instrument calibrations, monitoring to be performed, mandatory "hold" points, suitable environmental conditions to be maintained, condition of the test equipment, personnel qualifications, methods for physical identification of test specimen, methods for documenting or recording test data, and criteria for acceptance.

11.2.11.3 Results

The acceptability of documented test results should be determined by a qualified individual or group. The documentation should identify, as a minimum, the item tested, date of test, tester or data recorder, test description, result and acceptability, disposition of deviations, verifier, and evaluator.

11.2.12 Control of Measuring and Test Equipment

11.2.12.1 General

The PDP should establish procedures and requirements to ensure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified times to maintain accuracy within necessary limits.

Measuring and test equipment should be of the type, range, accuracy, and tolerance to demonstrate that specified requirements have been met.

11.2.12.2 Calibration Control

The PDP should establish procedures and requirements to:

1. Ensure that measurement and test equipment, such as gauges, fixtures, reference standards, and devices used to measure product characteristics, are calibrated, adjusted, and maintained at prescribed intervals or prior to use. The specified methods and intervals of calibration should depend on the stability characteristics of each item, the required accuracy, and the intended use. The measuring and test equipment should be labeled or tagged to indicate the planned date of the next calibration, and the calibration records should be identified and traceable. The equipment should be handled and stored carefully to preserve calibration.
2. Ensure that in-house reference or transfer standards used in calibrating measuring and test equipment are traceable to nationally recognized standards. Calibration standards should have known valid relationships to nationally recognized standards. If no known recognized standard exists, the basis for calibration should be documented.
3. Validate previous inspection and test results up to the time of previous calibration when test and measuring equipment is found to be out of calibration. If any measuring equipment is consistently out of calibration, the equipment should be repaired or replaced.

11.2.13 Handling, Storage, and Shipping

11.2.13.1 General

The PDP should establish procedures and requirements to control, in accordance with instructions, drawings, and specifications, the handling, storage, shipping, cleaning, and preservation of materials and equipment to be used to prevent damage or deterioration. When a particular product requires a special protective environment, such as an inert gas atmosphere, low humidity, or particular temperature levels during handling, storage, or shipping, those environments should be specified and provided.

11.2.13.2 Preservation

The PDP should establish procedures and requirements to specify necessary handling, storage, shipping, cleaning, and preservation procedures to preclude damage or deterioration, including that by environmental conditions such as temperature and humidity. When special handling, lifting, or storage provisions are required, criteria should be established for the use of cranes, shock absorbers, forklifts or other equipment in order to adequately identify and preserve packaging components or assemblies. To ensure that special handling tools and equipment are in acceptable condition and used correctly, they should be inspected and tested periodically and operated only by qualified personnel, as specified by written procedures.

Marking and labeling of items should ensure that, during handling, packaging, shipping, and storage, the items are identified and protected sufficiently, including stipulation of any need for special controls or environmental conditions.

11.2.13.3 Preparation, Release, and Delivery to Purchaser

The PDP should establish procedures and requirements to ensure that a final prerelease review has been completed before release and delivery to the purchaser. This prerelease review should ensure that products (1) are prepared for delivery to the purchaser in accordance with approved drawings, specifications, and government regulations; (2) have passed all applicable inspections and tests; (3) are properly identified by physical markings or tags; and (4) contain operating manuals, maintenance manuals, and generic procedures relating to use.

11.2.14 Inspection, Test, and Operating Status

The PDP should establish procedures and requirements to indicate, by the use of markings, such as stamps, tags, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the packaging. These procedures and requirements should provide for the identification of items which have satisfactorily passed required inspections and tests preclude inadvertent bypassing of the inspections and tests.

The PDP should also establish procedures and requirements to identify the operating status of components of the packaging, such as the tagging of components, to prevent inadvertent operation.

The PDP should establish procedures and requirements to:

1. ensure that the identification of the inspection, test, and operating status of items is known by the function responsible for quality assurance; and
2. control the application and removal of status indicators (such as tags, markings, stamps) and ensure that the bypassing of a required inspection or test or any other required operation is procedurally controlled and under the cognizance of the QA function.

11.2.15 Control of Nonconforming Materials, Parts, or Components

The PDP should establish procedures and requirements to control materials, parts, or components which do not conform to the procurement or fabrication documents in order to prevent the inadvertent use or installations of such equipment. These procedures and requirements should cover the identification, documentation, segregation, disposition, and notification of nonconforming items to affected organizations. Nonconforming items should be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

The PDP should establish procedures and requirements to:

1. identify nonconformances, using such means as Deviating Material Reports that specify detailed processing steps leading to identified item disposition and inspection requirements and corrective action, identify the individuals or groups responsible for approval of the disposition of nonconforming items, and label nonconforming items to provide positive identification yet not jeopardize their eventual acceptability; and
2. segregate nonconforming items and ensure that the items are quarantined or placed in controlled hold areas until proper disposition is completed.

3. dispose of nonconforming items, when necessary, and ensure that the acceptability of each item is verified by reinspecting or retesting the item against the original requirements after designated repair or rework.

Final disposition of nonconformances should be identified, justified, and documented by trained personnel who understand the requirements and have relevant support information available. Any accepted deviations should be identified in the as-built drawings. Nonconformance reports should be analyzed by the QA function to determine quality trends for management review and assessment.

11.2.16 Corrective Action

11.2.16.1 General

The PDP should establish procedures and requirements to ensure that conditions adverse to quality such as deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of a problem that is important to safety, the procedures and requirements should ensure that the condition causing the problem is identified and that corrective actions are taken to preclude repetition.

11.2.16.2 Reporting

Conditions that are detrimental to the quality of items or activities that have been identified as important to safety, including those that result from failures or malfunctions, should be promptly identified and reported to appropriate levels of management. The established procedures and requirements should also ensure that all suppliers provide deficiency and corrective action reports and ensure that follow-up actions, such as root-cause analyses, are documented and submitted to the purchaser to verify that the requirements of the QA plan were implemented and effective.

11.2.16.3 Follow-up

The implementation of corrective action should be verified and assessed to determine the effectiveness. Corrective action status should be monitored, and follow-up reviews, surveillance, or audits should be performed to determine whether corrective actions have been and remain effective.

Additional guidance on corrective action is provided in Appendix 16A-1 of ASME-NQA-1.

11.2.17 QA Records

11.2.17.1 General

The PDP should maintain written records to describe the activities affecting quality. A record system should be established at the earliest practicable time. The records should include the instructions, procedures, and drawings specified in Sect. 11.2.5 to prescribe activities and should include closely related specifications, such as required qualifications and training of personnel,

procedures, and equipment. The records should include the instructions or procedures which establish a records retention program that is consistent with applicable regulations and designates factors such as duration, location, and assigned responsibility. The PDP should retain these records for the lifetime of the package plus 5 years. If any portion of the written procedures or instructions is superseded, the PDP should retain the superseded material as prescribed above.

QA records should provide sufficient information to permit identification of each record with the items or activities to which the record applies. The records should include, as a minimum, design, procurement. manufacturing, and installation records; software testing, including verification and validation; supplier evaluations; nonconformance reports; results of inspections, tests, and audits; failure analyses; as-built drawings and specifications; qualification of personnel procedures, and equipment; calibration procedures; training and retraining records; and deficiency and corrective action reports.

Inspection and test records should contain a description of the observation; evidence of the completion of the inspection or test operation; results of the inspections or tests with data; conditions detrimental to quality; names of inspectors, testers, or data recorders; and evidence of acceptability.

The PDP should establish procedures and requirements to ensure that documents designated as QA records are legible and completed to reflect the work accomplished and are processed quickly to avoid unnecessary delay when the record is needed. To be valid records, the documents should be authenticated by being stamped, initialed, or signed and dated by authorized personnel.

The PDP should establish procedures and requirements to ensure judicious handling of documents that at some time might become records and to facilitate the acceptance of the documents as records.

11.2.17.2 Indexing and Classification of Records

QA records should be indexed to the location and item or activity, and classified as either "lifetime" or "nonpermanent." Lifetime (to packaging decommissioning plus 5 years) records include records pertaining to the design and fabrication of the package and those of a particular item installed in the packaging or stored for future use. These records demonstrate the capability for safe operation; provide evidence of repair, rework, replacement, or modification; aid in determining the cause for an accident or malfunction of an item; or provide a baseline for in-service inspection. A list of typical lifetime records is provided in Appendix 17A-1 of ASME-NQA-1.

Nonpermanent records are those that show evidence that an activity has been performed but do not meet criteria for lifetime records. Records pertaining to the use of a package should be retained for a period of 5 years after the shipment.

11.2.17.3 Receipt, Retrieval, and Disposition of Records

The PDP should establish procedures and requirements to:

1. Provide a receipt control system to include identification of individuals in each organization responsible for receiving records and to assess the current status of the records possessed. The system should provide procedures for specifying required records; expediting submittal of completed records; and receiving, identifying, and inspecting incoming records.
2. Ensure that records maintained in-house or at other locations are identifiable and retrievable and are not disposed of until prescribed conditions are satisfied.

The PDP should establish procedures and requirements to provide for the retrieval of data stored magnetically or optically. If this is impractical, the data should be stored in retrievable form.

11.2.17.4 Storage, Preservation, and Safekeeping

Facilities used to store records should be constructed to minimize the risk from damage or destruction by severe natural conditions such as wind, flood, or fire; temperature and humidity; and infestations of insects, rodents, or mold. Details of storage facilities that provide these features are given in Supplement 17S-1 of ASME-NQA-1.

Records should be firmly attached in binders or placed in folders or envelopes for storage in steel file cabinets. If dual facilities are used to ensure the integrity of records, the facilities should be sufficiently remote from each other to preclude damage to both facilities from a single event such as a fire or flood.

The PDP should establish procedures and requirements to:

1. Preserve special records, such as radiographs and microfilm, from excessive light, electromagnetic fields, and temperature.
2. Preclude the entry of unauthorized personnel into record storage areas.
3. Ensure prompt replacement of a record that is lost or damaged. If such replacement is impractical, consideration should be given to ensuring the quality of the related items by reinspection.

The PDP should establish procedures and requirements to control records that need to be removed from storage to support ongoing work efforts.

Storage procedure documentation should include designation of the procedure enforcement official; storage facility characterization; filing system description: requirements for records identification and acceptability; practices for files access, control, and release; and processes for receiving supplemental information and disposing of superseded records.

11.2.18 Audits

11.2.18.1 General

The PDP should carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the QA plan and to determine the effectiveness of the QA Program. The audits should be performed in accordance with written procedures or checklists by trained personnel not having direct responsibilities in the areas being audited. Audited results should be documented and reviewed by management having responsibility in the area audited. Follow-up action, including reauditing of deficient areas, should be taken when indicated.

A comprehensive audit program should include:

1. ensurance of the authority and organizational independence of the auditors;
2. commitment to adequate manpower, funding, and facilities to implement the audit;
3. identification of audit personnel and their qualifications;
4. criteria for reasonable and timely access of audit personnel to facilities, documents, and personnel necessary for performing audits;
5. use of checklists;
6. methods for reporting audit findings to responsible management of both the audited and auditing organizations;
7. criteria for access by the audit team to levels of management that have responsibility and authority for corrective action; and
8. methods for verification that effective corrective action has been accomplished on a timely basis.

11.2.18.2 Scheduling

The PDP should establish schedules for internal, external, and management audits, and procedures and requirements to ensure that key activities of the QA plan, such as design and fabrication are given priority consideration. For management audits, the schedules should identify the level of management, usually from the corporate office or another division, designated to assess the overall effectiveness of the implementation of the described in-house QA program. The PDP should identify and include the important-to-safety activities, such as procurement and training of personnel, in the audit program.

Internal audits of the applicable elements of the QA plan should be conducted at least annually or at least once within the life of the activity, whichever is shorter. External audits of the elements of a subcontractor or PDP QA plans should be conducted on a triennial basis. The 3-year period should begin with performance of an audit when sufficient work is in progress to demonstrate implementation of a QA plan having the required scope for purchases placed during the 3-year period. Management audits should be conducted at least once every year.

11.2.18.3 Preparation

A plan for each audit should be produced, in writing by the auditing organization, which specifies the audit scope, requisites, audit team leader and members, specific operations to be audited,

groups to be notified, relevant references, schedule of events, and written procedures and checklists.

Qualifications of auditing personnel including the lead auditor, should be established, and the responsibilities of the audit team members and the lead auditor with respect to evaluation and issuance of audit reports should be specified. The auditing organizations are responsible to establish qualifications for prospective audit personnel and for the use of technical specialists to accomplish auditing activities important to safety. The lead auditor should indoctrinate the audit team members prior to the audit.

Specific guidance for determining qualifications for individual auditors and lead auditors may be obtained by referring to Supplement 2S-3 of ASME-NQA-1.

A preaudit conference should be held between the management of the organizations being audited and the team conducting the audit, with the nature and scope of the conference being specified prior to an audit. The purpose of the conference should be to confirm the audit scope and planned dates, to meet counterparts, to discuss the sequence and duration of the audit, to set the time for the postaudit conference, to establish channels of communication, and to prepare an agreed-to agenda for the audit.

11.2.18.4 Performance

Written procedures and checklists should be the basis for conducting audits, and audited activities should be compared with defined requisites to an extent sufficient to verify effectiveness. The audit team should document the audit results for review by management for the audited activities who should be informed immediately of any situations that need immediate corrective action.

The PDP should establish procedures and requirements to conduct a postaudit conference between the audit team and the management of the audited organization to present the results and clarify misunderstandings.

11.2.18.5 Reporting

The PDP should establish procedures and requirements to identify time constraints imposed for issuing audit reports and the requested date for corrective-action response by the audited organization. The lead auditor should sign and issue the audit report, which should provide such information as audit scope, audit team personnel, persons contacted during the audit, concise audit results (including degree of effectiveness of the activities audited), and adverse-findings descriptions comprehensible enough to facilitate remedial action.

The lead auditor evaluates the response to the audit report, which should state the corrective action taken to prevent recurrences of nonconformances. In the event that corrective action cannot be taken immediately, the response of the audited organization should include scheduled dates for initiation and completion of the corrective actions.

11.2.18.6 Follow-up Action

The audit team leader should verify that the audited organization provides a timely response to the audit report, that the response is adequate, and that corrective action has been accomplished within the prescribed schedule. Audit records should contain plans, reports, responses, and status of corrective actions.

11.2.18.7 Records

Audit records should contain the audit plans and reports, written replies, and documentation that corrective action was completed. Audit records should be considered lifetime records.

Additional guidance on audits is given in Appendix 18A-1 of ASME-NQA-1.

11.3 COMPUTER SOFTWARE CONTROL AND ACCEPTANCE

11.3.1 General

This section discusses and provides guidance for computer software QA when used for packaging design, design review, or analyses and does not consider software development or maintenance. The application of specific requirements should be included in the QA plan and implemented by written instructions or procedures. The QA plan should describe:

1. the software product, version number, and usage to which the requirements apply;
2. the organization(s) performing the tasks that achieve software quality and the responsibilities;
3. the required documentation, including user manual and data base description;
4. the standards, conventions, techniques, or methodologies to be used to assure verification and validation;
5. configuration management including the methods for error reporting and corrective action for software versions;
6. the record retention time; and
7. the required software reviews and audits.

11.3.2 Acceptance Requirements

Either the PDP or the computer software user or group should provide the computer software test requirements, and acceptance criteria that should be based on relevant design or other technical documents. The specified tests, such as verification and validation tests, hardware integration tests, in-use tests, and other acceptance tests should be controlled through implementing procedures.

11.3.3 Verification Tests

Verification tests should show that the software (computer program) yields correct results to problems that lie within the allowable ranges of input parameters specified in the program documentation. Acceptable test-problem answers for comparison may be obtained by hand calculations, other similar proven programs, or empirical data and information from the technical

literature. The verification may be of a complete computer program or of each module, depending on the requirements to ensure that acceptable results are produced. Verification tests should be performed with the same computer that will be used for the package calculations, and verification should be repeated whenever hardware or operating system configuration changes are made.

11.3.4 Validation Tests

Validation tests should show that the computer results model reality accurately. For packaging, testing is the primary method for validation. Benchmark problems should be developed and documented to allow confirmation of acceptable software performance on a specific computer and operating system. The benchmark problems should be developed on the basis of experimental tests of prototypes, analytical solutions, and/or problems run with other computer software. In-use test problems should be run periodically to prove continued acceptable performance and whenever the software is installed on a different computer or hardware changes are made.

11.3.5 Test Procedures

Test procedures should stipulate the required tests and the order in which to perform; the time interval between periodic tests; the required ranges of input parameters; the points at which testing is necessary; the rules for defining test cases with test dependencies; the criteria for testing logic branches; the criteria for hardware integration; the expected results; the acceptance criteria and problem reporting procedure; and the reports, records, standard formatting, and protocol.

11.3.6 Test Results

Test results should be documented and reviewed by a responsible expert to ensure that they are acceptable.

11.3.7 Test Records

Test records should include the computer program tested, the computer hardware tested, the test date, the tester or data recorder, simulation models used, test problems, results and acceptability, disposition of deviations, and evaluator. Test records should be considered lifetime records.

11.3.8 Computer Software User Qualification and Analyses Review

Qualification requirements should be defined for users of complex computer software. For example, for Finite Element method (FEM) computer codes, user training and experience is required to develop good models and obtain spatially and temporally converged solutions. Modeling complexities and poor judgement may yield large errors, despite high quality software. Project reviews of analyses results should be conducted by experienced analysts.

More details on computer software QA applied to the overall software development cycle are given in Supplement 11S-2 and Subpart 2.7 of ASME-NQA-1.

11.4 SELECT QA TERMS FROM ASME-NQA-1

Acceptance criteria. Specified limits placed on characteristics of an item, process, or service defined in codes, standards, or other requirement documents.

Audit. A planned and documented activity performed to determine by investigation, examination, or evaluation of objective evidence the adequacy of and compliance with established procedures, instructions, drawings, and other applicable documents, and the effectiveness of implementation. An audit should not be confused with surveillance or inspection activities performed for the sole purpose of process control or product acceptance.

Audit, external. An audit of those portions of another organization's quality assurance program not under the direct control or within the organizational structure of the auditing organization.

Audit, internal. An audit of those portions of an organization's quality assurance program retained under its direct control and within its organizational structure.

Certificate of Conformance. A document signed or otherwise authenticated by an authorized individual certifying the degree to which items or services meet specified requirements.

Certification. The act of determining, verifying, and attesting in writing to the qualifications of personnel, processes, procedures, or items in accordance with specified requirements.

Characteristic. Any property or attribute of an item, process, or service that is distinct, describable, and measurable.

Commercial-grade item. An item satisfying (a), (b), and (c) below:
(a) not subject to design or specification requirements that are unique to nuclear facilities;
(b) used in applications other than nuclear facilities;
(c) is to be ordered from the manufacturer/supplier on the basis of specifications set forth in the manufacturer's published product description (for example, catalog).

Computer program. A sequence of instructions suitable for processing by a computer. Processing may include the use of an assembler, a compiler, an interpreter, or a translator to prepare the program for execution as well as to execute it.

Condition adverse to quality. An all-inclusive term used in reference to any of the following: failures, malfunctions, deficiencies, defective items, and nonconformances. A significant condition adverse to quality is one which, if uncorrected, could have a serious effect on safety or operability.

Corrective action. Measures taken to rectify conditions adverse to quality and, where necessary, to preclude repetition.

Design change. Any revision or alteration of the technical requirements defined by approved and issued design output documents and approved and issued changes thereto.

Design input. Those criteria, parameters, bases, or other design requirements upon which detailed final design is based.

Design output. Drawings, specifications, and other documents used to define technical requirements of structures, systems, components, and computer programs.

Design process. Technical and management processes that commence with identification of design input and that lead to and include the issuance of design output documents.

Design, final. Approved design output documents and approved changes thereto.

Deviation. A departure from specified requirements.

Document. Any written or pictorial information describing, defining, specifying, reporting, or certifying activities, requirements, procedures, or results. A document is not considered to be a Quality Assurance Record until it satisfies the definition of a Quality Assurance Record as defined in this Section.

Guideline. A suggested practice that is not mandatory in programs intended to comply with a standard. The word "should" denotes a guideline. The word "shall" denotes a requirement.

Inspection. Examination or measurement to verify whether an item or activity conforms to specified requirements.

Inspector. A person who performs inspection activities to verify conformance to specific requirements.

Item. An all-inclusive term used in place of any of the following: appurtenance, assembly, component, equipment, material, module, part, structure, subassembly, subsystem, system, or unit.

Measuring and test equipment (M&TE). Devices or systems used to calibrate, measure, gage, test, or inspect in order to control or acquire data to verify conformance to specified requirements.

Nonconformance. A deficiency in characteristic, documentation, or procedure that renders the quality of an item or activity unacceptable or indeterminate.

Objective evidence. Any documented statement of fact, other information, or record, either quantitative or qualitative, pertaining to the quality of an item or activity, based on observations, measurements, or tests which can be verified.

Owner. The person, group, company, agency, or corporation who has or will have title to the packaging.

Procedure. A document that specifies or describes how an activity is to be performed.

Procurement document. Purchase requisitions, purchase orders, drawings, contracts, specifications, or instructions used to define requirements for purchase.

Purchaser. The organization responsible for establishment of procurement requirements and for issuance or administration, or both, of procurement documents.

Qualification, personnel. The characteristics or abilities gained through education, training, or experience, as measured against established requirements, such as standards or tests, that qualify an individual to perform a required function.

Qualified procedure. An approved procedure that has been demonstrated to meet the specified requirements for the intended purpose.

Quality assurance (QA). All those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.

Quality assurance record. A completed document that furnishes evidence of the quality of items and/or activities affecting quality.

Receiving. Taking delivery of an item at a designated location.

Repair. The process of restoring a nonconforming characteristic to a condition such that the capability of an item to function reliably and safely is unimpaired, even though that item still does not conform to the original requirement.

Rework. The process by which an item is made to conform to original requirements by completion or correction.

Right of access. The right of a Purchaser or designated representative to enter the premises of a Supplier for the purpose of inspection, surveillance, or quality assurance audit.

Service. The performance of activities such as design, fabrication, inspection, nondestructive examination, repair, or installation.

Shall. See definition of the term guideline.

Should. See definition of the term guideline.

Special process. A process, the results of which are highly dependent on the control of the process or the skill of the operators, or both, and in which the specified quality cannot be readily determined by inspection or test of the product.

Supplier. Any individual or organization who furnishes items or services in accordance with a procurement document. An all-inclusive term used in place of any of the following: vendor, seller, contractor, subcontractor, fabricator, consultant, and their subtier levels.

Surveillance. The act of monitoring or observing to verify whether an item or activity conforms to specified requirements.

Testing. An element of verification for the determination of the capability of an item to meet specified requirements by subjecting the item to a set of physical, chemical, environmental, or operating conditions.

Traceability. The ability to trace the history, application, or location of an item and like items or activities by means of recorded identification.

Use-as-is. A disposition permitted for a nonconforming item when it can be established that the item is satisfactory for the intended use.

Verification. The act of reviewing, inspecting, testing, checking, auditing, or otherwise determining and documenting whether items, processes, services, or documents conform to specified requirements.

Waiver. Documented authorization to depart from specified requirements.

11.5 REFERENCE

1. U.S. Department of Energy, "Nuclear Safety Management," *Fed. Regist.* **59**(65), 15845 (April 7, 1994).

12. Package Operations

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12.1 INTRODUCTION

The functional requirements and operation of a package must be considered early in its design phase because there is little or no opportunity to change its configuration in any significant way once design and fabrication have been completed. Even during the design phase, as the analysis progresses, it becomes increasingly difficult to change the principal elements that impact operation (such as location and configuration of penetrations) because the cost of the analysis is high and schedule disruptions are not welcomed. The designer must consider that although a package will be designed and fabricated once, it may be operated hundreds of times.

The two most important aspects of repetitive operations with a fixed design, aside from the actual steps required, are operator dose and performance. To address these concerns effectively, the designer must consider how the package will be handled in the user's operating environment. The designer should consider how operator exposure could be reduced, above that achieved by the package shielding, from placement of components or process efficiencies. Package operational requirements (e.g., draining) and package fixtures (e.g., penetration valves) should be designed to reduce the potential for the operator to make an error in opening, closing, venting, draining, or using the package in other ways.

Efficient package operations sometimes require the use of other equipment (e.g., a lifting fixture or special tools) specified by the package designer. To the greatest extent possible, the package should be designed so that the use of special tools or equipment is not required. As the number of special tools and equipment increase, the total system reliability decreases (because the package system requires the use of these special items); operational complexity increases; and the transportation, handling, packaging, and decontamination requirements increase. Reducing or eliminating reliance on unique equipment, tools, fixtures, or fasteners reduces the burden on the package user. Nevertheless, the designer may identify unique equipment that reduces package turnaround time or decreases operator dose and the potential for operator error. Such items merit consideration for inclusion in the ancillary equipment provided to the user.

The package designer must develop procedures that describe the handling, loading, and unloading of the package. The procedures developed for inclusion in Chapter 7 of the *Safety Analysis Report for Packaging* (SARP) typically have a variable level of detail and are referred to as "generic procedures." The designer must ensure that these procedures cover all of the essential steps that are required to ensure that the package is properly handled. They should also contain warning or caution notes to the using facility if package conditions or operations justify their inclusion. The

generic procedures must include suficient detail to enable the regulatory reviewer to understand the operational sequence of steps and reach a conclusion about the adequacy of the procedures.

Generic procedures are occasionally used by the package owner to develop a more comprehensive set of procedures, which are incorporated into an operating manual. Typically, operating manuals include outline drawings of the package, a list of the principal package components, ancillary equipment items, package weight and dimensions, estimated operational handling times, and other information that may be beneficial to a user. Operating manuals are usually developed only when more than one package is to be placed in service and when there are many potential users.

User facilities typically have requirements relating to package handling which must be met while the package is in use. For example, these requirements may include a limitation on the height above a floor that a package may be lifted, ensuring that a given ventilation system or pump is in operation, ensuring that limit switches have a certain setting, etc. Typically, a facility anticipating use of a particular package develops a "facility procedure" that incorporates both the facility requirements and the package-handling requirements. These requirements are integrated to establish a smooth flow of operations. Prior to its being used, any facility procedure should be reviewed by the package owner to ensure that all necessary package-handling (or operational) requirements are fully met.

This chapter addresses the issues discussed previously and other specific aspects of package design that could be important to package operation and use.

12.2 PACKAGE DESIGN CONSIDERATIONS

The operation of a package must be addressed very early in the design process, and operations requirements must be integrated into the design program. There are two key issues: The first is that the package is designed once but may be used hundreds of times. The second is that changes in the design to incorporate specific operational requirements cannot generally be made late in the design process because of cost and schedule issues. In most organizations, changes to facilitate operations would not be considered once the SARP has been submitted to the regulator (unless, of course, the changes were suggested by the regulator). Consequently, operations should be considered in the design in the same detailed way in which the other aspects of the package—structure, thermal performance, shielding, criticality, and containment—are considered.

As design features are developed in support of any package performance requirement, the feature should be reviewed for its impact on both operations and maintenance. In selecting among design alternatives, emphasis should be on reducing operator dose, package-handling time, and the potential for operator error. In the design phase, which may extend over a period of many months, the designer establishes the individual operational steps and tasks that will be performed repetitively in each handling of the package over its likely 20-year life. The designer should carefully consider how the package operational requirements can be met quickly and efficiently and with a low potential for error. These objectives work to support the designers commitment to the "as low as reasonably achievable" (ALARA) principal and to the preservation of the user and public health and safety.

12.2.1 Operator Dose from Radioactive Sources

The dose received by a package operator arises from two sources: (1) the contents of the loaded package, and (2) the background radiation within the user facility. The package designer can directly influence the dose that the user is likely to receive from the first source by the amount of shielding that is designed in the packaging and by careful placement of design features that relate to package operations. The second source is addressed indirectly since its contribution to the dose that the operator receives is influenced by the operational characteristics of the packaging.

If the contents of the package require shielding, the designer sizes and incorporates shielding to ensure that the package (as presented for transport) meets regulatory transport dose rate limits. Frequently, features that the operator must use (e.g., penetrations, lift point attachments, and bolts) are in areas of the package that have high surface dose rates.

The designer can reduce total dose to the operator by adding additional shielding to the package (assuming the maximum gross vehicle, crane hook, or other design weight limitations are not exceeded) or by ensuring that features are placed so that the operator can handle the feature at "arms' length." Frequently, there is little weight margin available because it is often desirable to maximize the payload. Nevertheless, if weight is available, shielding should be added in those areas of the package that the operator must access.

Once the maximum amount of shielding is added, the designer should ensure that penetrations that cannot be placed in the side of the package are at least positioned near the edge in the top of the package. The operator should not be required to lean over the package to access fittings, lifting slings, leak-testing ports, etc. These should be easy to operate in order to reduce the amount of time that the operator is in the higher radiation area. The design of a package and placement of its features, as well as careful consideration of package functional requirements, can lead to reductions in operator dose over that achieved solely from the shielding in the package.

The second element of operator dose, that resulting from exposure to facility background radiation, can be reduced by decreasing the package "turnaround time." The designer should design the package and its ancillary equipment so that it can be handled quickly and efficiently at the user site. Having recognized that background dose accrues to the operator who handles the package and its ancillary equipment, the designer should then limit ancillary equipment to the minimum necessary. Reductions in turnaround times are achieved by the designer's (1) thinking through the handling process, (2) reviewing long-duration handling tasks in order to reduce time and resource requirements, and (3) reducing the amount of time spent handling "nonpackage" items.

By carefully considering opportunities to reduce operator exposure to both the radiation field of a loaded package and the user facility's background radiation, the designer can ensure that the regulatory requirement to achieve ALARA occupational radiation exposure is met.

12.2.2 Design for Operations and Human Factors

Many of the reported problems with packages arise from operator error.[1,2] To reduce the potential for operator error in the use of the package, the designer should consider each of the operational features of the package from a human-factors perspective. Eliminating or reducing the potential for error should improve package system use, reduce regulatory involvement in package use, and improve the health and safety of both the package users and the public.

In regarding human factors, one considers the equipment design, the personnel and equipment interfaces, the environment, the biomechanics of the body, and operator training. During the design phase, the most important of these factors are the equipment design and the personnel and equipment interfaces.

The fixtures of the package and the package ancillary equipment should be designed so that the operator can operate a valve safely; attach a fitting, fixture, or closure; verify conditions; and take any other actions needed to use the package correctly. Penetrations, such as test ports, vents, and drains, must be clearly and permanently labeled. The penetration fittings of the package should not be the same size or configuration (e.g., use of 5/8-in. "male" fittings on each penetration), and they should not be spaced symmetrically. The operation of penetrations should be active and positive. Penetration covers, the closure, and other attached items, such as impact limiters, should be match-marked to the body of the package so that the orientation and alignment of the item to the package are clear. Blind fit-up and blind attachment of items, such as the attachment of limiters to a package, should not be designed if easily accessible attachment points with direct verification of fit-up can be used. It may be difficult to achieve blind fit-up of components (e.g., align and start bolts in blind holes), or verify that all of the bolts in deeply recessed holes are, in fact, installed or installed correctly in the field. Inability to achieve blind fit-up can result in the application of excessive force to the item or the use of those bolts that can be started in an attempt to draw the item into place. Both methods are likely to result in damage to the component.

Similarly, controls and gages on ancillary equipment or fixtures should be unambiguous, and they should operate or indicate correctly. For example, the sequence for torquing lid-closure bolts should be stamped into the closure surface so that the torque pattern is evident. Manually operated valves should be designed so that the port cover of the valve housing cannot be installed if the valve is in any position except fully closed and so that, once installed, the housing prevents the valve from opening. The drain design for packages with top drains should be very carefully reviewed to ensure that the system is not susceptible to blind alignment or mating problems that could lead to "blow-by" at seals. (Blow-by describes the unplanned expulsion of gas that might blow by an improperly seated gasket or seal in the drain system during the act of using air pressure to force water out of the package.) The gas bubbles resulting from blow-by could give the operator a false indication that the water in the package has been removed. Solidly piped configurations reduce the chances for drain-line air leaks that could result in water being left in the cask cavity.

Even though other examples could be given, the point is that the potential for operator, procedural, or function error must be considered for each feature and combination of features that the designer incorporates into the package design or its ancillary equipment. This goal is achieved by considering how, or if, any feature can reasonably be mishandled, misidentified, misinterpreted, or incorrectly used.

A second potential source of error arises from the environment in which the package will be used. In most facilities, the work space is limited and radiological controls are placed on activities. The space limitation forces the designer to consider limiting the amount of ancillary equipment used with the package. This constraint may also mean that the operator has limited access to areas of the package. Radiological controls imply that the operators will likely wear protective clothing that limits manual dexterity. This clothing makes it difficult for the operator to grasp, handle or manipulate small controls or small items such as screws, washers, O rings, and retainer clips, springs, tools, etc. The designer should avoid the use of small items which the operator could drop

or find difficult to manipulate because he or she must wear multiple layers of gloves or because the use of a respirator faceplate may cause a loss of visual acuity.

The designer should also consider the biomechanics of the body. The operator should not be required to push, pull, or lift items, such as connector pins between the lift gear and the facility crane hook; these efforts may require more than average strength, dexterity, or reach.

In reducing the chance of operator error, operator training is perhaps the next most important element after package and ancillary equipment design. During training, the operator must ensure that users understand both how the equipment operates and why it operates in a particular way. From knowing the why's, the user can anticipate the next steps and understand the consequence of incorrect actions. The training of users should be conducted by the package owner and should incorporate any lessons learned by the package owner in the historic use of the package. The training should also ensure that the operation and use of ancillary equipment required with the package are covered in the same detail as is the package operation. User training is discussed further in Sect. 12.5.

12.2.3 Package Support Items

The package system must be passive in meeting its performance objectives and should be straightforward in its functional operation. The design should rely on the use of common tools and instruments. The use of unique or special tools, equipment, systems, features, and fixtures typically increases operational complexity, handling requirements and resource commitment of the user, and reduces package reliability.

The use of unique or special items requires increased resources by both the owner and user for packing and unpacking, decontamination, and setup and operation of the equipment. In addition, the package owner may have to provide operating and maintenance instructions, spare parts, or other services to ensure that failure of the special item(s) does not significantly interrupt package-handling or shipping operations.

For the best reliability, the package operation should rely only on the items and services already maintained in the user's facility and on the common tools that one might find in the typical mechanic's tool box. Ideally, a package would use only two sizes of common fasteners: (1) large (e.g., 1-in. hexagonal head bolts) for the closure, impact limiters, and tiedowns and (2) small (e.g., 0.5-in. hexagonal head bolts) for penetration port covers. The use of tools, line and hose, and other items already at the user facility assists in minimizing the generation of radioactive waste. These items do not need to be removed from the radiation control area and generally have other uses within the facility.

Instances may arise when special or unique items or systems are useful. Nevertheless, the implementation should be carefully considered. Two questions can be asked: (1) Does the tool, fixture, or process reduce operator dose? (2) Does it reduce handling time? Usually there is no other objective basis for considering the tool, fixture, or process. The estimated reduction in operator dose or handling time must include the dose associated with, for example, setup time, packaging and handling in the radiation environment, the decontamination necessary to remove the equipment from the contaminated area, and the consequence of failure of the item.

It is also possible for special items, or even external systems, to introduce unexpected problems. For example, a package used to ship high-level radioactive waste was received at a burial site several times containing significant residual water. The package owner determined that it would be helpful if a control panel were developed which would indicate the amount of water that was removed from the package during draining. In addition to the connections for monitoring the water removal, the control panel also contained gages, piping, and valves for controlling and monitoring air and inert gas flow, vacuum drying, and system pressures. Use of the panel did not require a revision to the package Certificate of Compliance. However, in practice, the panel proved difficult to use and did not provide the information needed.

Ultimately, the panel was discarded, and another solution to the problem was found; the size of the seal used in the (top) drain system was increased. This larger seal essentially eliminated leaks in the drain system that had previously prevented the water from being removed. However, this change required an amendment to the SARP.

While the panel was in use, three conditions arose which the package designer should find instructive. The first was that one user refused to allow the panel to be removed from the facility immediately after the shipments were made because it could not be demonstrated to the satisfaction of the user that the panel did not contain free water. A second user refused to allow a panel to be shipped from the site because the user could not estimate the isotopic and curie content of residual contamination in the piping, valves, and fittings of the panel. (Ultimately the internal piping was chemically decontaminated until flush-water samples were free of trace contamination.) The panel was finally discarded when another user found that the panel had a surface radiation reading of 0.7 mSv/h (70 mrem/h) as a result of residual contamination in the panel plumbing.

In considering the satisfactory performance of a package system, the designer should take the time to think through the use and the consequences of use of package-handling aids or proposed special equipment. Although items can be designed, built, and made to work, this does not mean that they should necessarily be designed or provided. In general, simplicity in the design should always be selected over "complexity" when simplicity is consistent with the passive functionality of the package.

12.3 CONDITIONS DURING USE

The package designer must always consider the loading and unloading conditions that the package will encounter in normal use. Typically, the package is intended for the transport of a specific type of material that is to be moved to or from existing facilities. At those facilities, the package will be handled either underwater (wet) or dry, depending on the nature of the contents. The package designer should consider the package-handling requirements for both environments when designing the package.

The principal design issues for underwater (wet) handling include removal of the water from the package cavity, accumulation of internal contamination, and alignment of the lift fixture and closure while the package is underwater. Dry handling may be done either directly ("hands-on") or remotely, depending primarily on the radiation level of the contents. Typically dry handling results in significantly less accumulation of residual contamination in the cask cavity over an extended use. Remote, or hot-cell, handling requires closure alignment aids similar to those needed

for underwater handling because the operator is some distance away from the package. The decontamination of the package is a concern in either environment.

Closure and leak testing are often long-duration tasks that are performed in a radiation environment. The designer should consider how these tasks are to be undertaken and ways by which the duration of the work can be shortened or facilitated. Specifically, the level of torque required for closure bolts, the number of bolts, and the number of leak tests to be done should be considered carefully. To the extent that the number of bolts or number of tests can be reduced, the package-handling time and, subsequently, the dose received by those associated with handling the package are also reduced. The designer should also consider how bolts will be positioned when the package is in the work station and on the transporter. Bolts should not be difficult to reach or require the operator to be in a high radiation field for installation or removal.

12.3.1 Wet Handling

Type B packages are frequently handled underwater. (Smaller quantities of radioactive materials generally do not require the shielding that is provided by the pool water.) The designer should consider several important consequences of underwater use of the package.

One consequence is that packages used underwater must have a provision for the draining of water from the cavity once the package is removed from the pool. The drain may be either in the bottom of the package or at the top. For top drain designs, air, or another gas (usually inert), is introduced into a vent in the top of the package; the air or gas then pushes the water up a stand pipe that extends from the bottom of the cavity to a drain port at the top of the package. Residual water that can not be forced up the drain tube must be removed by a vacuum-drying system, unless the effects of the residual water have been considered in the package analysis. The larger the quantity of residual water, the longer the vacuum-drying time. The amount of residual contamination that remains in the package cavity once the water is removed is related to the efficiency of the drain system.

The industry has had experience with both bottom and top drain systems (see Fig. 12.1). Neither system appears to offer a compelling advantage over the other. Bottom drains have proved to be somewhat more efficient in water removal, leading to a reduction in vacuum-drying time because of the smaller amount of residual water. A disadvantage is that the bottom drain is located away from the closure where most of the handling work is performed; consequently, personnel access is required to both the top and bottom of the package. Usually, two bottom drains are provided as added protection against the blockage of a single drain line that might be caused by the configuration of the contents, by shift or displacement of loose contents, or by debris; however, two drains are not required.

The principal advantages of the top drain are that access is required only to the top end of the package and that no chance exists for residual water to discharge accidentally out of the cavity and thus possibly contaminate personnel or the facility. Its principal disadvantage is that water is removed from the cavity only as long as there is a solid column of water in the stand pipe. Once air is in the stand pipe, the water in the pipe and in the external drain line above the package will drain back into the cavity. The water must then be vacuumed out (if liquid is not permitted or is not considered in the package analysis). If the stand pipe mates to the closure of the package, a gasket is required between the top of the pipe and the underside of the closure. During installation of the closure, a blind mate-up of the closure penetration and the vertical stand pipe is made at

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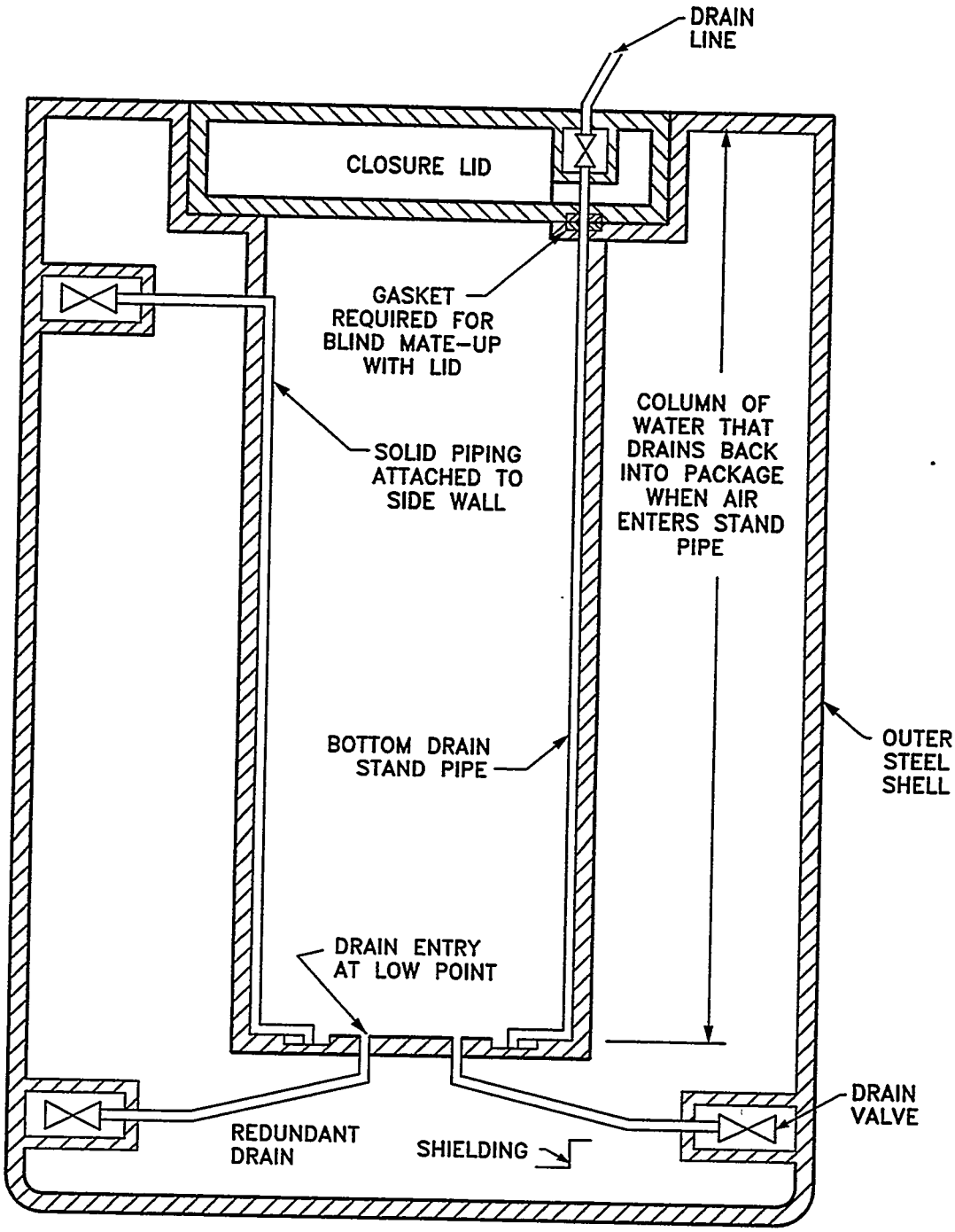

Fig. 12.1. Schematic of the possible drain systems of a vertically oriented package.

the underside of the closure. Historically, packages designed this way have proven susceptible to leaks at the gasket because of poor sealing or dislodgement of the seal during closure installation. If leaks occur at the gasket, the cavity water cannot be removed because the pressurized air or gas escapes to the drain through the gasket leak. This type of leak can be avoided by using solid piping that extends through the side wall near the top of the package. Even though this use of piping can be a design challenge, its cost will be more than offset by the time saved by personnel during the life of the package.

A second consequence of underwater handling is the accumulation of residual contamination in the package cavity. Particulate contamination that accumulates at the bottom of the package can be a significant problem for the operator when removing components (e.g., "baskets" or spacers) from the cavity and in performing maintenance tests, parts replacements, and inspections of those components. Residual contamination can also introduce unexpected isotopes into a user facility. Frequently, it results in a need to keep the cavity full of water during handling of the package with the closure lid removed in order to preclude airborne contamination.

The accumulation of activity in the cavity results from particles and dissolved substances (e.g., cesium) in the pool water that settle, or plate out, during underwater handling and from the assortment of particles (sometimes called "CRUD") that flakes off the contents and falls into the cavity during transport. Depending upon its constituents, the radiation from accumulated material in a package, such as from a spent-fuel shipping cask, can be 5 to 6 Sv/h (500 to 600 rem/h).[3] The residual material may also harbor "fleas," very small particles of ^{60}Co which are intensely radioactive. Individual particles found during cavity decontamination of a truck-mounted spent fuel cask emitted radiation at a rate as high as 0.7 Sv/h (70 rem/h). Dealing with residual contamination in the cavity during any handling operation can be a significant threat to the package operator.[9]

To decrease the amount of residual material and to improve the efficiency of water removal, the designer should carefully consider the flow mechanics of the drain and vent system and the configuration of the package at the drain location. The package cavity, and any necessary internal components such as spacers, should not be designed with areas or surfaces that trap water or contamination or which significantly impede the flow of water to the drain. When water flow is impeded, entrained particulates settle and remain in the cavity. The designer should ensure that the entry to the drain is at the lowest point in the cavity and that the bottom of the cavity contains channels or is slopped to direct water to the drain line or tube. Good water flow helps to ensure the removal of the maximum amount of suspended particulates, which in turn reduces the level of residual contamination in the package and, ultimately, reduces operator dose.

Drain lines or tubes should be sized so that the water can be removed from the package within a reasonable time. A second consideration is the potential use of the fill-and-drain system to circulate decontamination fluids through the package cavity. Internal decontamination is typically required many times during the life of a package, and it is facilitated if the vent and drain are sized to permit good flow of a decontamination liquid. Historically, drain and vent lines have ranged in size from 9.5 to 32 mm (3/8 to 1 1/4 in.), depending largely on the size of the package cavity. Based on experience, lines sized larger than 16 mm (5/8 in.) have proved to be better than the smaller ones from an operational point of view. Smaller lines can become clogged with miscellaneous debris. However, for packages with top drains, lines larger than 1 in. tend to allow a significant amount of residual water to remain in the package.

A third consequence of underwater handling is that the alignment of lift fixtures and closures can be difficult when the package is at the bottom of the loading pool. The principal reason is that the operator is often directly above the package during underwater operations. This arrangement results in the operator's having poor visual perspective of the makeup between the lift fixture and the cask, as well as an impeded view of the cask itself, because of interspersed hardware (e.g., the crane block and tackle; lifting yoke extension pieces; other lines, cable, or hose); water surface conditions; and limitations of underwater lighting.

To properly install a lift fixture or a package closure, the operator needs alignment guides (or match marks), and alignment keys, pins, or other guidance hardware to ensure that equipment is properly engaged. To the greatest extent possible, the match marks and keys should be a permanent part of the package, its closure, or the lift gear. Removable items increase handling time (and therefore increase operator dose), increase the amount of radioactive waste (because they must be wiped down on removal), require a storage location adjacent to the work area, and must be separately handled for packaging and shipping for removal from the facility. It is important that operating personnel (the crane operator in particular) never rely solely on the use of alignment devices to achieve an engagement of the lift fixture to the package. Full engagement should always be visually verified (as a handling procedure step) regardless of the convergence of alignment marks or other indicators.

12.3.2 Dry Handling

For packages that are used in a dry-handling environment, the designer may have fewer concerns about alignment of equipment and residual contamination in the package than for packages that are handled in a wet (underwater) environment. One reason for this is that the contents of most packages that are handled dry are typically much less radioactive than the contents of packages that are handled wet. The lower level of activity normally allows the operator to get much closer to the package, thus facilitating the connection of lifting fixtures and alignment of the closure lid. The exception is the manipulation of a Type-B-quantity package in a hot cell, wherein the handling is done remotely. In this situation, the operations and the problems are much the same as those encountered during underwater handling.

The amount of residual contamination in a dry-handled package is often much less than that for a "wet" package because contaminated water and water-contaminated contents are not introduced into the package. Nevertheless, the package design should allow for decontamination of the package by a vent and drain system. If the designer is certain that the package will not be used in a wet environment, it may be more expedient to design a "maintenance closure" (which incorporates a stand pipe drain and vent) for the package than to incorporate a drain system. If no package drain system exists, liquid in the package must be removed by using a sump pump, wet/dry vacuum (such as a shop vacuum), blotters, or other absorbers. Although operationally viable, these methods can lead to unacceptable levels of airborne contamination and unnecessary doses to package operators.

12.3.3 Automated Operations and Handling

The principal advantage of automated handling of a package is preventing radiation exposure to the operators. Although not necessarily appropriate at all locations, automation of package

operations may be considered when the package is routinely handled in a given facility. An example might be a 200-L (55-gal) drum-handling at a waste facility.

In designing a package that will be handled remotely, robotic systems that are employed must have optical targets to locate package components (such as bolts, fixtures, ports, lift ears or trunnions, etc.), and the robotic grapple device (the "end effector") must be able to attach to and hold the package's removable items, such as bolts, that the end effector engages.

Optical targets must allow the robotic device to align itself to package features that it will operate. Once properly aligned, the device uses a map held in memory to locate and operate the features. Theoretically, the robot is able to perform all the actions that the package user would perform as long as the package is designed to accommodate robotic "limitations." One example is that closure bolts should have a conical dome to facilitate the centering of the socket onto the closure bolt. Another is that the bolt head must have a slot or groove that permits the end effector to hold the bolt for retraction and for moving it to a temporary storage slot so that the next bolt can be removed. Alternatively, bolts can be designed to be retained by a removable piece (e.g., the package closure or a port cover). These bolts could be spring-loaded so that they move away from, and disengage, the threads as they are unscrewed; they can be removed once all bolts are completely unscrewed, and the removable piece is then released from the package. Port covers and closures must have a similar feature so that they can be grappled and moved. Attachment of lines to package fittings to support their removal can be somewhat more complicated; therefore, careful attention to the remote coupling aspects of the fitting is needed. Other nuances of interfacing the package with an automated system can be found in a Sandia National Laboratories report, *Cask System Design Guidance for Robotic Handling* (SAND89-2444).[5]

Effective use of an automated system requires that the package, facility, and robotics designs be coordinated to ensure that the total system works in an integrated way.

Even if a completely automated system is not used, there may be benefits derived from automating specific aspects of the package-handling process. Individual activities could include performance of radiation and contamination surveys,[6] loading or unloading of contents,[7] or transfer of the package between work stations within the facility.[8]

12.3.4 Package Decontamination

The package designer should consider the required external decontamination of the package in two ways: the package surface design or configuration and the proposed decontamination practice. Usually, two categories of surface contamination must be removed. The first type includes particulates and dissolved substances that adhere to the package surface primarily because of microscopic roughness in the surface material of the package or because of oily films or other residue on the surface. The second category includes volatiles that become (loosely) ionically bonded to or adsorbed in the oxide layer of the package's surface material. This bonding is principally exhibited by ^{134}Ce and ^{137}Ce. Cesium is a fission product usually found in the spent-fuel storage pools of nuclear reactors. Removal of ionically bonded and adsorbed material can sometimes require intense effort.

To facilitate package-surface decontamination, the designer should ensure that the external surface of the package is smooth (e.g., free of nooks and crannies, holes, bolt holes, and other locations inaccessible to decontamination personnel and/or material). From the perspective of surface

decontamination, the smoother the surface, the better. Electropolishing and painting result in smoother surfaces than can be obtained from mechanical polishing, which is normally applied to the finished surfaces of the package during its fabrication. Experience with electropolished packages is limited, but many casks have been electropolished, including the NAC-LWT legal weight truck casks and at least two of the IF-300 rail casks. No published U.S. data shows that electropolishing these casks has resulted in a significant reduction in residual surface contamination compared to mechanical polishing. However, the owners of those casks (Nuclear Assurance Corporation and VECTRA Corporation, respectively), are of the opinion that electropolishing has significantly reduced the residual surface contamination on these casks. A few packages, primarily some used in dry-handling conditions that are carbon steel, have been painted with epoxy-based paints. However, for packages that are designed to be handled underwater, there is currently little evidence that painted surfaces are easier to decontaminate than are unpainted, polished surfaces. A different method that can be used to improve the ability to decontaminate a package is to protect it from contacting contaminated pool water by placing the package within a plastic bag or a metal skirt, or applying a strippable paint directly to the surface before it is immersed. However, U.S. experience with the use of strippable paints to temporarily protect the cask surface from contaminated pool water has been unsatisfactory.

When considering bagging or skirting the package, the designer should be convinced that the decontamination effort saved by protecting the surface from pool water or other contaminated environment is greater than the effort required to install, remove, and otherwise handle the protective skirt. For example, several spent-fuel shipping casks have been designed with a skirt that protects their several-thousand heat-rejection fins. It would be very difficult to decontaminate these fins if they came in direct contact with contaminated pool water. Consequently, installing and removing the skirt used in these package designs are considered a necessary part of the loading and unloading operations.[3]

Once the cask is removed from the loading/unloading pool (or from any radiation-controlled area), the decontamination method applied at most facilities consists of manually wiping the surface with a rather benign decontamination solution; other facilities decontaminate with steam or pressurized water. Aggressive solutions, containing phosphorus, chlorides, and other similar chemicals, are not welcome in most liquid radioactive-waste-processing systems. Many facilities that routinely ship packages containing radioactive materials already have developed a preferred decontamination methodology. That practice usually involves the use of any of a large number of commercially available decontamination agents, followed by the use of water and manual wiping; this preferred methodology should be used unless the decontamination agent is known to attack the package surface, or the method appears to be ineffective in use.[10,11] In the package operating instructions or in an operations manual, the designer should suggest at least one commercially available decontamination product that is safe to use on the exposed package surface and which is believed to be effective in routine decontamination.

Package-surface contamination (and radiation levels) are limited by federal regulations,[12] but often limits are further reduced by specific user-site requirements, in particular at U.S. Department of Energy (DOE) facilities.

External surface contamination that exceeds regulatory limits in transport has been a problem area for packages, especially those that are exposed to dissolved cesium during loading or unloading in spent-fuel storage pools. Historically, about 5% of the total number of pool-loaded Type B shipments that have been transported through the public sector have been reported as having smearable surface contamination in excess of the regulatory limits upon receipt of the package at

its destination.[13] Many of these incidences are attributed to a phenomenon (sometimes called "weeping") in which smearable surface contamination appears to increase while the package is in transit. In support of the Office of Civilian Waste Management, Sandia National Laboratories (SNL) has undertaken a number of studies to address the causes and to identify solutions to weeping. To date, most work has centered on identifying primary constituents of the contamination and on the mechanisms by which they become attached to the surface and later become mobile. (The principal constituents of surface contamination for pool-loaded-unloaded packages are cobalt, cesium, manganese, and iron. Of these, cesium appears to be a principal cause of the surface activity.[13]) One objective of the SNL work is to identify specific methods that can be applied to remove the surface contamination and thereby reduce (if not eliminate) the number of cases of reportable surface-contamination incidents.

Even though weeping may account for a number of cases of reportable surface contamination, there are also opportunities for error (1) in the decontamination activity, (2) in the subsequent transfer of the package to its transporter, and (3) in the exit and receipt surveys. Both the package owner and the user need to address each instance of high surface-smearable activity recognizing that

- spots on the package surface can be missed in an individual decontamination process,
- contamination can be transferred to the package surface during handling,
- the efficiency of the receipt smear can be higher than the exit smear, and
- counting equipment may be contaminated or improperly calibrated.[14]

These elements, as well as package surface conditions, should be reviewed in each case because it is important to the success of the shipping program that the correct cause(s) of reportable conditions be identified and corrected. Lack of definitive action by the package owner and user can lead to intervention by regulators and to likely adverse consequences for both parties. The package user should always respond to user inquiries regarding surface contamination or to incidences of reportable contamination and should provide immediate technical assistance or recommendations as needed.

Weeping is evidence of substantial inventory of residual contamination on the package surface, most of which is (loosely) ionically bonded to the oxide film that protects the package's steel surface. Bonding to the film "fixes" the contamination immediately following decontamination using routine methods, but allows the contamination to become removable under the influence of temperature and moisture differences that arise in the time between the exit and receipt surveys.

For extended shipping programs, the package owner may wish to encourage the radiation health and safety personnel from those sites to meet to evaluate their inspection methods and the efficiency of their monitoring equipment before the startup of the program. This discussion can help reduce the potential for reportable events by reminding the users to adjust decontamination requirements as needed to comply with the survey methods or procedures used at the respective sites.

As a general rule, identification of locally high-activity readings (hot spots) imply errors in the decontamination or handling process, whereas high-activity readings found more uniformly distributed over the surface of the package imply a weeping phenomenon.

One method of monitoring the surface smearable inventory, which is often applied to spent fuel shipping casks, is to take a beta-gamma radiation reading of the (unloaded) package surface,

followed immediately by a gamma radiation reading in the same location. The difference between the two readings is an indicator of the gross-surface smearable inventory. If the reading is 0.15 mSv/h (15 mrem/h) or more, then aggressive decontamination of the package surface, using an oxide film stripper followed by a surface pacifier, is indicated. An aggressive decontamination method, often using citric or oxalic acids, should be identified by the package owner in the general operations requirements and recommendations contained in the package supplemental procedures or operations manual.

Although decontamination using chemicals is most frequently used to remove gross surface contamination, electropolishing has also been used. The electropolishing process uses an electric current that passes between the package surface and a target device (sometimes called a "wand") to transfer surface material and base metal from the package to the wand. Therefore, the process has the effect of smoothing the package surface as surface material is deposited on the wand. Electropolishing is generally used only if chemical decontamination has proved to be ineffective which occasionally happens because the package design does not allow the chemicals to be effectively applied to surfaces or because of interference from package fixtures. Because electropolishing does remove (very) small amounts of base metal, the designer should ensure that the package dimensions will remain within the tolerances specified in the license drawings after the process is applied.

12.4 OPERATIONS PROCEDURES

The procedures for handling a package are developed by the designer, and reflect the fact that the package is intended to be handled or operated in a specific way. These procedures are presented in the SARP, reviewed and approved by a regulator, and are then usually incorporated into the package Certificate of Compliance by reference to the SARP. According to Regulatory Guide 7.9, three procedures must be described: (1) loading, (2) unloading, and (3) preparing the unloaded packaging for shipment. These procedures comprise the principal content of Chapter 7 of the SARP.

These procedures are referred to as "generic" procedures because they do not reflect the use of the packaging at a specific facility or with specific contents. They specify the steps that must be taken sequentially in order to use the packaging properly. The procedures usually include some steps taken from the general regulatory requirements for all packaging, such as those specifying the applicable "Routine Determinations," to assist in ensuring that these requirements are met (see 10 CFR Part 71.87).

The procedures prepared for inclusion in Chapter 7 of the SARP need not contain an extensive level of detail relative to individual steps, but they must describe the essential handling tasks and permit the regulator to understand the sequence of events. They should also present important data, such as bolt torque and cavity charging pressure (if appropriate), and acceptance criteria where they may apply (e.g., for a leak test).

In addition, these procedures must specify any constraints placed on package operations by the assumptions of the analysis provided in the SARP. Even though not typically applicable to packages regulated by the NRC, constraints may be important for packages used within the DOE complex that are moved under site administrative controls. Administrative controls may be applied as compensatory measures or to preclude certain events for which the package has not been fully

analyzed. [Note that packages used for on-site transport where administrative controls can be applied are evaluated in a Site Safety Assessment (SSA), rather than a SARP.]

Examples of administrative controls could include limits on: transporter speed; the quantity of fuel in the transporter (with accompaniment by fire fighting equipment); package lift height; package, contents or ambient temperature; route controls (to avoid bridges and overpasses and to exclude other traffic); solution molarity or concentration; contents water content; or time in transit.

Even though these constraints may appear in relevant chapters of the SARP, the user cannot reasonably be expected to identify constraints assumed in the various chapters and bring those to the operations procedure used to handle the package. It is the designer's responsibility to ensure that constraints on use of the package are identified in the operating procedure(s). This is similar to the designer's responsibility to ensure that temperatures, pressures and stresses calculated or assumed for the package or contents are consistently applied in the analysis of the package.

Prior to placing the package in service, the designer should have a set of procedures that contain enough detail to allow the user to handle the package without continuous guidance by the operator. These procedures may contain many steps which do not relate to regulatory performance requirements.

For example, a SARP handling step might be:

X.X Remove closure lid.

The equivalent detailed procedure steps might be:

X.X Inspect tamper indicating seal for evidence of tamper.

X.X Remove lockwire and tamper indicating seal. Give seal to responsible person. (Caution—seal may be surface contaminated.)

X.X Remove 8 closure bolts. [19-mm (3/4-in.) socket required. Lightly coat the threads of each bolt with Neolube™, or similar facility-approved lubricant after removal.]

X.X Attach a double-wire rope or nylon sling to each of the two closure lift attachment points. [Double-sling 9.5 mm × 2 m rated at 680 kg (3/8 in. × 6 ft, rated at 1500 lb); estimated weight of closure is 170 kg (375 lb).]

X.X Attach opposite end of wire rope or sling to auxiliary hook.

X.X Lift closure and move to the previously designated temporary set-down area.

An advantage of this approach is that it allows the operator to make changes to the detail of implementing the procedure without having to obtain an amendment to the certificate to accommodate the change. This permits some flexibility in interfacing the package to the user facility, and it allows accommodations to be made to the facility physical plant, or available equipment, and provides some allowance in the use of lifting cables or shackles, lubricants, etc. [Note that the level of detail given in the procedures can be expected to evolve once the package has been fabricated and placed in service.

If the procedures presented in Chapter 7 of the SARP were not broadly written with little detail, then any changes to the operating procedures, found to be necessary through experience, might require an amendment to the certificate and additional interaction with regulators.

In addition to the package procedures, additional procedures may be needed for operation of ancillary equipment, such as the package lift gear, test equipment, or other items. Because these procedures are not part of the package certification requirements, they are not addressed in the SARP.

The package operator usually finds that it is convenient to assemble procedures related to the use of the package and publish them in an operations manual and procedures related to maintenance of the package in a maintenance manual. However, development of these manuals is not required for either satisfactory use or maintenance of the package.

Formal operations manuals are typically developed if there is more than one type of package or more than one user. The manual contains (1) the package operating procedures, which are presented in more detail than is included in the SARP; (2) the ancillary equipment operating procedures; and (3) supplemental information or procedures that may be useful to the user. It is intended to allow the potential user to evaluate the handling of the package in the user's own facility by providing dimensional and clearance requirements, handling-time estimates, resource requirements (both personnel and facility services), and supplementary procedures for the ancillary equipment, in addition to the handling procedures. It identifies the tools that are required, recommends spare parts the user should have on hand, and provides outline drawings of the package and ancillary equipment. Topics of an operations manual might include:

- Lists of standard and special tools required
  - — Descriptions and sketches of special tools
  - — Table of bolt and wrench sizes
  - — Table of torques required for specific bolts
  - — Temporary set-down area requirements (overpacks, impact limiters)
  - — Table of weights and dimensions of removable components
- Services required (air, water, electricity, disposal of liquid and gas effluents)
  - — Craft and technical support [health physics, crane operators, quality assurance (QA)]
  - — Estimates of waste disposal volumes (water, off-gases)
  - — Sling requirements for specific lifts
- Spare-parts recommendations
  - — Documentation and storage
- Package-handling procedures
  - — Loading, unloading, and shipment of unloaded packages
- Other procedures
  - — Replacement of O-rings
  - — Operation of a recommended leak-test device
  - — Use of a recommended vacuum pump
  - — Inspection and operation of lift gear
  - — Technical specifications for recommended lubricants
  - — Decontamination requirements and recommendations

Development of an operations manual is probably unnecessary for the packaging that is specifically designed for very few items (or a single item) and that moves between two specific facilities. For this type of package, the procedures used in the SARP could be made more detailed

and could be directly employed by the user. Such a manual is not required for packages designed to contain less than a Type A quantity; Type A packages often have a single sheet of instructions and operations that are obvious by inspection.

Few facilities handle the package using procedures taken directly from the SARP or even from more detailed documents, such as an operations manual that is provided by the package operator. It is more common to develop a facility-specific procedure. The advantage of using a facility procedure is that facility constraints to handling the package can be incorporated directly into the package-handling flow at the point(s) that the constraint impacts package handling. For example, when placing a spent fuel shipping cask in a commercial reactor pool, the procedural steps might include:

X.X Prior to lifting cask, verify that crane-rail mechanical stops E-44B and W-51K are installed on the east and west crane rails, respectively. Verify that there are no vent or drain lines attached to the cask.

X.X Lift the cask approximately 30 cm (1 ft) and install the cask bottom protective cover. Secure the cover to the cask lower trunnions using two 1-m (3-ft) bungee cords, one on either side of the cask. After the cover is attached, lower the cask approximately 18 cm (7 in.).

X.X Move the cask to the cask-loading area of the pool. The cask shall not be raised more than 15 cm (6 in.) above the floor during travel. Crane operator shall ensure that the cask follows the load path shown in Fig. Y.

X.X Position the cask over the cask-loading pit. Do not engage the mechanical stops. Open Breaker CC-1474T to prevent any north movement of the crane trolley.

X.X Lower the cask to the 148-m (484-ft) level. The crane operator shall use slow speed for the last meter (3 ft) of travel.

The procedure would also provide space for sign-off and sign-off and dating of individual steps, plus space for recording package and/or contents identification information. Use of a facility-generated procedure facilitates the internal review and approval of any procedure that is required by a facility safety committee. Facility procedures must always be reviewed by the package owner before the package is used in order to ensure that the intent and specific requirements of the package procedure are correctly incorporated in the facility document. This review should not be taken lightly because it can prove to be surprisingly difficult to obtain facility permission to deviate from a previously approved procedure during use.

A consequence of writing the steps necessary to handle a package is that the steps are presented in a sequential order. In developing a detailed procedure, the package designer should consider that some steps can be performed concurrently with other steps. If the detailed procedure is written to indicate that concurrent performance of some steps is acceptable, then the handling time of the package is usually reduced (when compared with the time for strictly sequential handling). The time savings is more important for large packages than for smaller ones.

The package designer and the package user may wish to review NUREG/CR-4775, *Guide for Preparing Operating Procedures for Shipping Packages*, for additional suggestions on the possible content of package handling procedures. In addition to this NUREG document, the designer should

be aware that from time to time, the NRC publishes information notices in response to the national experience with radioactive materials packages. These notices, sometimes called "I&E notices," may have specific application to the type of package, contents of the package, procedures, or reporting requirements. Some notices that may be relevant are given below, but this list is intended to provide perspective rather than to be inclusive.

Information Notice 92-62, "Emergency Response Information Requirements for Radioactive Material Shipments," August 24, 1992.

Information Notice 92-58, "Uranium Hexafluoride Cylinders—Deviations in Coupling Welds," August 17, 1992.

Information Notice 90-27, "Clarification of the Recent Revisions to the Regulatory Requirements for Packaging of Uranium Hexafluoride for Transportation," April 30, 1990.

Information Notice 85-46, "Clarification of Several Aspects of Removable Radioactive Surface Contamination Limits for Transport Packages," June 10, 1985.

Information Notice 87-31, "Blocking, Bracing, and Securing of Radioactive Materials Packages in Transportation," July 10, 1987.

12.5 TRAINING

Training of package users should be conducted by the package owner using personnel experienced in the operation of the package. Training provides information to help ensure that the package is handled, loaded, unloaded, and/or prepared for shipment in accordance with its certificate or general license requirements and helps protect the package from damage or misuse. Several strategies can be used to accomplish training (e.g., classroom, video, etc.), but it is recommended that all training include "hands-on" operations with an actual package. To some extent, the type of training used should consider the scope of the shipping program.

12.5.1 Package Handling

The specific information provided during training to users on cask handling and on the operation of ancillary equipment is based on requirements for the specific package, the requirements of the user, and the extent of the proposed shipping program, and information derived from user experience with other packages. Some details of the training are found in the handling procedure developed for the facility. Use of this procedure ensures that the facility constraints and requirements incorporated within the package handling steps are also reviewed and understood.

For single shipments, or for very infrequent shipments, it may not be necessary to train the user in all aspects of package handling. Instead, the user may choose to rely on the package owner to perform the hands-on work or to provide close, direct supervision of those doing the work. The user provides personnel to operate the facility crane, load or unload the package, and provide health physics support and decontamination of the package. The principal advantages of this approach is that people who have a wealth of experience in handling a particular package are

carrying out direct operations, thus avoiding potential problems associated with the use of inexperienced personnel.

For multiple shipment programs, the package operator provides training to facility personnel who subsequently perform the package-handling and -loading (or -unloading) tasks. The training may be either conducted at a "dry run" before startup of the shipping program or training can be given during the first use of the package. In the dry run, the package-handling operations are performed, but the package is not loaded (or unloaded).

Dry runs may be used if there is a perceived complexity in the package-handling operation or if there are a number of interfaces that must be considered between the package and the facility physical plant. A dry run may be recommended (1) to verify the operation of adapters that must be used to mate the package lift gear to the facility crane(s); (2) to examine the use of any special, or unique, equipment items; (3) if there are areas in the package-handling path that have close clearances; or (4) if concerns about physical handling of the package exist.

If a facility review of the package-handling procedures indicates that the handling is not complex, then use of a dry run is probably not warranted, and training should be provided on the first handling of the package. The principal reasons for this are to avoid the personnel exposure associated with the practice handling, the subsequent radioactive waste that is generated during the practice, and the training costs and to ensure that the training is timely. Any significant delay between dry-run training and first use of the package reduces the value of the dry run training. If the facility is planning to ship, the package training should be concurrent with the first package handling for loading. Facility personnel should conduct the initial loading under the supervision, and occasional assistance, of the package owner's personnel. All facility personnel who require training should witness the first use of the package.

Hands-on training, either dry run or first use, should be preceded by classroom sessions in which the package-handling procedure is reviewed and discussed. The procedure should be reviewed in a step-by-step fashion, with the trainees being given (1) the purpose for each step if the purpose is not self-evident and (2) the opportunity to ask questions.

Trainers should always discuss any cautionary notes provided in the procedure to ensure that personnel understand radiation hazards in the working area around the package, pinch points that occur in lifting operations, required sequences of activities, etc. The trainer should ensure that personnel understand the (1) condition of the package as the handling procedure is performed and (2) the expected radiation fields around the package. Finally, careful attention to the procedures and to operational details assists the user in avoiding human errors that could occur in package handling operations.

If the package operator is providing personnel to perform the package-handling operations, a walk-through of the procedure in a classroom environment is still recommended. This walk-through enables facility management to understand the process and allows facility support personnel such as the crane operator, radiological safety technicians, and others who have an interest in the shipment to ask questions, understand the level of support required, and inform package-handling personnel of the interactions that are required with the facility's personnel and physical plant.

User personnel should also be trained in recovery from some of the more common problems that may be experienced with the use of a specific package. These events may include recovery from

a failed leak test, replacement of broken thread inserts, chemical decontamination of external surfaces, or other events based on the package operator's past experience.

All completed training should be documented, and for continuing shipping programs, training records should be updated annually. In most cases, package-handling training records, including the qualifications of the trainer(s), become a part of the facility QA records.

12.5.2 Personnel Qualifications

Most of the package-handing operations are mechanical and often fall within the skill level of persons who are employed to perform work in and around spent-fuel pools, radiological waste-handling areas, and similar areas of the facility. Usually, no specific training is required to perform a package operation unless the operation is unique, sequence is important, or a special technique is known that facilitates (or is required for) its completion.

One example of specific training can be found in the training provided for typical truck or rail-mounted spent fuel shipping casks. These casks are transported in the horizontal position, but they are rotated to a vertical position for removal from the vehicle. During training, the crane operator is cautioned to ensure that the cask is vertical and straight before lifting it out of its rotation trunnions. It is reasonable to expect the crane operator to ensure that the cask is vertical with respect to the lifting cables before lifting the cask because that method is consistent with the facility-based training that the crane operator has already received. The cask owner normally emphasizes the requirement because if the cask swings as it comes out of its rotation fixture, only a few structures in and around the transporter can safely absorb the large kinetic energy of the swinging cask. If a swinging cask strikes the transporter or rotation fixture hardware, the hardware will most likely be damaged, and may not remain usable.

Some package-related activities may require the use of technicians who have specific, certified (or documented) skills, such as inspection of sealing surfaces or the conduct of leak tests. The skill level of those who conduct these inspections or tests is often dependent on the preferences of the facility management. For example, the management may prefer that inspectors who meet American Society of Mechanical Engineers (ASME) level 2 or 3 requirements perform inspections or tests required during the preparation of the package for shipment.

Finally, the package owner should provide training or guidance in package-handling safety, package-specific radiological hazards and precautions, decontamination of the package and its ancillary equipment, etc., if it is warranted because of the characteristics of the package or by operator experience. For example, a rail-mounted, spent fuel shipping cask often exhibits a high radiation field at the point where the cask lid meets the cask body. This high radiation field diminishes as the closure fasteners are tightened, until it drops to the background level when the closure is fully seated and the fasteners are torqued to the required level. During training and during procedure review, the users are cautioned about this local radiation area and are taught how to avoid the higher radiation fields that exist during the handling operations.

Finally, note that usually the technicians who handle the package may be expected to have had training at the facility in safe work practices, radiological health and safety, rigging and hoisting, facility configuration and procedure compliance, respirator qualification, etc. The principal concern

of the package owner should be to address the specific issues that may arise in the handling of the package.

12.6 RECORDS AND DOCUMENTATION

The package user must have in his possession the following three categories of records and documentation prior to the use of the package or else must generate them during package use:

- the package's Certificate of Compliance and the applicable operating procedures,
- a signed-off operating procedure (each handling), and
- shipping documents and radiation surveys (each receipt and shipment).

However, numerous other records may be accumulated in the course of preparing for the shipping program and completing specific shipments. In addition, other records or documentation based on the contents may be required.

The user is required to maintain additional records that are generated during the use of the package: shipping documents, signed-off test results, receiving and shipping dose-rate measurements, contamination surveys, records of any maintenance performed, and similar information as described in 10 CFR Part 71.91. User records must be retained for at least 3 years following the use of the package.

12.6.1 Before First Use

A user must have only a few documents in order to use a given package: the Certificate of Compliance or other approval record and the documents or drawings listed in the certificate relative to use and maintenance of the package (10 CFR Part 71.12.c.1). In practice, the user may also have other documents that are believed to be relevant to the satisfactory use of the package at the facility. It is sufficient to say that as the planned shipment or shipping campaign becomes imminent, the documentation kept at the user facility can become voluminous.

Before the first use of the package, the user must register as a "user of the package" with the regulatory authority. The point of this registration is that the regulatory agency can make the user aware of issues that affect the use of the package up to and including a notice of withdrawal of the package from service. In the course of registering to use the package, the user asserts that he or she is in possession of the required documentation for use of the package. To obtain the latest copy of the package procedures, the user should be placed on the "controlled distribution" list maintained by the package owner for the period of time the package is being used. The user would then receive a copy of the certificate if it is revised by the regulatory agency and a copy of the procedures as they are revised by the package operator. Note that, generally, package certificates and procedures are revised only infrequently.

The management of the user facility should exercise care in naming the recipient of the information from the regulatory agency and the package owner. In general, the information should go to a "position," as opposed to an individual, and the position selected should be one that is responsible for package use or shipment.

The user should conduct an audit of the package owner to ensure that the package has been maintained, tested, and inspected according to its certificate and SARP and that all required records are maintained (e.g., periodic inspection and package repair records, calibration records for operator-supplied calibrated equipment, maintenance of operator personnel qualification and training records, control of spare parts for the package, and similar records relative to the QA program for the package.) The records that the package owner maintains which are subject to audit are of the type described in 10 CFR Part 71.91.c. Some of the audited information, (e.g., copies of records documenting current package periodic maintenance) is usually retained by the package user.

During planning for shipment, the user may receive additional drawings; operating or maintenance manuals; data on interfaces between the package or its ancillary equipment and the facility; and sketches, and perhaps analyses, that are intended for use in demonstrating that the package may be satisfactorily used in the facility, or alternatively, what supplemental equipment may be needed to allow use of the package. Concurrently, both facility and package owner personnel must ensure that the proposed contents are permitted by the package certificate. Conformance of the proposed contents to that allowed by the certificate should be documented by the package user.

12.6.2 Package Use and Shipment Records

Before a package can be shipped off-site, the user must have documentation that shows that the package was handled in accordance with its approved procedures and that certain actions (called "routine determinations" in 10 CFR Part 71.87) have been taken. As a practical matter, the routine determinations applicable to the package are incorporated into the handling procedure so that no separate documentation of these is required.

Individual steps in the procedure are signed as they are completed, and the signed procedure is retained as evidence of correct handling. To satisfy the documentation requirements, the procedure must also be dated and must identify the package by serial number. For convenience, the identification number and expiration date should be recorded in the procedure for each item of calibrated equipment used. This facilitates the location of the original records for these items if a need arises. This record must be retained for 3 years following the shipment.

Shipping records consist of four essential documents—a bill of lading, which describes the commodity; a radiation survey record; a smearable (removable) contamination record; and "instructions to the driver," which provide emergency instructions and a point of contact who is knowledgeable about the shipment. Supplemental information can be, and frequently is, included with the shipping documents. These shipping records demonstrate that the package meets the regulatory requirements for external radiation and for smearable (removable) contamination and provide other information that is package- or contents-specific. Copies of shipment records must also be retained for 3 years after the shipment.

Many other documents may be generated in support of a given shipment. Most of those documents are "contents" driven and are not specific to the package. Some of these documents may accompany the shipment (e.g., if the shipment is "Highway Route Controlled," a copy of the route must be included in the shipping papers). Others may be sent directly to the receiver (such as an NRC form 741), and still others may be sent to third parties (such as that required as part of the advanced notifications in 10 CFR Part 71.97). These records must also be retained for 3 years after the completion of the shipment.

12.7 OTHER OPERATIONAL CONSIDERATIONS

This section addresses some issues related to the transporter and the ancillary equipment that are used to handle, load, and transport the package as a system through the public domain. This system should be carefully integrated to allow its use without reliance on special or unique tools, fixtures, or processes.

12.7.1 Transport Mode

In the United States, packages are transported primarily by truck and generally range in weight from a few kilograms up to about 25,000 kg (55,000 lb). A few that weigh more than 25 t are transported by rail and are used primarily to transport spent fuel or to make large-volume radioactive waste shipments.

Within this range of weights, the vast majority of existing packages weigh less than 2,270 kg (5,000 lb). More than one of these smaller packages may be shipped on a single truck or be mixed with general freight on a single truck provided that the regulatory radiation and surface contamination and the Transport Index limits for the individual package(s) and cumulative Transport Index for the truck are not exceeded. If the package contains fissile material, the package SARP requirements for transport configuration control must also be met.

Packages weighing more than about 4,540 kg (10,000 lb) are usually shipped one at a time on a single transport vehicle as "exclusive use." Exclusive use means that the package is to be transported and unloaded only as designated by the shipper of the package.[15] (It does not mean that the package is the only item permitted aboard the transporter.) It also allows for somewhat higher dose rates and surface smearable readings than are allowed if the package is shipped as general freight primarily because there is shipper control of the package in transport. For exclusive-use shipments, the package dose rate is measured at 2 m (6 ft) from the edge of the transporter rather than from the package surface.[16] The actual weight of the package is not related in any way to the exclusive-use category of shipments. Exclusive use may be applied to the transport of any container that, because of the contents or other conditions, could benefit from the regulatory provision.

Exclusive-use transport conditions are usually assumed in the shielding analysis for Type B packages. If exclusive use is to be the routine category for shipment, a dedicated transporter is usually provided. A dedicated transporter allows the designer to closely control the package tiedown geometry and configuration and typically eliminates the variability that exists in the condition and configuration of a general-purpose transporter.

For a package to be transported by truck without overweight permitting, the total gross vehicle weight (including the loaded package) must be no more than 36,360 kg (80,000 lb), with limits on each axle and axle set. The designer must identify the weight of the tractor, trailer, and package and distribute it appropriately over the available axles. The distribution of the weight must take into account the "bridge formula," which ensures weight distribution of loads according to bridge-construction-design assumptions.[17] Application of the bridge formula to the truck transport of heavy loads often results in a weight limit for the loaded package of about 25,000 kg (55,000 lb).

Packages weighing between 25,000 kg (55,000 lb) and 36,360 kg (80,000 lb) can be transported using "overweight" permits issued by individual states for a specific load. Although these packages have a high capacity, or payload, the advantage of overweight truck shipments can be offset by restrictions that are frequently applied to overweight vehicles. The restrictions vary from state to state. There is somewhat more flexibility in states west of the Mississippi River, but, typically, overweight transport is not allowed at night or on weekends or holidays. A decision to use overweight truck shipments should be based on the distance to be traveled, the states crossed, and the relative merit of a possible reduction in the number of shipments because of the somewhat larger payload compared with a legal-weight truck shipment.

Infrequently, packages weighing more than 36,360 kg (80,000 lb) are moved by truck in a heavy-haul configuration over short distances, using special rigging and multiaxle transporters. The typical application of heavy haul is to move a package between a point of use and the nearest point of rail service. Even though local conditions vary considerably, 64 km (40 miles) seems to be a reasonable upper limit on routine heavy-haul transfer, based largely on the typical slow speed of travel and other restrictions. A more typical distance is 1–3 km (1–2 miles). The shipper must also obtain permits for heavy haul, which normally establish time of day and transit route restrictions. Shipment by heavy haul usually requires the designer to provide special equipment for tiedown, and for the package transfer required between the roadway transporter and the railcar. These package transfers are usually called "intermodal transfers."

Rail transport is typically used for packages weighing from about 36,360 kg (80,000 lb) up to about 136,000 kg (300,000 lb), although any package could be shipped by rail after first being adapted for rail shipment. The cross section of railcar dimensions is restricted to the "Plate B" dimensions given in the Association of American Railroads car register. If the railcar dimensions exceed those described in Plate B, the free interchange of the railcar among railroads may be limited, and the car could be restricted to certain main and major branch lines. This could prevent the use of the package at some facilities because of prohibited rail access. Packages larger than 136,000 kg (300,000 lb) could also be shipped by rail, but their use would be limited by practical lifting, drop-accident, and handling considerations.

Barge transport has been used sparingly in the United States. Large casks used for the long-term storage of spent fuel have been shipped by barge to the Surry Station in Virginia, and an extensive shipping program to move slightly irradiated reactor fuel from the Shoreham Station on Long Island to the Limerick Station in Pennsylvania has been completed. The Shoreham shipments were made using a rail cask mounted on a skid which could be tied down to supports on the barge deck. The barge transport configuration conformed to an American National Standards Institute, Inc. (ANSI) Standard, N14.24, "Domestic Barge Transport for Highway Route Controlled Quantities of Radioactive Material."[18] This fuel was not sufficiently radioactive to be classed as spent fuel even though it was shipped in a certified spent-fuel cask. A barge-to-rail transfer was made at the end of the barge transport segment, and rail service was used to complete the shipment to Limerick.

Currently, spent fuel and high-level waste shipments by barge are not addressed in the regulations that apply to the transport of these materials. It is anticipated that shipment by barge will be

addressed in the future, particularly as such shipment might apply to the removal of spent fuel from reactors for shipment to a repository.

12.7.2 Transporter Vehicles

During the design process, the designer should review the transporter configuration to ensure that all appropriate operational and regulatory requirements are addressed. Operational concerns that should be considered include (1) adequate personnel access to the package, (2) accessibility of package and impact limiter attachment points, (3) proper structural supports for the package, and (4) weld designs that can be easily inspected. Regulatory requirements that should be considered include (1) restrictions for personnel access, (2) verification that the package is secured to the vehicle, (3) periodic inspection of the tiedowns, and (4) verification of the surface contamination and external dose rate of the package.

An important element that is often overlooked is that the transporter may have to be decontaminated either for maintenance or because its surface contamination exceeds either regulatory or user-site limits. To the greatest extent practicable, the need for periodic decontamination should be considered in the transporter design. Transporter contamination can arise from several sources. Most frequently, it is transferred from contaminated coveralls, boots, gloves, or tools to the transporter surface, or it falls onto the transporter from an overhead crane, hook, cables, or lift gear. Occasionally, contamination may migrate from the package surface to the transporter.

To help mitigate any release of contamination from the transporter, the designer should consider placing a drip pan in the area underneath the package to protect the vehicle flooring and closing the ends between the transporter main support beams utilizing a metal barrier. Health physics personnel from the package user's organization should be advised by the package operator of the need to survey the (usually inaccessible areas of the) transporter surfaces once the package is removed from the vehicle.

The designer should also consider that if the package requires the use of a personnel barrier because of either package surface temperature or operating preference, the barrier might be designed to allow access to the package by opening the barrier or sliding it out of the way. If the barrier is to be lifted out of the way, then the attachment points of the barrier lift gear, should be accessible to the user from the transporter surface, or from the ground. The user should not be required to use a bucket truck or other equipment to secure the personnel-barrier lift gear to a facility crane. The barrier should be secured to the transporter with common fasteners (e.g., standard-sized bolts or manually operated latches) rather than fasteners that require special tools.

The package owner or user will have to be on the transporter and will have to handle package tiedowns twice for each complete turnaround (receipt or shipment) of the package. The package tiedowns should be limited in number to the those that are necessary to secure the package to its transporter. Typically, if the package weight is more than about 2,270 kg (5,000 lb) and if the package is mounted on the transporter in the horizontal position, four attachment points are used. These are usually machined to mate with a fixture on the package that is designed for the purpose and which meets the transport-load requirements.[19] At least two of the attachment points are usually the package-lifting or rotation trunnions. Horizontal mounting is normally used for packages that are more than about 2.5 m (8 ft) tall in its operating configuration so as to avoid possible clearance and high center-of-gravity problems. Wire-rope tiedowns are usually used on

packages that are mounted vertically on the transporter. A recent study prepared by ORNL has shown that 8 wire-rope tiedowns are sufficient to secure a vertically mounted package on a transporter traveling at the legal weight limits.[20] Usually, the wire ropes are attached to the package's upper impact limiter. As part of that study, a personal-computer-based code, "The Tiedown Stress Calculation Program," was developed to assist in the specification of wire-rope sizing based on package weight and geometry.

If more tiedowns (or impact-limiter attachments) are used than are structurally necessary, both handling time and operator radiation exposure increase. Package tiedowns should be easily operated using common hand tools. Package tiedowns must be inspected visually each time they are used and must be inspected periodically using a nondestructive examination (NDE) method to ensure continued satisfactory performance. The tiedown design should facilitate these required inspections.

Finally, the designer must consider the height of the center of gravity of the package above the surface of the road or top of the rails. A high center of gravity can result in instability of the transport system during turns. The Association of American Railroads currently specifies that the center of gravity of any rail load be not more than 2.5 m (98 in.) above the rail surface. No corresponding limit exists for truck-mounted packages; however, in at least one instance, the NRC revised the certificate of a truck-mounted spent fuel cask to limit the center of gravity of the cask, trailer, and mounting system to 1.9 m (75 in.) above the road surface.[21] This revision resulted from a tipover incident as the cask trailer was rounding a corner. The designer should attempt to keep the center of gravity of the transport system as low as possible in order to contribute to system stability during transport.

12.7.3 Ancillary Equipment

Ancillary equipment includes any item that is needed to handle the package properly according to procedures or that is provided for the convenience of the user. The quantity and type of ancillary equipment are usually a function of the design of the package including its physical size and the nature of its contents. Many small package designs have no need for any ancillary equipment.

Ancillary equipment items can either be supplied by the package owner or by the package user. The owner should not supply any item that the user can be expected to have on hand. The package owner should advise the using facility early in the planning process of the required equipment that the owner normally supplies and the required equipment that the user is expected to supply. These items must be available, and serviceable, at the start of package handling by the user.

12.7.3.1 Operator-Supplied Equipment

A typical large package has at least two ancillary equipment items. The first is dedicated lifting gear or fixtures designed to lift the package, the package lid, or other heavy equipment. This gear may be a fixture commonly called a "yoke" or a series of wire-rope or other slings that are used with eyebolts or a set of fixed attachment points. The gear must be load tested or inspected according to applicable standards such as ANSI N14.6 or similar requirements. The gear may be shipped with the package, but in its own shipping container because it is usually surface-

contaminated. A heavy fixture, such as a yoke, must be designed so that it can be lifted from its shipping container and rotated to an upright position for attachment to the facility hook. If the fixture requires pins to connect the lifting fixture to the facility hook, there should be trays or other convenience devices to assist the operator in lifting and sliding the pins to make the connection. The lifting gear must be designed to be shipped, stored, and decontaminated between uses.

The second set of ancillary equipment includes the fittings and connectors used to mate the package to the necessary facility services, such as air, water, inert gases, or test equipment. A package is usually designed with quick-disconnect fittings installed at the package penetrations. The facility-half of the fitting is connected to the lines or hoses so that connections can easily be made to the package. (The package typically has the half of the fitting that does not contain the latching mechanism, which is more susceptible to failure.) However, the penetration may also be a simple design such as a threaded fitting closed by a pipe plug.

Other kinds of ancillary equipment can include special tools and fixtures, test instruments, equipment and fixtures, vacuum pumps, mechanical pumps, spacers, lift yoke actuators or operators (if they are powered), lines and hose, seal protection fixtures, pin- or package-handling tools, spacers and stands for closures, gage assemblies or fixtures, and any number of other items that the operator believes the user may need.

As discussed in Sect. 12.2.3, the use of any item of ancillary equipment is discouraged unless it is required or unless it can be shown to enhance reliability or reduce turnaround time or operator radiation dose. The principal disadvantages of ancillary equipment are that it requires separate handling, decontamination, packaging, and shipment, its use increases the volume of radioactive waste (from wipe down and decontamination) generated at any user site and the reliability of the package can depend on the reliability of the ancillary equipment. This means that the package designer should not provide equipment that the user may not need, and should not take unnecessary material or equipment inside of radiologically controlled areas. The designer should always consider the additional cost associated with handling, maintaining and shipping ancillary equipment.

12.7.3.2 User-Supplied Equipment

Ideally, the package does not have ancillary equipment, but to the extent that it does, the designer should specify as many of the items as possible as "user supplied." The advantage of this is that items such as lifting slings, eyebolts, pressure gauges, test instruments, torque wrenches, etc. are used and maintained according to the user-site QA program and have user-site-issued load-test tags or -calibration stickers. This user-supplied list reduces the administrative and documentation requirements on the package user. The user can also choose not to remove items from the radiation-control area. This option reduces radioactive waste because items need not be cleaned to releasable levels; it reduces the rem burden of the package users because the items do not have to be packaged, and finally, it avoids the associated shipping costs.

The owner should avoid supplying hose, line, fixtures, or devices (such as piping or pressure gages) that are likely to be internally contaminated by either water or gases. Significant effort can be expended in trying to remove these items from the user site because it can be very difficult to estimate the residual contamination inside the item (which is needed for shipping papers) and to ensure (guarantee) that the item is dry internally. If hose is used, for example, with an air- or

water-pressure-operated yoke actuator, the owner needs to be able to show to user-site health physics personnel that the hose, valves, fittings, and other hardware do not contain free-standing water and contain little or no residual contamination.

Similar concerns exist for virtually any item that is brought into a radiation-control area. Test instruments can be come internally contaminated by air drawn through the instrument by cooling fans or items can become externally contaminated in inaccessible locations. Certain vacuum pumps can create a mixed waste during use if the discharge stream can mix with the compressor oil; therefore, special care should be taken in the case of instruments and pumps.

The designer should recognize that any item brought into the user site by the package owner may remain there because it can not meet site release criteria. While it is tempting to think in terms of "adding it to the cost of doing business," it is more responsible to design the package so as to avoid the use of special or uncommon tools and equipment and to use the tools and equipment that can be expected to be found at the user site and that are provided by the user.

12.7.4 Emergency Response and Recovery

A package designer must recognize that the package may need to be recovered from an incident or accident. While accident scenarios are widely variable, it is in the interest of the designer to consider ways whereby the package that is designed to some level of accident survivability (e.g., a Type B package) can be recovered after an accident.

Most accident scenarios assume either that the normal lifting gear is not available with the package or that the features of the package that are normally used for lifting are damaged, thereby preventing use of the lifting gear even if it were available.

Given this assumption, two basic recovery strategies can be used by the designer. The first is (1) to consider how existing features can be used with field rigging to move the package from its resting place to a temporary transporter and (2) to identify those features. The second is to design specific recovery features, or overdesign existing features, that could be used to facilitate package recovery with field rigging.

The industry has had very limited experience with package recovery. Based on the accident record[22] and on the fact that the robustness of a typical accident-resistant package lends itself to field rigging for recovery, there seems to be little justification for the inclusion of specifically designed features for recovery on the package. First, the variability of accident scenarios could, at least statistically, render the design feature unusable or inaccessible. Second, the inclusion of the design feature may complicate the analysis and regulatory approval of the package or increase handling or maintenance time because of decontamination, inspection, or other requirements of the feature. Third, extensive evidence exists that package recovery can occur using field rigging without resorting to the use of features designed specifically to aid in the accident recovery.[23]

A useful alternative strategy is to review the package design with the objective of stating how the package would be recovered in the field. Field recovery generally involves securing the package in some way using slings or wire ropes if the lifting yoke cannot be used or is not otherwise available. One possible method is to slip a single cable 'noose' around the package that tightens as the cable is pulled, for example, by a mobile crane positioned near the package. This arrangement allows the package to be rotated, dragged, or lifted as necessary. The designer should

consider where there might be sufficient structure on the package that such a noose could be used without (additional) damage to the contents. The package may also have appurtenances that are suitable for recovery lifting or have existing features such as impact-limiter-attachment bolt holes that may be adequate for the eyebolt attachment of a three legged sling. Other alternatives can likely be identified given that it is unlikely that a damaged package can be returned to service in any case. The method of recovery is constrained only by the need to preserve the remaining integrity of the damaged package and to ensure that the contents are not exposed.

If a strategy for recovery seems clear, reasonable, and operationally feasible, then no additional action may be warranted except to record the strategy against future need. Should the review indicate that the recovery of the package is problematic, then it may be prudent to enhance the design in a way that facilitates any future recovery effort.

The package owner should have an emergency response and recovery plan in place at any time that the package is in service. The plan should

- provide guidelines, instructions, or procedures for evaluating and responding to the field conditions;
- identify the individuals who will respond to the event;
- provide for notifications to meet regulatory and other informational requirements;
- identify the spokesperson(s) for providing definitive information to site responders, regulators, and response agencies, and to respond to media and public information requirements;
- provide for radiation-detection instruments, anticontamination clothing, personnel monitors (film badges or dosimeters), rope, signs, flashlights, some common hand tools (impact limiter, port cover, closure, and tiedown bolts), package drawings and diagrams, plastic bags, binoculars and safety glasses, full-face respirators, etc.; and
- provide for identifying and obtaining recovery-assistance resources such as cranes or other heavy equipment, temporary transporters (which include tractors, trailers, or railcars), shoring and tiedown binders, welding capabilities, materials, etc.

The plan should also address any unique aspects of the package or its contents that may be appropriate to the recovery operation.

The shipping papers for Type B quantities of material must contain a telephone number that can be called for obtaining the emergency information on the package and its contents.[24] The package owner should ensure that planning for response and recovery does not begin only when that telephone rings.

Planning for major accidents has the benefit of providing the framework for response to more likely, and usually less serious, situations—traffic accidents or railcar derailments that do not involve the package in any way. For accidents that involve the vehicle but not the package, the package owner, together with the current user, must be able to provide guidance that can lead on-the-scene personnel to determine whether the package is in a satisfactory and serviceable condition. These personnel must also begin the coordination of resources to move the package to a serviceable transporter, if necessary, and continue the shipment, if possible.

Planning is also needed should operational accidents occur. A package that is involved in an operational accident at a given site might be damaged as a result of the loss of control of the package during lifting or other handling activity. The consequences of operational accidents are somewhat mitigated (1) by the immediate availability of personnel who are familiar with the

package and who have a variety of appropriate skills to deal with the accident, (2) by the radiation controls that can be imposed, and (3) because of the generally limited access the public might have to the given site.

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13. Package Fabrication and Acceptance Testing

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13.1 INTRODUCTION

Each radioactive material shipping package fabrication project is unique in terms of the details of its execution. The project elements are dictated by the type of packaging, packaging construction, basis for procurement, purchaser's and vendor's experience, project and purchasing rules with which the purchaser must comply, project responsibility split between the purchaser and vendor, etc.

This chapter outlines some of the fundamental considerations of packaging fabrication and acceptance testing. It will identify those that are extremely important for a successful project, and it will offer some suggestions based on experiences which have been shown to facilitate project execution. This chapter also provides some fundamental guidance concerning selected manufacturing processes.

13.2 PREPARATION FOR FABRICATION

As part of any packaging design activity for projects where hardware production is anticipated, the designer must provide sufficient information to permit the planning for fabrication. Frequently, the designer is not experienced in fabrication methods; thus, the provided information is generally insufficient for actual construction purposes. At a minimum, the design package should represent a set of requirements to which a fabricator must perform. Of course, if the design is simple and the designer is knowledgeable about fabrication, then, in addition, the design media can become the fabrication media (i.e., drawings and specifications), but this is generally not the case.

13.2.1 Design Drawings

The packaging design drawings must be consistent with the configurations, dimensions, materials, standards, etc. that form the basis of the safety analyses contained in the *Safety Analysis Report for Packaging* (SARP). Unless the packaging is unusually simple, the fabricator will take the design drawings and then create a set of fabrication or shop drawings. The shop drawings are keyed to the way that a particular vendor does business. For example, a design drawing may have a number of components or subassemblies on a single sheet, whereas a fabricator may find that a piece-part drawing system (i.e., one component per drawing) works best in its shop or that a

supplemental drawing, such as a raw-forging, is a necessary part of the construction sequence, but it is not a part of the design drawings.

Shop drawings may differ from design drawings and therefore it must be emphasized to the fabricator that overall packaging approval, as well as subassembly and component acceptance, will be based on the design drawings. It takes an exceptionally good fabrication control system to simultaneously bring the design drawings and shop drawings through the construction process and still ensure configuration consistency.

13.2.2 Fabrication Specifications

The fabrication specification, sometimes called procurement and fabrication specification, is the definitive requirements document for packaging fabrication; it is the written counterpart to the design drawings. As with the design drawings, the fabrication specification contains requirements against which the packaging and its components will be evaluated. Based on this specification, the fabricator will establish processes and procedures to ensure compliance. The fabrication specification contains both programmatic and functional requirements. Examples of each of these two follow.

13.2.2.1 Programmatic

Programmatic requirements that may be contained in a fabrication specification include

- Scope
- Responsibilities
- Definitions
- Description and function
- Applicable codes and standards
- Quality assurance requirements
- Deliverables
- Vendor data requirements

13.2.2.2 Functional

Functional requirements that might be contained in a fabrication specification include

- Material requirements
- Material handling and storage
- Material substitutions
- Inspections and tests
- Purchased component requirements
- Fabrication/installation requirements and limits
- Special processes
- Design drawings

13.2.3 Identification of Special Processes/Materials

The identification of special processes or special materials must be included in the fabrication specification as indicated in Sect. 13.2.2.2. It is important, however, to recognize that many general fabrication facilities may have no experience with certain special processes or materials. Therefore, it is usually incumbent upon the designer or purchaser to define completely any special processes or materials. It is not uncommon for a designer or purchaser to develop a special process or material and write the procedures for implementation. Under these circumstances, a vendor needs only to incorporate the preestablished procedures and requirements into his system.

A word of caution: If the vendor has not been involved in the development of the procedure or process, he may be unwilling to bear any responsibility for failure as long as he has followed instructions. This suggests, then, that any special processes or use of special materials be closely coordinated with the fabricator. Experience has shown that many special processes are developed in laboratories or controlled environments but subsequently that they are very difficult to execute in the actual fabrication setting because there environment and personnel are less controllable.

An additional consideration is propriety. If a packaging design contains uniquely developed features or processes, these may have commercial value to the developer. If the cask designer or process developer is also the owner, then there may be no proprietary issues. However, if the design or if the feature or process is the property of others, one must carefully assess the implications of having a sole-source "intellectual product" as part of the packaging. This can be handled through licensing or royalty arrangements, but one should be aware of the need for special arrangements.

13.2.4 Quality Assurance (QA) Requirements

QA is discussed in detail in Chapter 11. It is important to note that there are two parts to QA: The first part is the underlying QA program that flows through the entire packaging project and encompasses all of the participants. This program, which is based on 10 CFR Part 71, Subpart H, and American National Standards Institute, Inc./America Society of Mechanical Engineers (ANSI/ASME) NQA-1, is the principal subject of Chapter 11.

The second part is the packaging-specific QA requirements. To meet these requirements, the unique project requirements are integrated into the vendor's generic QA program. Chapter 11 discusses the graded approach, which requires a specific determination of safety-level and associated quality requirements, on a component-by-component basis, for the entire packaging. The packaging-specific QA requirements are established by both the design drawings and the fabrication specification. This same documentation transmits the requirements to the fabricator.

Integrating the packaging-specific requirements into the general framework of the approved QA program is a vital part of QA management. The erroneous classification of a component, for example, could lead to the inappropriate application of quality-affecting requirements. If such an error were nonconservative, it could, upon discovery, result in costly rework or even jeopardize the entire packaging project.

13.2.5 Project Management Planning

The management of the project by the purchaser and the fabricator is almost as important to success as a set of quality drawings and fabrication specifications. In many cases the purchaser has an established system for project management, which includes such activities as project reviews, work breakdown structures, progress reports, milestone activities, budget reports, and scheduling.

No attempt will be made herein to recommend the composition of a comprehensive system because there are many ways to manage a project successfully. Section 13.5 details some key elements of any successful project management plan. Experience suggests that a management plan should be kept as simple as is needed to achieve the project objectives. Excessive reporting, reviewing, meeting, etc. shifts attention from hardware to paperwork and can increase project cost without a commensurate improvement in control or communications.

For this section of the handbook, it is sufficient to say that a project management plan must be in place before the start of the activity. Furthermore, the plan and the purchaser's philosophy must be one of flexibility with respect to implementation. That is, as the project unfolds and weaknesses in the management system arise, as they often do, the management plan must be able to accommodate changes to improve control, communications, etc. For example, the fabrication project may not begin with a purchaser's representative at the fabrication facility, but the management plan should permit such if it is later determined to be beneficial to the project.

13.3 VENDOR SELECTION AND QUALIFICATION

Selection of the packaging vendor can be the most important part of the entire packaging project. This decision has a direct effect on cost, schedule, product quality, package usefulness, and package life. This section of the handbook offers some fundamental thoughts on vendor selection from the perspective of quality of construction. Issues of schedule and cost are unique to any given project and will not be discussed.

Although not always required, vendor selection is typically performed based on a Request for Proposal (RFP) or a Request for Quotation (RFQ) as part of a competitive bidding process. The bid invitation should include a request for vendor information relative to manufacturing capability, QA, project management, engineering support, and qualifications. Obviously, cost, schedule, financial stability, etc., are also part of an RFP or RFQ, but these considerations will not be developed in this section. The selection and qualification process should include (1) documented responses to the RFP or RFQ and (2) a site visit for first-hand confirmation of vendor claims.

13.3.1 Assessment of Manufacturing Capability

The selected vendor should have the ability to perform the required work. Unfortunately, this is not always the case. For packagings designed to carry radioactive materials (RAMs) the QA requirements in 10 CFR Part 71.105 and Part 71.115 mandate that prospective fabricators comply with the appropriate parts of the purchaser's QA program and that the purchaser perform a source evaluation to confirm a prospective vendor's ability to meet the requirements of the fabrication specifications.

In some cases, these evaluations have focused more on the management and QA systems and less on production of the actual hardware. All are important to a successful fabrication effort. Any vendor qualification team should include project management, QA, procurement, design engineering, and manufacturing engineering personnel.

From the standpoint of manufacturing capability, there are three primary considerations: physical plant, personnel, and prior experience.

The physical plant must have the appropriate machine tools, welding equipment, lifting capacity, inspection and testing facilities, and material-handling capability. The evaluation must take into account expected sizes, weights, and tolerances. The plant also must be capable of accommodating any special processes or, alternatively, must have access to special processes that will be required. In some instances, it may be necessary for the vendor to expand the physical plant. For instance, the machining of depleted uranium and/or the pouring of lead are not general shop activities and may require a facility addition.

The personnel are important, but given the fluid nature of employment, it is difficult to obtain the commitment of individuals to specific tasks. Rather, one should examine the vendor's hiring and training practices. This is especially true of welding and nondestructive testing (NDT) personnel where national standards exist for their qualification. Experience is probably the best measure of those individuals who will be managing the fabrication effort and those who will be providing quality control (QC) and QA.

A knowledge of the vendor's prior experience including recommendations is valuable information, but for a general fabricator, RAM packagings are not likely to be a frequent product. One must break the packaging down into manufacturing processes such as forging, plate-rolling, precise machining, welding, etc., and then look for examples of prior work on the part of the vendor that employed these activities. Coupled with this, one must be assured that the vendor has maintained a high QA/QC during previous jobs.

A prospective vendor that is an ASME Certificate Holder under Sect. III or VIII should be given some special consideration. ASME qualifies its Certificate Holders and performs periodic audits to ensure compliance with Code requirements. Although being a Code shop is not a guarantee that the work will be performed correctly, it should give the purchaser a greater feeling of confidence.

Note that ASME has no code that applies strictly to nuclear material packagings. However, the U.S. Nuclear Regulatory Commission (NRC) in several NUREG documents (e.g., NUREG/CR-3019 and NUREG/CR-3854) recommends the use of ASME Code criteria for fabrication and welding of radioactive-material-shipping packagings.

13.3.2 QA Program Assessment

Chapter 11 focuses on with the contents of a QA plan. The packaging designer and owner are compelled by regulations to possess an NRC-approved QA program that meets the requirements of 10 CFR Part 71, Subpart H. In the case of subcontracted fabrication, Part 71.109 states this requirement: "To the extent necessary, the licensee shall require contractors or subcontractors to provide a QA program consistent with the applicable parts of this part." That is, the fabricator must have a QA program that is essentially an extension of the NRC-approved designers' or

owners' programs. Although this secondary-level program is not NRC-approved, it, nevertheless, will become part of any NRC audit of the fabrication activity.

The fabricator's QA program assessment can take several routes, depending on the vendor's background. An ASME N-stamp holder will have a satisfactory program in place because NQA-1 mirrors NRC requirements. A vendor that has produced radioactive-material-shipping packagings in recent years is likely to still have a model program available for reinstitution even if it is not currently active. The most difficult scenario is one in which a general fabricator has never been required to operate under a formal QA program.

It will be obvious during the vendor's survey activity, which of the previously discussed categories best describes the prospective fabricator. In the case of the vendor with no prior QA experience, the issues for consideration are (1) the cost of implementing the program and (2) the possibility of problems associated with the lack of QA experience. The cost considerations must be balanced with the total fabrication effort, and only can be answered for a specific application. The experience considerations generally revolve around the vendor's commitment (e.g., hiring someone with experience to run the program) or the purchaser's commitment to provide additional QA oversight to back up that of the vendor.

The importance of QA should not be underestimated. Documented instances exist where packagings have been temporarily or permanently removed from service because of inadequate QA.

13.3.3 Assessment of Vendor Project Management Capability

The vendor's project management is the counterpart of the purchaser's project management. A responsibility network must exist within the vendor's shop to provide the internal control of the fabrication activity and function as the principal interface external to the project. These external interfaces are usually with the purchaser and any subcontractors. RAM packaging fabrication projects generally require significant interaction between purchaser and fabricator and is often dictated by the QA requirements and NRC inspection. In some cases, NRC certification of the design occurs in parallel with fabrication. These two simultaneous activities place a great burden on the project managers. One should not underestimate the value of the abilities of the vendor's project management.

Important considerations in the evaluation of a vendor's project management capabilities are (1) existence of a formalized (i.e., documented) management plan, (2) prior experience with interactive projects, (3) management skills of the assigned individuals, and (4) amount of time a vendor can or will commit to project management. With respect to this last item, the ideal situation would be to have a manager who is assigned solely to the project. However, unless the packaging fabrication is a major project, the manager may have to divide his or her time among various shop activities. Dilution of management time can have serious impacts on schedules, cost, quality, and compliance with other contracted commitments.

Project management plans, experience, personnel, and commitments should be supplied by prospective vendors as part of their bids.

13.3.4 Assessment of Manufacturing Engineering Support

As indicated in Sect. 13.2, design drawings and specifications are generally converted by a fabricator into a format that is more suited to its own style of construction. This conversion is one of the many manufacturing engineering activities needed to complete a project satisfactorily. Other activities include tooling, preparing and monitoring special processes, selecting manufacturing methods and sequences, assessing subcontractors, and resolving fabrication difficulties.

Although manufacturing engineering support information should be requested in the RFP or RFQ, there is no substitute for a site visit by a purchaser's representative who is skilled in this discipline. Only by observing the facilities and by interviewing the personnel can one grasp the quality and extent of the manufacturing engineering capabilities. Typical questions might include

- Is manufacturing engineering a separate function, or is it a "shop floor" activity? (The former is preferred.)
- Are shop drawings produced, or are the design drawings used directly? (Again, the former is generally preferred.)
- Are the manufacturing engineering personnel knowledgeable in areas specific to processes required for packaging fabrication?

13.3.5 Subtier Supplier Qualification Process

Subtier suppliers to the packaging fabricator can be providers of subcontracted services or components, suppliers of raw materials, or suppliers of manufactured products. QA requirements specify that subtier suppliers must be qualified to the same extent as the prime vendor if their products or services are identified as important to safety. If the vendor has an established QA program that selects its subtier suppliers in a manner consistent with the owner's QA program, vendor-supplied evidence of this program should be sufficient. Such evidence should include documentation of subtier supplier qualification activities by the vendor and copies of the subtier suppliers' QA programs. If there is a question, the purchaser should consider an independent assessment of one or more subtier suppliers. Direct subtier supplier qualification would follow the same process as that for the packaging contractor.

13.4 FABRICATION CONTRACT ADMINISTRATION

Every organization has its own way of administering contracts, and for the most part, most of these work well. What is offered in the following are some experience-based suggestions with respect to contracts and their administration.

13.4.1 Contracts and Contract Options

If the packaging design activities and the NRC certification process have been completed and if the fabrication of the packaging does not involve unusual special processes, then a fixed-price fabrication contract is recommended. The reason for the preconditions is to avoid or reduce the number of purchaser-requested changes during the fabrication period. Such changes to a fixed-price contract essentially reopen the cost-schedule negotiations. Because fabrication will be in progress, the purchaser is at a disadvantage during these renegotiations.

If fixed-price contracting is selected, it is essential that the scope of work be defined in exacting detail. Because every project has its deviations and changes, a loosely written scope of work can result in an ongoing process of resolving whether changed items are covered in the fabricator's original bid or are extra-cost items.

In situations where there is an uncertainty over some elements of the fabrication, such as the use of a unique special process, a purely fixed-price approach may be inappropriate. If forced to offer a fixed-price quotation on a project with significant risk, such as the use of a to-be-developed process, fabricators often inflate the price to protect themselves against loss. It could be to the purchaser's advantage to obtain a fixed-price quotation on the more routine activities and handle the unique aspects on some other arrangement, such as a time-and-materials (T&Ms) contract. The T&M arrangement will require that the purchaser exercise greater administrative control and involvement in that particular aspect of the project, but it could be more cost-effective.

T&M contracts usually involve a fee or profit. A fixed fee is preferred by a purchaser over a fee that is a percentage of the total expenditure. The former arrangement removes any incentive to expand the project, although for a straightforward fabrication effort, there is little room for "padding." This is more likely to arise in cases in which fabrication requirements are unique.

Where schedule is very important one might consider some type of incentive-penalty provision in the contract—that is, a monetary reward for completion ahead of schedule and a monetary penalty for failure to meet the schedule. A drawback to this is similar to that for a fixed-price contract. Both purchaser and fabricator will be seeking ways to protect themselves, primarily through documentation of deviations from the contract. More administration time will be required under a contract with incentive-penalty clauses than under one of the more conventional forms.

With respect to payment, the use of progress payments is almost universal. Frequently, there is an initial payment, and often there is a hold-back amount (e.g., often 10%) to be paid upon receipt of the finished product. The progress payments can be equally divided based on the estimated fabrication period. They can be based on cost incurred to date, or they can be based on a percentage of completion of the project. Other payment schedules are possible. Clearly, some relationship should exist between the payment and the progress of the fabrication to provide an incentive to meet or shorten the schedule.

13.4.2 Changes and Issue Resolution

As stated in Sect. 13.4.1, it is almost a certainty that during any fabrication project, some sort of deviation or change will affect the contract. The resolution of these and other contract issues is the main task of the contract administrator. This individual must work closely with the project manager, assuming that these are separate functions, to get the complete history and status of the project and the facts surrounding specific issues. The contract should anticipate the potential for revision and should include language that establishes some framework for accomplishing it.

Most contract-affecting matters can be resolved within the language and spirit of the contract. On rare occasions, however, an issue arises of such significance, usually monetary, that purchaser and seller revert to a legal interpretation of the contract provisions. In the extreme, this could involve lawsuits and other legal action. Although these are infrequent occurrences, they are painful enough to suggest the need for a thorough legal review of the contract before execution. It may be a sad

commentary on contracting, but even though one hopes for a flawless project, he or she contracts for the worst-case scenario.

In the final analysis, one must count on the integrity of the fabricator and the relationship between purchaser and vendor to amicably resolve contested contractual issues. This need to avoid litigation further emphasizes the importance of the vendor-selection process.

13.4.3 Termination and Closeout

All contracts have specific language which addresses the termination of the contract. Typical closeout provisions include final payment upon delivery and acceptance of the completed product.

For RAM transport packaging, there are several items of concern with respect to termination and closeout in addition to the product itself. These concerns are rooted in the quality requirements associated with this product. The contract must provide for the delivery of all quality-affecting documents, as described in the fabrication specification, the QA plan, and the appropriate regulations. It must also provide for a certification by the fabricator that the work was executed in full compliance with the applicable requirements.

For a prematurely terminated fabrication project where the work-in-progress will be continued by others, the previously mentioned quality-affecting documents are equally important.

13.5 FABRICATION PROJECT MANAGEMENT AND SUPPORT

As with contract administration, most organizations have developed methods for successful project management. The following will present some experience-based suggestions on several of the more important considerations in RAM packaging-fabrication project management.

There are two fundamental project management objectives: (1) delivery of the completed product within the project parameters (e.g., cost, schedule) and (2) establishment and execution of a plan that ensures a documented, traceable record of the fabrication process and ensures that the completed product is in full compliance with all applicable requirements.

The level of effort in project management is directly related to the magnitude of the fabrication activity and the complexity of the packaging.

13.5.1 Project Organization and Communications

Project organization and communications define the framework within which the project will function. The purchaser's organization and the fabricator's organization do not have to duplicate one another, but they must have at least one common position, the project manager. These two individuals, or their designees, represent the only "official" interface between purchaser and vendor.

Informational communications may, and will, occur between purchaser and vendor representatives at the working level. However, the chain of command must be unequivocal; only communications

via the project managers is official. One exception to this rule is the following: The QA representative can unilaterally stop the job.

13.5.2 Vendor Reporting Requirements

The fabrication contract will define the vendor's reporting requirements. These requirements should include the frequency of reporting, the contents and format of the reports, the form of the reports (i.e., written and/or oral), and the distribution list. Reports are generally produced on a periodic basis, such as monthly or semimonthly. The intent of the reports is to document the project status. This information is primarily for the information of others rather than the project manager, who should not be relying solely on the periodic reports for information. It is recommended that reports always be written; however, they also may be presented orally if it is deemed important to the project.

Reporting requirements often are keyed to project milestones (e.g., long-lead-time material ordering, beginning of acceptance testing, etc.). Milestones are project-specific, and they play an important role in tracking the progress of work relative to the schedule. Other reporting requirements could include (1) costs vs budgets for T&M contracts or (2) the status of activities related to the actual fabrication such as special process development work or personnel training.

The reports are a mechanism for keeping vendors on track, and they also serve to record the history of a project. The distribution of the reports also can serve an informational purpose for those who have an indirect interest in the project.

A few cautions that should be observed when considering the vendors' reports. First, the reporting frequency should be reasonable, such as monthly. The vendor should not be spending excessive amounts of time generating reports. Second, the reports should be concise. Finally, the report should contain progress and status information and describe expectations for the next reporting period. Report highlights should focus on items that jeopardize the schedule or negatively impact the cost or product quality.

13.5.3 Project Audits

Most of the project-auditing activities will be in the area of QA. However, for a large project, it is not uncommon to audit the project management system for performance. Such an audit may include compliance with contract terms, project communications, cost-accounting methods, recordkeeping, adherence to commitments, and other similar subjects. As in a QA audit, the effort should be conducted formally, and the results should be documented and reviewed with vendor management.

13.5.4 Project Change Approvals

Just as there is a formal process for making hardware-related changes during the fabrication, a process must be in place for changing project-management-related items as a result of circumstances occurring after the start of fabrication. The fabrication contract is the vehicle for defining this change process. Examples of change are revised schedule, revised order quantity, suspension of work pending project manager decisions, changes in the cost, changes in vendor

commitments or purchaser requirements, etc. Contractual or project management problems can be as damaging to the schedule as hardware problems, and the project must have the capability of making mid-course adjustments if such are needed.

13.5.5 Resident Oversight

As was mentioned previously, projects for which the schedule, quality, or cost are critical, it is highly recommended that a resident project management representative be placed in the vendor's shop. This person may have either broad or limited authorization with respect to making on-the-scene decisions, depending on how the purchaser's project manager wishes to control the effort. Generally the resident representative operates within certain restrictions and must check with the purchaser's project manager on major issues. The value of the resident is a result of his or her day-to-day direct interaction with the project. The communications between the resident and the vendor project manager must be formal and systematized. The informal communications between the resident and his project manager form the basis for official purchaser-to-vendor actions.

The vendor also benefits from the resident in that decisions can be made or problems can be resolved sooner. The resident has quicker access to the purchaser's organization and can lessen the usual turnaround time for conventional communications. Experience has shown that for a large project, a resident project representative can be of immense value.

13.6 QA AND QC

QA and QC, although organizationally independent of project management, are nevertheless integral to the successful completion of the project. By virtue of this, there is a necessary interface between project management and QA. This interface manifests itself in several ways.

Initially, project management must convey to the vendor the importance of QA and ensure its implementation. The requirement for a specific vendor QA plan and the inclusion of QA requirements in the fabrication specifications are illustrations of the importance of this discipline in the fabrication project. Next, project management is charged with the responsibility for ensuring that QA-identified matters are implemented. For example, although a QA audit might identify a vendor's failure to follow procedures, it is the project manager's responsibility to correct the deficiency with the vendor. Of course, QA can work the opposite way because QA audits are conducted also to ensure that the purchaser complies with its own QA plan. Again, deficiencies are the responsibility of the project manager to correct. Finally, at the termination of the project, the project manager must ensure that all quality-affecting activities are properly completed and documented and that those QA/QC-related documents are transferred to the purchaser with the hardware.

13.7 MANUFACTURING PROCESS CONSIDERATIONS

This section sets forth general philosophies as well as specific recommendations regarding the fabrication of shipping containers for transporting radioactive materials. The predetermination of all fabrication requirements for the many sizes, types, capacities, and intended uses of shipping casks is not possible. Rather, this section consists of recommendations for the designer and

purchaser to consider for inclusion in the packaging specifications as well as for the vendor to consider during fabrication planning.

As a fundamental consideration, the use of the ASME Boiler and Pressure Vessel Code (B&PVC) is recommended. The NRC, in two of its documents (NUREG/CR-3019 and NUREG/CR-3854 for cask welding and fabrication, respectively), references ASME Code, Sects. III and VIII. Although no ASME Code exists for RAM shipping packages, the Code offers a basis for manufacturing processes that have been validated by national standards processes.*

13.7.1 Metal Forming

Packaging fabrication can involve all types of metal forming. Typical processes include cutting, rolling/braking, casting/forging, and machining.

13.7.1.1 Cutting

Mechanical cutting (e.g., sawing, shearing) is a straightforward process that generally leaves no residual effects on the metal other than a rough surface finish. When oxyacetylene or an arc process is used to cut metal, all slag and previously molten material should be removed by mechanical means before further fabrication or use.

Edges that will be exposed in the finished packaging should be rounded (grinding is permitted) to a radius of at least 3 mm (1/8 in.) or chamfered at 45° to at least a 4 mm (5/32 in.) flat. Stainless steel is to be cut only by machining, abrasive cutting, or arc cutting with inert-gas-shielding tungsten arc, plasma arc, laser, or flux-injection methods. The use of iron powder as a flux for arc cutting should not be used. In instances where cut edges are to be welded, the disturbed material should be removed by machining or grinding to eliminate all irregularities and to provide a clean, bright, smooth surface on the cut edge.

When flux-injection cutting is used, the cut surface should be machined or ground back at least 0.8 mm (1/32 in.) to remove all contamination. Grinding or machining of stainless steel should not be performed with tools that have been previously used on carbon steel. Tools and grinding wheels should be freshly sharpened and dressed. Wire brushing of stainless steel should be performed only with stainless steel brushes that have not previously been used on materials other than stainless steel. Clamps, wedges, clips, etc., welded or mechanically fastened to stainless steel should be composed of stainless steel.

13.7.1.2 Rolling and Braking

Rolling and braking processes are generally accepted forming methods which are frequently used to produce cask shells. The rolls or dies used in the process should be free of contamination from previous metals when forming stainless steel components. Because plastic deformation occurs during cold-forming processes, it may be necessary to perform postforming heat treatment to

*Note: At the time of this writing, the ASME has formed SG-NUPACK (SC III) to write construction rules for the containment systems for high-level waste and spent fuel transport packaging. These rules do not address the entire cask, but they represent a start toward a comprehensive cask construction code.

remove or reduce the residual stresses in the component. The bending and rolling processes can reduce the metal thickness; thus, the feed material should be sized to accommodate this occurrence. When rolling cylinders for seam welding, the weld prep should be premachined. Because the rolling process is not considered a close-tolerance operation, the rolled and welded cylinder may require machining depending on the cask design; a machining allowance should be included in the selection of the thickness of the material chosen.

13.7.1.3 Casting and Forging

Packaging designers generally use castings and forging for closure flanges; monolithic heads or bottom pieces; basket components; manifolds or, in the extreme, the entire structure. Foundry practices are well developed. One important consideration in casting and subsequent forging is the presence of voids and inclusions. A casting void or inclusion will become a crack when the component is forged. Thus, it is important to perform nondestructive examinations on these components during the forming process. ASME Code, Sect. V, covers criteria for such examinations.

Frequently, physical and chemical tests are performed on completed castings and forgings to verify material parameters. The foundry must be informed of these requirements such that the appropriate amount of excess material can be included in the molds. A machining allowance is also required on a casting or forging. These matters are best left to the foundry, with the designer simply specifying the requirements.

13.7.1.4 Machining

Essentially no restrictions apply to any of the activities generally classified as machining; these activities include milling, boring, drilling, broaching, planning, and lathe operations. When performing these operations on austenitic stainless steel, one must be aware of the material characteristics, but this should not become a problem if the material is being processed at an experienced shop. For example, these stainless steels should not be exposed to chlorinated cutting fluids.

The surface finish for exposed components is important if the component surfaces are going to experience a contaminated environment in service. All such surfaces should be 125 root-mean square (RMS) or better unless a postfabrication surface improvement such as glass bead impingement or electropolishing is to be applied. In these latter cases, the surface finish may be 250 RMS or better.

13.7.2 Welding

Production welding shall not be undertaken until the welding procedure and the welders or welding operations have been qualified. All such qualifications must be approved by the inspector. Brazing and pressure-welding processes should not be used.

13.7.2.1 Welding Processes and Filler Metals

Stainless steel to stainless steel. Any arc-welding process may be used without including impact tests as a part of the procedure qualification.

Stainless steel to carbon steel. Any arc-welding process may be used. The filler metal shall be stainless steel American Society for Testing and Materials (ASTM) A298 or A371 class E309, Inconel ASTM B295 class ENiCrFe-2, or ASTM B304 class ERNiCrFe-6.

No impact tests are required.

Carbon steel to carbon steel. The shielded metal-arc process may be used with ASTM A316 class E8016 or 8018-C1, -C2, or -C3 electrodes without requiring impact tests as a part of the procedure qualification. The use of other electrodes with the metal-arc process or the use of other processes (e.g., the inert gas-metal-arc or submerged-arc processes) requires three impact tests of the deposited weld metal. The weld must have an average of not less than 20 N·m (15 ft·lb) of impact energy (Charpy keyhole) at –40°C (–40°F) to qualify as an acceptable process.

Covered electrodes. Covered electrodes should be dry when used. Except for 7010-A1 electrodes, covered electrodes should be used within 9 h after their removal from a sealed receiving container or a vented electric oven maintained at 40–175°C (105–350°F). If not used within 9 h, the electrodes should be placed in a vented oven maintained at 65–175°C (150–350°F) for at least 8 h. If the electrode manufacturer's recommendations for drying do not allow the previously described method, approval of the drying method to be used should be obtained from the purchaser. Type 7010-A1 electrodes should not be held in heated ovens.

The fabricator's QC program should establish and enforce written procedures for (1) procurement of all welding filler metals and fluxes; (2) an identification, storage, and issuance system; and (3) records to ensure the use of the correct filler material.

13.7.2.2 Qualification of Welding Procedures

Each welding procedure to be used in construction should be qualified by the manufacturer according to Sect. IX of the ASME Boiler and Pressure Vessel Code. In addition, the test welds should be subjected to the same nondestructive inspection methods as will be required for the production welds. The welds must pass the NDT acceptance criteria established for the production welds in order to be acceptable. A copy of the procedure qualification test report and the welding procedure should be, after approval by the inspector, incorporated into the fabrication record.

The welding procedures should include control of the width of any bead of welding as follows. String beading is preferred. Weave-beading with covered electrodes shall not result in a weld bead with a width greater than

1. six times the diameter of the electrode core wire for 70xx and 80xx electrodes,
2. three times the diameter of the electrode core wire for Type 300 series stainless steel electrodes, or
3. three times the diameter of the core wire for Inconel filler metal.

Wash passes or remelting to improve the surface appearance should be avoided.

13.7.2.3 Qualifications of Welders and Welding Operators

Performance qualifications of welders and welding operators should conform to Sect. IX of the ASME Boiler and Pressure Vessel Code and the NDT tests defined in Sect. 13.7.2.2. A copy of the performance qualification test report should be, after approval by the inspector, incorporated into the fabrication record.

A test conducted by one manufacturer should not qualify a welder to do work for any other manufacturer.

13.7.2.4 Lowest Permissible Temperature for Welding

Preheat temperatures should be established according to the weld procedures. However, when the temperature of the base metal is lower than 15°C (60°F), the area to be welded should be preheated until it is warm to the hand. All surfaces should be dry and protected from rain, snow, and high winds.

13.7.2.5 Fitting and Alignment

Edges to be weld should be uniform and free of all foreign material. Parts to be welded should be fitted, aligned, and retained in position during the welding operation such that at the point of welding the root face should not be offset by more than 1.6 mm (1/16 in.), or one-eighth of the nominal thickness of the material, whichever is smaller. Joints should be held adequately by suitable clamps or sufficient tack welds, and a welding sequence should be used to attain the complete joint spacing and alignment. Tack welds should be free of cracks or other discontinuities and must be completely fused into, and form a homogeneous part of, the root weld layer if they are not removed as the welding progresses.

The edges of butt joints should be held during welding in such a manner that the tolerances, which are stated in Sect. 13.7.3, are not exceeded in the completed joint.

13.7.2.6 Cleaning of Surfaces to be Welded

Surfaces to be welded should be clean and free of foreign material such as grease, oil, lubricants, and marking paints for a distance of at least 75 mm (3 in.) from the welding edge. When weld metal is to be deposited over a previously welded surface, any slag should be removed.

13.7.2.7 Joints—Alignment Tolerances

After welding, the offset should not exceed 1.6 mm (1/16 in.) or one-eighth of the nominal thickness of the part at the joint, whichever is smaller. Weld taper in excess of this amount should be avoided unless detailed on the engineering design drawing.

13.7.2.8 Finished Joints

Joints should have complete penetration and should be free from cracks, severe undercuts, overlaps, abrupt ridges, or valleys. Fillet welds should have complete fusion at the root of the fillet. To ensure that the weld grooves are completely filled so that the surface of the weld metal at any point does not fall below the surface of the adjoining part, weld metal may be built up as a reinforcement on each side of the joint. The thickness of this reinforcement on each side should not exceed the following dimensions:

Thickness of part (mm/in.)	Maximum thickness of reinforcement (mm/in.)
≤ 12.7/0.5	2.4/0.094
> 12.7/0.5 < 25.4/1	3.2/0.125
> 25.4/1	4.8/0.187

13.7.2.9 Miscellaneous Welding Requirements

The reverse side of double-welded butt joints should be prepared by chipping, grinding, or melting out so that sound metal is secured at the root of the weld before filler metal is applied from the reverse side. This requirement is not intended to apply to any process of welding by which proper fusion and penetration are otherwise obtained and by which the base of the weld remains free from impurities.

If the welding is stopped for any reason, extra care should be taken when restarting the process to achieve the required penetration and fusion. For submerged arc welding, chipping out a groove in the crater is recommended.

In instances where single-welded joints are used, particular care should be taken during alignment to ensure complete penetration and fusion at the root of the weld over its full length.

In the case of plug welds, a fillet around the bottom of the hole should be deposited first.

13.7.2.10 Repair of Weld Defects

Visible defects such as cracks, pinholes, and incomplete fusion, as well as defects that can be detected only by prescribed examinations or tests, should be removed; then the joint should be rewelded.

Approval to make a weld repair and approval of the repair procedure should be obtained from the purchaser before the removal of metal from and the repairing of a defective weld area. Repair welding should be done in a manner conforming to the requirements of the purchaser's specifications. The repaired weld should be retested, and, in order to be acceptable, it should meet the quality requirements for the original weld.

Removal of undercutting by reduction of the base metal section adjacent to the welded seam should be avoided.

13.7.3 Weld-Metal Cladding of Carbon-Steel Qualification Requirements

A separate welding procedure should be qualified for the corrosion-resistant weld-metal overlay cladding of carbon-steel-base metal in accordance with Sect. IX of the ASME Boiler and Pressure Vessel Code. In addition, when any of the changes listed below are made, the procedure should be requalified. Although other changes do not require requalification, the procedure specification must be corrected to show the changes. (Note that these requirements are similar to those specified in Sect. III of the ASME Boiler and Pressure Vessel Code, 1992 edition.)

13.7.3.1 Procedure for All Welding Processes

1. A change from one welding process to any other welding process or combination of welding processes.
2. A change in the composition of the deposited weld metal from one A No. in Table QW-442 of Sect. IX of the Code to any other A No., or to an analysis not listed in the table. However, each American Iron and Steel Institute (AISI) type of A-7 or A-8 analysis (Table QW-442) requires separate qualification.
3. The addition of welding positions other than those already qualified.
4. A change in the specified preheating temperature range.
5. A change in the specified heat-treating temperature or an increase of 25% of more in the total time at temperature.
6. A change from a multiple weld layer to a single weld layer, or vice versa.

13.7.3.2 Procedure for Shielded Metal-Arc Welding

1. A change in the electrode diameter used for the first layer.
2. A change from one F No. in Table QW-432 to any other F No.
3. An increase of more than 10% in the amperage used in the application of the first weld layer.

13.7.3.3 Procedure for Submerged Arc Welding

1. A change in the composition or type of flux used. Requalification is not required for a change in flux particle size.
2. A change from single-wire to multiple-wire techniques.
3. A change from ac to dc or from dc to ac current, or a change in the polarity of the current.
4. The addition or elimination of oscillation of the electrode.
5. A variation of more than 10% in travel speed.
6. A change in wire diameter.

13.7.3.4 Procedure for Gas Tungsten-Arc Welding or Gas Metal-Arc Welding

1. A change from one gas to another, or to a mixture of gases.
2. A reduction of 25% in the rate of gas flow.
3. A change from an inert gas or a mixture of inert gases to a shielding gas containing more than 2% of an active gas (e.g., oxygen or hydrogen).
4. A change from single-wire to multiple-wire techniques.

5. A change from ac to dc or from dc to ac current, or a change in the polarity of the current.
6. The addition or elimination of oscillation of the electrode.
7. A variation of more than 10% in travel speed.
8. A change in wire diameter.

The procedure qualification should be made on a test plate simulating the conditions to be used in production, except that the test plate may be thinner. However, the thickness of the plate should not be less than either 19 mm (3/4 in.) or the thickness of the fabrication material, whichever is less. The postweld heat treatment of the test plate shall be equivalent to that to be applied to the parts, except that the total time at temperature may be achieved during one heating cycle.

The weld overlay surface should be examined using a liquid penetrant in accordance with Sect. V of the ASME Boiler and Pressure Vessel Code. Following this examination, the test plate should be sectioned to make four side-bend test specimens, two of these parallel and two perpendicular to the direction of welding. These specimens should have dimensions identical to those of the guided side-bend specimens noted in Sect. IX of the ASME Boiler and Pressure Vessel Code. They should be bent in a test jig and should meet the acceptance requirements of Sect. IX. In addition, a chemical analysis should be obtained from the overlay at a depth from the surface of at least 0.5 mm (0.020 in.) The chemical analysis obtained should be within the range of analysis given in the procedure specification.

13.7.3.5 Performance Qualification Test

Welders and welding operators should be qualified for working with metal plate having a thickness that is either not less than 19 mm (3/4 in.) or is equal to that of the material to be used in fabrication, whichever is thinner, in accordance with the requirements of a qualified weld-metal overlay cladding procedure specification.

The essential variables of paragraph QW-400 of Sect. IX of the ASME Boiler and Pressure Vessel Code should apply. The test plate should be subjected to the penetrant and bend tests as noted in Sect. 13.8.1., except that the chemical analysis need not be made. Any welder or welding operator who qualifies the procedure is automatically qualified.

13.7.4 Joining Integrally Clad or Weld-Metal-Overlay Clad Material

Each welding procedure used for joining clad material should be qualified by the manufacturer in accordance with Sects. VIII and IX of the ASME Boiler and Pressure Vessel Code. A separate welding procedure (and qualification) is required for welds that join clad material to austenitic stainless steel. Test welds should be heat treated if the fabricated material is to be heat treated.

The performance qualification of welders to join clad materials should conform to Sects. VIII and IX of the ASME Boiler and Pressure Vessel Code. Performance tests, involving clad material should be made by a welder before he is permitted to weld the base metal, cladding, or composite joint.

13.7.5 Postweld Heat Treatment

Any postweld heat treatment procedure shall be documented, submitted to the cask owner for review and approval, and made a part of the fabrication record.

13.7.6 Installation of Neutron Shields

Neutron shielding is unique to packages that are designed for high-burnup spent fuel or for neutron sources. Shields are solid or liquid, with the former being more frequently specified for current casks even though the latter has been used to a significant degree. The liquid shields offer good heat transfer (through convection), but they can suffer a loss of shielding under accident conditions or operational error. Solid shields retain integrity better than and require less maintenance than do liquid ones, but solid shields usually require conductive circuits (e.g., imbedded copper strips) to permit heat transfer through them because of their low thermal conductivity.

13.7.6.1 Liquid Neutron Shields

Liquid neutron shields are usually mounted external to the heavy-metal gamma shielding; for some neutron source packages, there is no gamma shield. The liquid, frequently water or a mixture of water and ethylene glycol, is contained in a tank or series of compartments. The compartments are equipped with drain, fill, and overpressure protection devices and have expansion accommodation chambers for the liquid.

Construction of the liquid neutron-shielding containment vessel follows the rules generally applicable to that for the primary system containment (i.e., cask cavity), although the safety classification is somewhat lower. This shielding and its containing structures are regarded as having nuclear safety significance and must be classified accordingly. NRC documents (NUREG/CR-3854 and NUREG/CR-3019) suggest that neutron shields (or, more precisely, the structures that contain the shielding) be fabricated following ASME Sect. III, Subsect. NF, or ASME Sect. VIII, Division 1, rules.

Because of the nature of the shield material, liquid leak-testing is mandatory. A hydrostatic test at 125% of the design pressure should precede the leak test. No leakage of the test liquid (assumed to be water) is permitted, and the determination of leakage can be conducted in conjunction with the hydrostatic test of the shielding containment(s). Alternative leak-testing methods are acceptable. Access to all external surfaces of the neutron-shielding containment structure is necessary. Guidance for hydrostatic testing can be found in ASME Boiler and Pressure Vessel Code, Sect. III, Subsect. NB, Article NB-6000, and Sect. VIII, UG-99. ASME Code, Sect. V describes other leak-testing methods.

The fabrication record for the shield components should contain all the data prescribed by the fabrication specification, and will usually contain data similar to that developed for the cask containment.

The liquids that actually constitute the shield must be carefully controlled. The water should be deionized and demineralized. Material certifications and material safety data sheets must be

obtained for any chemical additives to the water. Chemical analyses must be performed and placed in the fabrication records.

13.7.6.2 Solid Neutron Shields

Like the liquid shields, the solid shields consist of materials that are high in hydrogen content (e.g., as polyethylene, borated polypropylene, organic resins, or high-water-content cement). Solid shields are also conventionally mounted external to the gamma shielding.

Some older packagings are designed with the solid neutron-shield material exposed on the cask surface. This spent fuel cask design requires a handling feature that prevents the exposed surfaces from contacting the contaminated pool water at nuclear reactors. Contemporary designs have the solid shield encased within a shell to isolate the shield from the external environment. This shell does not generally form a line of containment in the same sense as that for the liquid shields; thus, the fabrication requirements are not as stringent. However, it is recommended that "containment" components be treated as if they were retaining a liquid shielding medium to ensure an overall high quality level for the packaging. Some reduction of QA requirements may be permitted if the structures are not classified as safety related, but usually following ASME Sect. III, Subsect. NF, or Sect. VIII, Division 1, will result in a quality structures.

The actual assembly of the solid neutron shielding is a function of the specific packaging design and is beyond the scope of this document. However, the fundamental assembly objective is assurance of performance. For shielding, this implies the absence of voids or radiation streaming paths. For shielding with heat-transfer augmentation, this means the absence of voids as well as the preservation of the casks thermal capabilities.

Solid neutron shields are similar to their gamma counterparts in that the correction of a discovered manufacturing error can be a costly and time-consuming activity. The actual solid shielding material must be regarded as being safety related and treated accordingly from the standpoint of material accountability and recordkeeping. This is particularly true if the solid is composed of a number of constituents that must have a specific distribution in the finished product. The distribution issue is further compounded if the material is cast in place during fabrication. This method strongly suggests the need for process qualification and control as well as a sampling plan for confirmation of success.

13.7.7 Impact-Limiter Construction

Impact limiters vary widely in composition and construction. Appendix A provides details of the varieties of materials that have been used for these packaging components.

The NRC regards impact limiters as safety-related components, although they are not classified as high as are primary containment components. The rules of ASME Code, Sect. III, Subsect. NF, or Sect. VIII, Division 1, should be applied to the construction of impact limiters. Note that most of the actual energy-absorbing materials are not in the Code; consequently, the application of ASME rules usually pertains to the structures that hold and retain the energy-absorbing medium. Other than impact limiters that are of solid, metallic construction (e.g., fins, shells of revolution), impact-limiter materials are usually contained in a thin-walled metallic cladding and integrated into a strong steel structure that is the interface/attachment to the packaging.

The most important fabrication consideration is the conversion of the design and analysis parameters into the actual product. This is particularly true for the "soft" limiter materials such as foams and woods. Parameters, such as grain direction, moisture content, bubble size, rise direction, etc., can have significant affect on the limiter performance. If a limiter is either too soft or too hard relative to the design basis, its performance in an accident may be unacceptable. Many materials have a very narrow performance band.

In the case of organic foam limiters, the material is usually foamed in place. This means that some method of nondestructive evaluation must be provided to measure for uniformity of material properties, lack of significant voiding, and completeness of chemical reactions. Process control and qualification is essential to the construction of foam limiters.

Limiters of wood construction rely on careful control of moisture content, wood type, grain direction, and gluing of components. Receiving inspection, including testing for moisture and mechanical properties, is important as is the storage of the wood prior to assembly. The wood components are encased in sealed metal jackets as part of limiter assembly and thereafter do not gain or lose moisture.

Metallic foams and honeycomb components are generally premanufactured, but they should undergo receiving inspection, where may include testing if testing has not been conducted by the supplier. The joining of metallic components may require process development and qualification before actual production.

The application of QA/QC procedures is very important in impact-limiter construction. This extends to the mounting and attachment hardware because retention of the limiter under impact conditions is a fundamental performance criterion. Material and heat-treatment documentation, as appropriate, must be entered into the fabrication record along with the usual complement of items identified in the fabrication specification.

13.7.8 Neutron-Poison Installation

Neutron poisons are frequently installed in fuel baskets to provide criticality control. These poisons are boron-containing components (e.g., borated stainless steel, Boral™, Boroflex™, or boron-carbide-filled tubes). Some poison components are clad to prevent contact with reactor pool water or to prevent the corrosion and/or loss of the boron constituent. Boron stainless steel, on the other hand, requires no encapsulation.

The NRC regards these criticality control components as safety related. The rules for fabrication are in ASME Code, Sect. III, Subsect. NG. This classification is somewhat higher than that for shielding and impact limiters, but it not as high as that for containment systems.

Basket designs tend to include the neutron poison as nonstructural components. That is, under normal and accident conditions, the components have to support only themselves. Nevertheless, their presence is essential to basket performance, and, therefore, their construction and integration into the basket are important.

From a construction perspective, little difference exists between the rules of Subsect. NG and those of Subsect. NB, the highest classification. This means that the traceability of poison

materials, qualification of personnel and processes, documentation of work activities, and inspection and testing of the product must be performed to exacting standards.

Important considerations regarding neutron-poison materials are density, boron distribution, and enrichment (if required). Poison suppliers are able to provide data on their products, but some form of confirmation may be appropriate. For mobile poisons (e.g., boron carbide powder) or those that must be sealed to prevent water intrusion, containment testing is required.

The application of Subsect. NG must extend to the basket cask interface. Indeed, in most applications where poison or geometry controls are required, the entire basket is constructed to Subsect. NG. Particular attention should be paid to attachment welds and containment welds.

NRC regulations discuss the demonstration of poison efficacy. However, this is usually not carried out through a test on the completed basket. Rather, it is accomplished indirectly by examining the actual basket and the documentation supporting its proper construction. This form of validation places significant emphasis on QA/QC records.

13.7.9 Visual and Nondestructive Examination Inspections

The following identify tests that have been accepted by the NRC and will be published as NUREG-1567, *Standard Review Plan for Dry Storage Facilities*. A draft for comment was issued October 31, 1996, and contained this subsection.

Cask components should be fabricated and examined in accordance with an accepted standard used for their design (e.g., for the confinement vessel, the ASME Boiler and Pressure Vessel Code, Sect. III, NB or NC). These codes define the examination requirements by reference to the following sections:

- Sect. II ("Materials Specifications and Properties");
- Sect. V ("NDE Specifications and Procedures"); and
- Sect. IX ("Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators").

The nondestructive examination (NDE) of weldments must be well characterized on drawings, using standard NDE symbols and/or notations [as given in (American Welding Society) AWS A2.4]. Each fabricator should be required to establish and document a detailed written weld inspection plan in accordance with an approved Quality Assurance program that complies with 10 CFR 72, Subpart G. The inspection plans should:

- include as applicable: visual (VT), dye penetrant (PT), magnetic particle (MT), ultrasonic (UT), and radiographic (RT) examinations;
- identify welds that will be examined;
- identify the examination sequence;
- identify the type of examination;
- state the appropriate acceptance criteria; and
- require that inspection personnel be prequalified in accordance with the current revision of SNT-TC-1A (as specified by the ASME B&PVC).

Confinement boundary welds, and welds for components performing redundant sealing, must meet the requirements of Sect. III, NB-5200 or NC-5200 ("Required Examination of Welds"). This

generally requires RT for volumetric examination and either PT or MT for surface examination. The ASME-adopted specifications for RT, PT, and MT are provided in Sect. V, Articles 2, 6, and 7, respectively.

Acceptance criteria for RT should be in accordance with Sect. III, NB-5320 or NC-5320. Unacceptable imperfections such as a crack, a zone of incomplete fusion or penetration, elongated indications with lengths greater than specified limits, and rounded indications in excess of the limits in Sect. III, Division 1, Appendix VI, should be rejected. Repaired welds should be reexamined in accordance with the original examination method and associated acceptance criteria.

For confinement welds that cannot be volumetrically examined using RT, 100 percent UT may be used. The ASME-adopted UT specifications are provided in Sect. V, Article 5. Acceptance criteria should be in accordance with NB-5330 or NC-5330, "Ultrasonic Acceptance Standards," Cracks, lack of fusion, or incomplete penetration are unacceptable, regardless of length.

The NRC has accepted multiple surface examinations of welds combined with helium leak tests for inspecting the final redundant seal welded closures.

Nonconfinement welds of the confinement SSCs should meet the requirements of Sect. III, Subsection NF or Sect. VIII, Division 1, as applicable. The required volumetric examination of welds is either RT or UT, as discussed in Sect. III, NF-5200, and in Sect. VIII, UW-11 of the Code. The appropriate specifications of Sect. V are invoked in Article 1 for RT and in Article 5 for UT. Acceptance standards are provided in Sect. III, Subsection NF, NF-5320 ("Radiographic Acceptance Standards") for RT, and in NF-5330 ("Ultrasonic Acceptance Standards") for UT. Required welds should be reexamined in accordance with the original acceptance criteria.

Nonconfinement welds in the confinement SSCs that cannot be RT examined should undergo UT in accordance with Sect. V, Article 5. Acceptance criteria should be in accordance with Sect. VIII, Division 1, UW-53 and Appendix 12, and the repair of unacceptable defects should be in accordance with UW-38. Repaired welds should be reexamined in accordance with the original examination methods and associated acceptance criteria. The SAR should also justify the rational for not requiring RT examination of these welds.

Finished surfaces of the cask should be examined visually in accordance with Sect. V, Article 9, of Reference 1. The acceptance criteria for VT examined welds should be in accordance with Sect. VIII, Division 1, UR-35 and UR-36, or NF-5360, "Acceptance Standards for Visual Examination of Welds of the Code."

PT testing should be used to detect discontinuities, such as cracks, seams, laps, laminations, and porosity, that are open to the surface of nonporous metals. PT should be performed in accordance with Sect. V, Article 6. Acceptance criteria for confinement welds should be in accordance with Sect. III NB-5350 or NC-5850. Repair procedures should be in accordance with NB-4450 or NC-4550.

Acceptance criteria for nonconfinement welds of components designed to ASME B&PVC Sect. VIII should be in accordance with Sect. VIII, Division 1, Appendix 8, or NF-5360 ("Liquid Penetrant Acceptance Standards"). Repair procedures should be in accordance with Sect. VIII or NF-2500 ("Examination and Repair of Material") and NF-4450 ("Repair of Weld Material Defects").

Fabrication controls and specifications should be in-place and field verifications should be performed to prevent post-welding operations (like grinding) from compromising the design requirements (like wall thickness). The specifications should be clear that reduction of wall thickness at the weld region is unacceptable.

13.8 LEAD POURING

Lead pouring for heavy-metal shielding is a special process that requires experience or the careful development of the pouring process. The lead melting and pouring (or pumping) require unique equipment, including that for the control of the toxic fumes. The lead should conform to ASTM A-29 for grade and quality.

Before any shielding chamber is filled with lead, the integrity of the chamber should be demonstrated by a leak test. This testing should be performed while joints are accessible for repair and before any leak paths can be plugged with molten lead. Any leak-detection method is acceptable as long as its sensitivity is at least 1 cm^3/h of air.

Lead melts at a fairly low temperature (327°C or 621°F); consequently, it has a propensity for solidification during the pouring operation should it contact a colder surface. Premature solidification is a cause of void formation in the shield, which is highly undesirable. To prevent this phenomenon, the lead is usually heated to a temperature substantially above its melting point, and the steel components into which it will be poured are preheated.

Lead pouring should be performed in a single, continuous operation unless an alternative method is developed and proven by a test which will ensure that there will be no localized gaps at the lead interfaces which in turn could result in some high external radiation readings from a radiation source in the cavity of the package. The manufacturer shall prepare a detailed lead-pouring procedure that will provide information concerning the grade and purity of the lead to be used; the grade of any tinning compounds; a description of the heating, pouring, and cooling facilities (including sketches); the sequence of operations; a precleaning or pretinning procedure (if used); the method and speed of pouring; preheating and controlled cooling methods; temperature control requirements and measurements; and details of pouring and vent connections.

After approval by the inspector, the lead pouring shall be performed in accordance with the procedure, and a copy of the procedure needs to be incorporated into the fabrication record.

13.8.1 Lead Bonding

Lead bonding is a subprocess of lead pouring. When lead pouring is performed and the filled structure cools to room temperature, a small gap will be created at the lead-steel interface, resulting from the differences in coefficient of expansion between the two metals. If heat transfer or other considerations require that no gap be present at the lead-steel interface, then lead bonding may become a requirement. Bonding involves "tinning" or precoating the steel surfaces with lead or with a lead-tin alloy before the lead pour. Since this process requires a considerable knowledge of lead-pouring technology, it is important that the bonding process be well planned, closely followed, and documented. Evaluation of the success of the bonding process may be determined by installing a heat source in the cavity of the packaging and measuring the temperature at various locations throughout the shield.

13.9 DEPLETED URANIUM PROCESSING

Depleted uranium (DU) is not a commonly used cask-shielding material, but its high density and metallic form give it properties that are advantageous in applications where packaging efficiency is important.

Although DU has been used in wrought form, the general application is in the form of machined castings. The DU metal is usually alloyed with molybdenum (Mo) to improve strength and ductility without sacrificing density. Alloys ranging from 0.2 to 8% Mo have been used. More exotic alloys have been formulated to further improve mechanical properties, but the effect on density is greater than with Mo. Metallic "derbies" are vacuum melted, and the product is poured into graphite molds. The molds are coated to prevent excessive carbon contamination of the DU. Control of the cooling process is necessary to ensure the correct phase and grain structure. Casting should be performed only by a vendor that is thoroughly familiar with DU metallurgy.

Casting and mold techniques can produce reasonable surface finishes, although the density of surface material can be less than that of the main casting. Rough-machining is recommended after casting. This material machines very much like mild steel. DU metal is pyrophoric in finely divided form such as machining turnings or chips. The material burns somewhat like magnesium, but not as hot. The management of chips and turnings is important, and quantities of cutting fluid should be used to reduce temperatures. Freshly machined surfaces will oxidize quickly, leaving a dark, thin film resembling soot; consequently, cleanliness and a dry storage environment are important.

Although not highly radioactive, DU metal chips and turnings must be collected and disposed of properly. Similarly, the machining operation requires ventilation, monitoring, isolation, protective clothing, and a monitoring program. Machine tools dedicated to this work require decontamination before being returned to general service. A state or an NRC license is generally required for DU storage and processing.

13.10 ACCEPTANCE TESTS

Following the fabrication of the shipping packaging, it must be subjected to a series of confirmatory tests to validate certain performance parameters and its general operability. These tests are usually performed at the fabricators facility under controlled conditions.

13.10.1 Test Planning

The acceptance tests are an integral part of the fabrication. Planning and documentation for these tests are as important as they are for any other critical fabrication activity. A test plan must be written and approved; approval should include QA sign-off. From the test plan, specific procedures are written, again with the appropriate approvals. The procedures shall include the format for the recorded data, which will become a permanent part of the fabrication record.

In most instances, it is necessary to interact with the cask designer for confirmation that the test results are within the package's performance envelop. The designer may also be involved in test-data reduction and interpretation.

13.10.2 Heat-Transfer Tests

Although heat-transfer tests are not required by NRC regulations, for casks in which the thermal load is high, it is considered prudent to conduct some form of verification test. Thermal testing is usually conducted under simulated normal conditions of transport, except that the test ambient is usually not the regulatory 100°F air temperature. For obvious reasons, the fire-accident test is not performed. The results of the acceptance test can be adjusted for ambient air conditions and compared with the calculational predictions. This benchmarking is also useful for operations by correlating heat load with exterior surface temperature at a given ambient. This permits the verification of heat load by taking surface temperature measurements.

In some cases in which an off-normal scenario can be simulated without damaging the packaging, a test may be appropriate. For example, if a liquid neutron shield could be inadvertently drained and change the thermal characteristics of the cask, this might be a test candidate. A low-power or transient test might be acceptable if the full heat load under steady-state conditions could potentially damage the packaging.

The decay heat is best represented by electrical heaters. To accomplish this, a test closure (or lid) must be constructed for the electrical, as well as for any internal thermocouple penetrations. The heat load should be variable over the expected range, and the axial distribution of the heat in the cavity should simulate that expected in service. The test closure must be capable of sealing the cavity to permit the use of an inert gas during the test should such a gas be used in normal operations. Some thermal tests are conducted under a vacuum to establish the thermal behavior of the contents (e.g., the basket) under vacuum-drying conditions.

The packaging should be positioned in its normal transport orientation. The test facility should be essentially draft free. Instrumentation consists principally of thermocouples, although a pressure measurement of the cask cavity would be appropriate if high values are expected. The test should be conducted, initially at a low-power setting and then at progressively higher values, up to the design-basis heat load. At each power increment, the packaging should be allowed to reach steady state.

One recommended test segment is to begin with the packaging at ambient temperature and perform a heat-up transient at the design heat load. This step will produce a transient envelop that will be useful during operation. The generated transient, adjusted for ambient, can be used to confirm the content's heat load. Because reading steady-state conditions can take a very long time to occur even for small casks, the initial portions of the heat-up transient for the actual contents can be compared with that recorded in the test at the design-basis heat load. If the actual transient is within the test transient envelope, then the actual heat load is within the package limit. Such data can be incorporated into the packaging operating manual.

13.10.3 Shielding Integrity Tests

Shielding integrity tests are very important because a void or low-density region will create a "hot spot" or elevated radiation region on the cask surface when in service. Both gamma and neutron shielding, as appropriate, should be tested for integrity. The methods used to perform these tests vary.

13.10.3.1 Gamma Shielding

The integrity of either lead or uranium gamma shielding is usually tested using a gamma ray source. Shielding analyses are made to determine the minimum acceptable void or reduced density. As discussed below, when the gamma probe or gamma scan method is used, one can expect to detect a 2 to 3% reduction in shielding effectiveness. A 5% reduction in effectiveness is generally regarded as the rejection (or rework) threshold.

These methods use a scintillation detector and recorder on the outside of the shield and a radium or ^{60}Co source on the inside. Thicknesses up to 30 cm (12 in.) of lead, or equivalent, can be tested. A count rate is established using a thickness calibration standard. Deviations from the standard, in the form of increased count rate, indicate a reduction in shielding effectiveness. Large shields are usually blocked off into grids so that the measurements can be systematically conducted. For a gamma probe test, the source and detector are stationary; for a gamma-scanning test, they are moving. In this latter case, with the proper selection of source and detector, scanning speeds up to 25 mm/s (1 in./s) can be achieved.

Other less-used methods for gamma-shield integrity testing include neutron probing or scanning and ultrasonics. The neutron methods are applied similarly to that described for gammas. Because sources and detection equipment are expensive, this is not a popular technique. Ultrasonics can work well on wrought shielding forms (i.e., rolled sheet or plate), but are inaccurate when applied to a shielding form with a large grain structure such as cast material. The large grain structure reduces penetration ability and complicates the resolution of the reflected pulses.

13.10.3.2 Neutron Shielding

Solid neutron shields are less susceptible to voiding because they are either cold-formed from manufactured material or poured at relatively low temperatures and are not prone to premature solidification. Nevertheless, some integrity examination is necessary. Integrity testing methods are similar to those for gamma shields.

It is important to remember that in order to check the integrity of the neutron shield (as opposed to its effectiveness), it is best to employ the use of a gamma source and test standard. The gamma source selected will have to be of reasonably low energy such that shielding variations will have a reasonable effect on the count rate.

Neutron shields are generally much less dense than are gamma shields—hence, the need for a low-energy source.

If the neutron shielding cannot be tested separately from the gamma shielding, the measurement is more complicated. In these cases, the gamma method may still work—but with a stronger source behind the gamma shield. The gamma shield must be tested first to form a baseline for the subsequent neutron-shield testing.

Neutron probing or scanning might be effective for inseparable gamma and neutron shields. Ultrasonics could be used for small-grain neutron shields.

13.10.4 Containment Verification Tests

Containment verification tests are appropriate at several stages of fabrication. Usually, the containment system of a shielded package is the inner shell, piping, and valves connected to the cavity, the closure flange, all cavity seals, and closure fasteners. Sometimes the last item, fasteners, is not considered part of the containment system because fasteners do not "see" cavity pressure or cavity environment.

13.10.4.1 In-Process Testing

In the course of packaging construction, there are situations when the inner containment has been completed but additional fabrication activities, such as lead pouring, will obscure it from further detailed inspection. Although the containment might have already been examined by nondestructive examination (NDE) methods, the opportunity exists to temporarily close the interior and perform an integrity test. This test could be as simple as internally pressurizing the containment with air and applying a soap solution on the exterior to look for signs of bubbles. On the other hand, a helium leakage test and assessment could be performed if the design-basis leakage limit is particularly restrictive (e.g., requiring the containment of the packaging to be "leak tight").

13.10.4.2 Final-Containment Verification Testing

The final-containment verification test is performed at the end of construction on the finished packaging. This acceptance test should be conducted according to Sect. 6.3, "Containment System Fabrication Verification," of ANSI N14.5-1987, *American National Standard for Radioactive Materials—Leakage Tests On Packages for Shipment.* Chapter 8, "Containment," discusses this ANSI standard.

Containment verification tests should be to the design-basis leakage rate as determined through the use of the N14.5 methods. This design-basis leakage rate pertains to the entire packaging which requires an integrated leakage assessment from all potential sources. Some interpretation is necessary for the integrated assessment. Many cask designers and vendors choose to test the cask penetrations, vent and drain closures, and the cavity closure. Typically, these are gasketed or non-integral (removable) components. Others choose to conduct tests which measure the leakage rate of the entire packaging, usually by "bagging" the cask and performing an outside-to-inside helium-leak test.

This latter test begins to be impractical when a large cask is involved. If component testing is conducted during fabrication, the former testing approach is acceptable. ANSI N14.5, which is discussed in Chapter 8, describes various testing methods.

For packagings whose cavity is pressurized during operation or might become pressurized under accident conditions, the leakage assessment, consisting of a 150% pressure test, is usually conducted. This test is generally a hydrostatic test which follows ASME Code. The hydrostatic test is actually a stress test (not a leakage test) of the containment boundary. If any distortion of sealing surfaces or disruption of the containment boundary occurs as a result of the hydrostatic test, it will be detected by the subsequent leakage test. The hydrostatic test is too insensitive to be used for leakage assessment unless the design basis leak rate is quite large. Leakage testing

using a reduction in pressure of a water-filled and pressurized cavity is a recognized method and should not be confused with a simple hydrostatic test.

13.10.5 Neutron-Poison-Efficacy Tests

In packagings in which fissile materials are to be shipped, neutron poisons such as Boral™ or Boroflex™, are often used as part of the internal structure (basket). The functioning of these components has direct safety significance with respect to the prevention of accidental criticality. The efficacy of these materials is related to the distribution and concentration of the poison medium. Tests for efficacy are generally beyond the capability of a packaging manufacturer. The usual procedure is to obtain verification of the efficacy of the material from the material manufacturer. The test results are analogous to material certifications and should be treated as such in the QA and record retention processes. Similarly, the supplier or tester must be qualified to attest to the efficacy of the material since it is a safety-related component of the packaging.

13.10.6 Operational/Functional Tests

The operational and functional tests involve putting the packaging and its components through the routine handling operations. The type of tests include closure removal and replacement, lifting and uprighting, load testing, filling and draining, seal replacement, basket insertion/removal, spacer insertion/removal, and checking the operation of the lifting-gear, vacuum drying system (if required), and any special tools or equipment.

The functional tests should duplicate the actual operating environment as nearly as possible. For underwater service, duplication may not be feasible, but the remote operation can be simulated. Sufficient repetitions should be performed to be certain of acceptable operation in actual service. This point is the time at which the adjustments should be made, not later in the field.

13.10.7 Testing Documentation

Acceptance tests are an integral part of the packaging fabrication and thus are subject to QA and documentation requirements to the same extent as are other fabrication operations. In some instances, the test records must be compared with the analyses submitted as part of the packaging safety analysis report to confirm particular functions.

The acceptance test records should be retained in the fabrication record file. It is not unusual to create separate reports on particular acceptance tests. For example, the lifting-device functional test and load test are often documented in a report that can be given to a user that requires these data to fulfill certain regulatory requirements. Similarly, thermal acceptance test data have been incorporated into operating manuals as heat-up curves for the verification of the contents' heat load.

13.11 RECORDS AND REPORTS

At the risk of oversimplification, all records and documents generated in the course of packaging fabrication must be retained for three years beyond the lifetime of the package (10 CFR

Part 71.91). The exact list of documents to be retained is a function of the packaging and should be identified in the fabrication specification.

A systematic method of categorization is an important aspect in the retention of fabrication records. Such categorization will greatly aid in the auditing and retrieval of records, especially if time has elapsed since construction or if personnel or package ownership has changed.

13.11.1 Manufacturing Drawings/Design Drawings/As-Built Drawings

All drawings used in the production of the packaging must be in the record file. These drawings include those manufacturing or shop drawings that were created from the design drawings to facilitate the fabrication of the package. Also included must be any drawing changes that were made in the course of construction. Often these are in the form of Engineering Change Notices (ECN) or a similar document that accompanies the work product. Frequently ECNs are maintained to the end of a project, and then the drawings are updated only once rather than at the time of the issuance of each ECN.

As-built drawings are not required for every package. When they are required, usually only critical dimensions are measured and recorded. An as-built drawing is specific to each packaging being constructed because dimensions are expected to differ somewhat from unit to unit. The as-built dimension drawings are frequently given numbers different from the design or shop drawings so that the numbers can be related to their respective packaging. As-built drawings are often carried forward for the lifetime of a packaging to record any modifications or repairs that occur in service.

13.11.2 Process Procedures

The preservation of the manufacturing processes is very important because packagings must be repaired or modified according to the procedures that were used during the original construction. In some cases, procedures are unique to a particular vendor—welding procedures are a typical example. However, having the procedure will allow another party to write a procedure that is comparable to that originally used.

13.11.3 Processes and Personnel Qualifications

These records are generally used to defend the packaging fabrication activities in the event of an audit following the completion of fabrication. In the event of packaging repair or modification during service, requalification of processes or personnel might be necessary.

13.11.4 Job Package

The job package is the collection of shop "travelers" and other in-process documentation that indicate that certain manufacturing steps and inspections were performed and to prove that authorized signatures accompanied the activities. The package would be supplied as part of the postfabrication audit documentation.

13.11.5 Material Certifications

It is extremely important to retain material certifications. These are the qualification documents for every material used in the packaging. It is vital that each part be traceable to its material certification and that each certification document have a consistent numbering system to create this "paper trail." This tracking becomes complicated when material is withdrawn from a vendor's stock or when one lot of material is used on a number of different parts. Nevertheless, the lack of material traceability has resulted in more packaging audit problems than any other issue.

The certification and the ability to relate it to the appropriate component are also important when performing repairs or modifications, particularly where replacement materials might be needed. Another important use of the documentation is in the event of material failure of a component when it then becomes necessary to engage in material forensics to determine the cause of the failure. This occasionally occurs with heat-treated materials.

13.11.6 Certification of Completion

Every packaging must have a certification from the vendor which attests that the packaging has been constructed in full compliance with the drawings and specifications. This stipulation keeps the vendor as a responsible party during part of the service period. Any vendor unwilling to offer such a certification should not be considered during the selection process. Although packaging problems in service are issues for the user, nevertheless, there is some redress possible from the vendor should the problem be related to construction.

13.12 PACKAGING DELIVERY

The delivery of the completed packaging must be defined in the fabrication specification. The owner should take delivery only after being satisfied itself that the design specification has been met, the records are complete, all QA issues have been resolved, the contracted work scope is complete, and the fabricator has executed a Certificate of Completion.

The packaging should be delivered clean and ready for service. Some components (i.e., lifting devices, spare parts, alternative internal structures, etc.) may be shipped separately. Care should be taken to prevent damage during shipment. Most of the shipping details are a function of the size of the packaging and its components and of the quantity of packagings to be delivered.

A receiving inspection is appropriate even though the principal packaging acceptance occurs before taking delivery.

14. Package Maintenance

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14.1 INTRODUCTION

This chapter describes the principal aspects of package maintenance and repair. Its emphasis is on the content of Sect. 8.2 of the *Safety Analysis Report for Packaging* (SARP) for a package, as recommended by the U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide 7.9.

Package maintenance can be either planned or unplanned. Planned maintenance is performed on a periodic basis that is established by the designer and/or specified in the Certificate of Compliance. This maintenance usually consists of a series of inspections and tests to demonstrate that the package retains the capability specified in the package SARP for safely containing and transporting the authorized contents. Periodic maintenance requirements occur on some cyclical basis specified by the designer. In the United States the cyclical basis is typically the calendar year, and the periodic maintenance is called an *annual.*

Unplanned maintenance occurs when a defect is found on inspection of a package before or during use or when handling or transport conditions cause the package to be impaired or otherwise not in conformance to its licensed configuration. For convenience, this unplanned maintenance is herein referred to as *repair.*

Some actions that could be considered maintenance may be required on a package per-use basis, such as seal or O-ring replacements or specific inspections and tests. The requirements for these ordinary or routine tasks are contained in the package operating procedure(s) presented in Chapter 7.0 of the SARP.

Under normal operating conditions, some package maintenance that had been neither planned nor scheduled may also be required. This maintenance should be considered incidental maintenance and could include, for example, the replacement of broken thread inserts (such as helicoils); replacement of damaged couplings and valved fittings; or replacement of a damaged, removable part (such as a port cover or bolt). If the cask designer describes the replacement of these items in the package procedures, time can be saved in returning the package to service. If procedures must be developed, then considerable time can be lost in the procedure development, review, and approval process. The correction of the defect is technically a repair and must be documented as such.

Maintenance activities must be performed according to an approved quality assurance (QA) program, and documentation of the completion of the maintenance must be retained for the life

of the package. For multiple packages of the same design, those records must be kept by the serial number of each individual package.

Finally, completed maintenance or repair activities must always return the package to the "as certified" condition. If this cannot be done or if a condition is found that has resulted in a degradation of package performance, then the regulating agency must be notified of the condition, and the package must be removed from service.

Maintenance must also be performed on the items that are provided to handle, load, transport or otherwise use the package. For convenience, these items are often grouped together under the title of "transporter maintenance," and "ancillary equipment maintenance." The principal difference in performing maintenance in these two areas is that package maintenance is provided for in the SARP and the Certificate of Compliance, while maintenance of the transporter and ancillary equipment is not.

14.2 PACKAGE SYSTEM MAINTENANCE OBJECTIVES

The package system is considered to include the package, transporter, lifting equipment, and required ancillary equipment. For purposes of this discussion, the package is considered to be the certified system, and its maintenance requirement is established by the SARP and by its certificate conditions. The other items—transporter, lift equipment, and ancillary equipment—are not typically part of the certified system, but are required for the package to operate. If the package designer supplies this equipment to the package user, then the designer must also provide the maintenance requirements that are identified for these items. For some items, such as transporters and lift fixtures, industry standards may apply that specify both per-use and periodic maintenance. For example, for trailer-type transporters, American National Standards Institute, Inc. (ANSI), N14.30 may apply, and for lift fixtures, ANSI N14.6 could apply. Inspection and maintenance requirements from these standards must be incorporated into maintenance and use procedures for these items. Other items supplied with the package, such as pumps or leak detectors, may have maintenance requirements that must be established and documented by the designer in appropriate procedures.

As discussed in Chapter 12, the designer should strive to limit the items that are supplied with the package. The reason is that packing, unpacking, handling, and decontaminating the items are a burden for both the package user and the package owner when the shipping program begins and when it ends. Moving special tools or equipment into and out of a spent fuel pool or any other contaminated area requires that the tools or equipment be wiped down, a procedure which generates radioactive waste. In addition, if the package relies on a special tool or unique equipment item, then the reliability of the package depends on the reliability of the tool. To the greatest extent possible, the package designer should design the package so that only common or standard tools, fittings, instruments, and devices, which can be obtained by the package user and remain at the user's facility, are required with the package.

A principal purpose of periodic maintenance is to ensure by inspection and testing that the performance of the package has not deviated or degraded during use from the evaluated configuration that is specified in the SARP. This maintenance ensures that the package system is in the condition described by the SARP and that it continues to meet its performance-based

requirements. The designer specifies the periodic maintenance inspections and tests to be performed in Sect. 8.2 of the SARP.*

The maintenance requirement is unique in that there is no specific mention of required maintenance in the *Code of Federal Regulations* (10 CFR 71), but it is described in Regulatory Guide 7.9 as an element of the SARP. The basis of the maintenance requirements are centered primarily in the codes and standards that must be applied to elements of the package during design and fabrication. The package must continue to meet these codes and standards during its useful lifetime, and maintenance activities are used to guarantee this condition. For example, the ANSI standard used to determine the leak rate of a package specifies design constraints, processes, controls, and the qualifications of technicians who employ it to ensure the correct application of the standard and the continued satisfactory performance of the component.

14.2.1 Calendar vs Cycle-Based Periodic Maintenance

Foreign packages (those certified in countries other than the United States and Canada) typically base periodic maintenance on cycles-of-use of the package. This practice arose because, in moving spent fuel from Taiwan, Korea, or Japan to Europe, a package would be in transit and at the loading site for a long period of time. Consequently, it was not uncommon for an annual periodic maintenance requirement to arise after only one roundtrip shipment of a spent fuel package. To overcome this one-annual-for-one-shipment condition, the use of a cycle-based maintenance program was adopted.

Cycle-based maintenance has not been adopted in the United States because the travel time of packages between possible origins and destinations is short, thus making it much less likely that a scheduled maintenance requirement would occur during transit.

It is standard practice in the United States to use calendar time periods for identifying when periodic maintenance should be performed. With periodic maintenance based on the calendar, the cask designer must specify the replacement of components that have a cycle-based failure mode at the appropriate periodic intervals. The cask designer should not mix cycle-based maintenance with an annual maintenance requirement because this could be expected to result in a package being removed from service for maintenance more than once per cycle, or period, thus reducing package use and increasing the records-keeping requirement for each package.

The package designer should consider incorporating a statement in Sect. 8.2 of the SARP to the effect that specified maintenance need not be performed during a period of nonuse of the package, but that the package will be brought to full compliance before the first use after the inactive period. The principal purpose of this is to avoid the cost of maintenance of a package that has low or infrequent use. The avoided costs are measured in many ways, including reduced personnel radiation exposure burden and reduced production of radioactive wastes.

*Even if no other requirement existed, periodic maintenance would be established because it has long been recognized as a good business practice, particularly for high-value components and for components whose failure could result in significant risks to package users or to the public. The continued satisfactory performance of radioactive materials shipping packages is very important to the preservation of public health and safety. The use of a nonconforming package or failure of any package during use is not acceptable.

14.2.2 Criteria for Testing and Repair or Replacement of Components

Typically, two maintenance opportunities occur for the inspection, repair, replacement, or testing of the package or its components: The first is the scheduled replacement or rebuilding of components on some periodic basis, which might, or might not, be annually. The second is the replacement or rebuilding of components on an as-needed basis.

Scheduled maintenance must be established by the carefully considered judgment of the package designer. The designer must establish the periodic requirement based on the assumptions of the analysis as reported in the SARP, knowledge of the materials used in fabrication and their performance parameters individually and in combination, the effects of cycles of use, good practice, and the potential for degradation of the material or its performance over time. The frequency of scheduled maintenance must be often enough (and the tests must be sensitive enough) to ensure that incipient failure or degradations in performance are identified before package performance is below the performance level established in the SARP.

Design engineers may specify unique inspection, test, or replacement requirements (these are collectively referred to as "tests" herein) on components based on loading, thermal cycling, radiation, or other similar factors. However, the designer should not specify more tests than are necessary to determine the package condition, nor should those tests be more complex than are necessary. The principal reason is that the regulator will accept the designer's plan for testing so long as, in the regulator's view, the tests confirm the condition that must be demonstrated. The regulator will not generally recommend a test cycle that is shorter or a test that is easier to perform even if those tests would provide the same level of assurance. Specification of the tests rests with the designer.

The tests specified by the designer should always consider the package conditions and the environment of the package at the time of the tests. The designer should recognize that a package intended for use in loading pools will be more contaminated on both its interior and exterior surfaces as a result of its being placed in pool water. For pool-loaded packages, there is a tendency for some residual contamination to remain in the cask cavity after the water is drained because of torturous drain paths, shelves, and other water traps in the cask and its piping and valving, and the configuration of the contents. Packages loaded dry (and having dry contents) typically have less residual surface contamination, but they cannot be considered to be (radiologically) clean. The residual contamination in the package cavity, piping, valves, and internal baskets or on contents spacers can pose a significant radiation risk to technicians performing the tests. Further, the tests are typically conducted in the package-handling area of the facility. These areas are not as clean as might be found in a laboratory environment. There is limited space and limited ability to use special tools or test gear, and it is often a radiologically controlled area. Getting test equipment into and out of the area can require significant time in packing and repacking and in decontamination and survey. Further, instruments, tools, or other equipment that become internally contaminated during use may be "lost." Health physics personnel will be reluctant to release items if the amount of internal contamination cannot be determined and if it cannot be shown that the equipment contains no trapped water in internal voids, pipes, hoses, valves, fittings, or other fixtures. The designer should consider the possible radiation exposure that the technician could receive in conducting the test and ensure that dose is as low as reasonably achievable (ALARA). The test procedure should contain the caution or other notes needed by the technician to perform the tests safely.

Another situation that affects maintenance activities involves defects that are found during use. Even though these problems seldom occur, the usual one that is discovered is the failure of a seal, and that defect is most often found during a test to confirm the leak tightness of the package. Because the acceptable leak rate is specified in the acceptance criteria, the package user can replace seals or take other necessary actions to achieve the required leak-rate conditions.

It is not uncommon to have damage to the package surface (e.g., scratches, nicks, dents, and gouges) occur during the course of package use, but it can be difficult to describe the criteria under which one dent is not a problem but another one is. In general, nonpenetrating surface damage free of hard edges and cracks is acceptable, while other types of damage are not. In some cases, the designer may have to rely on words that could be considered subjective in specifying visual inspection requirements. Those words could include *distortion*, *bends*, *cracks*, *chips*, *penetration*, *sharp edges*, *elongation*, *fraying*, *discoloration*, not *seated*, *twist*, *thinning*, or the like. The designer can rely upon common meanings and can assume some level of mechanical expertise for the package user.

14.3 PACKAGE MAINTENANCE REQUIREMENTS

The periodic maintenance requirements are generally established by applying information contained in the national or international standards used in the package design, by analysis or conclusions reached by the cask designer, by cask designer experience, by industry experience or good practice, and occasionally by the regulatory authority.

During the course of design analysis, the cask designer may make certain assumptions regarding the performance of a component, or reach a conclusion, which may dictate a periodic maintenance requirement for that component. For example:

1. A structure, called a *former*, was designed for use in a spent fuel shipping cask to convert the circular cavity of the cask to a square hole to accept a square, light-water reactor spent fuel assembly. The former was an aluminum structure, encased in stainless steel, with numerous voids to save weight. The structural analysis, the thermal analysis, and the criticality analysis were based on the assumption that there would be no leakage of water into the void spaces. This assumption dictates that some periodic inspection or test be performed to ensure that no inleakage of water has occurred during pool loading or unloading of the package. Such a required inspection should be included in Chapter 8.2 of the SARP for this package.

2. A Specification 7A drum-hoop closure bolt was analyzed to determine the fatigue stress resulting from cyclical tightening of the bolt. The analysis showed that the bolt should be replaced about every 200 shipments or, according to the designers evaluation, about every 3 years. As noted previously, such a required replacement should be included in Chapter 8.2 of the SARP for this package.

The point of this discussion is that the periodic, or per-use, maintenance requirements of a package must be developed from a careful review of the other chapters of the SARP for the package. Requirements specified by use of standards, assumptions or conclusions of the package analysis, properties or performance of the package materials or components, operation of the package components, design or fabrication techniques, and good practice must be considered in determining the inspections and tests established for the package.

A maintenance requirement can also be established by the use of a purchased component in the package design. Usually, the vendor specifies both the frequency and type of maintenance and testing that are required for the component, and usually the vendor's recommendation should be incorporated into the package maintenance requirements. For example, neutron-poison materials may be tested periodically for neutron blackness (to verify continued absorption of thermal neutrons).

Finally, additional or alternative requirements for the periodic maintenance of the package may be strongly recommended as a result of the regulatory review. The regulator might determine (1) that a proposed design feature poses a failure mode not considered by the designer or (2) that a proposed test is not sufficiently sensitive for the condition being tested. Regulators may also recommend additional inspections or tests or increase their frequency, based on experience (or lack thereof) with a specific design or with a proposed fabrication process. The designer may accept the regulatory position, propose an alternative test procedure, or challenge the recommendation by engineering analysis and reasoned argument.

14.4 PACKAGE, ANCILLARY EQUIPMENT, AND TRANSPORTER MAINTENANCE

This section describes the typical maintenance requirements of the three principal elements of the package system.

14.4.1 Package

The package consists of the assemblage of the package body, closure lid(s), port covers, valves and fittings, fasteners, seals, lift and tiedown attachments, internal and external impact limiters, internal fixtures, and components or devices, including contents basket or separators, and spacers. It is this assemblage that is presented to the regulator as meeting the requirements of the packaging regulations, when used to ship the proposed contents.* The designer must specify in Chapter 1 of the SARP exactly what elements or components constitute the package and therefore what needs to be maintained.

The most common periodic maintenance inspections or tests include:

Inspections—usually include visual inspections of the package surface, and particularly sealing surfaces for defects occurring as the result of use; could also include the inspection of welds by nondestructive means, and dimensional checks.

Pressure testing—typically at 125% of the maximum calculated pressure of the volume being tested. Pressure testing may be performed on the package cavity, neutron shield enclosures, impact limiters, etc.

*Note that the empty or unloaded package does not include the package contents. The designer should also note that *empty* has a legal, regulatory meaning and should not be used in describing the package unless, once unloaded, the package meets the *empty* definition.

Containment verification testing—leak testing, using vacuum or pressure, of the containment boundary in order to show containment.

Scheduled replacements or rebuilding—Particularly of O-rings or seals in the containment boundary, but could include other items, such as venting devices or bolts that have a low-use cycle-to-failure, and of closure or relief valves that must be rebuilt.

Load tests—of package lifting fixtures, fixtures used for lifting the closure lid or other devices.

Of these more common periodic tests, the pressure testing should be carefully considered before it is specified. Its principal and best use is to check operational or functional elements, such as a manually operated valve or a device that is pressure rated, in the package containment boundary. For those packages that do not have these features, a periodic pressure test is not normally necessary. In addition, such tests may not have sufficient sensitivity to check or to verify package leak tightness and, therefore, may not be useful for that purpose.

Other periodic maintenance requirements, if appropriate, could include:

Thermal testing—to verify that the heat rejection of the package continues to be adequate.

Gamma-shield testing—to ensure absence of voids.

Neutron-shield testing—including basket poisons, to ensure continued "blackness" to thermal neutrons.

These latter tests, plus pressure tests, are more frequently performed only after fabrication during the package "acceptance testing." They are performed periodically only if the package designer or regulator has reason to believe that performance of those components could change over time or with the rigor of use. The designer must consider the possible conditions under which mechanical bonds could separate, materials could delaminate, chemical reactions could occur, or voids could form in the package materials. If these conditions cannot occur under evaluated conditions, then tests for them should not be specified.

This list should not be considered a guide for items that should be included in a maintenance program. Those requirements should always be determined by the considered judgment of the designer.

14.4.2 Ancillary Equipment

Ancillary equipment consists of the items that are required to correctly use the package. Such equipment could include lifting gear (sometimes called a lifting yoke), lift fixtures (such as eyebolts and slings), alignment pins, test fixtures and devices (such as helium leak detectors and hand-operated hydrostatic pumps), and any special tools—that is, any required tools that would not likely be found in a mechanic's tool box.

The most common item of ancillary equipment which requires periodic maintenance and per-use inspection is the dedicated lift gear for the package. (*Dedicated* means lift gear that is specifically designed for the package and which typically cannot be used to lift other loads.) Lift gear should

be maintained according to ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More for Nuclear Materials." This standard establishes inspection and maintenance requirements that should be proceduralized by the designer for the benefit of the package users, but is not an element of the package SARP. Usually, any dedicated lift gear should be maintained according to ANSI N14.6, even if the package weighs less than 4,500 kg (10,000 lb). The principal reasons are the wide acceptance that the Standard has among package users and regulators and the document trail that is established by complying with an established national standard.

The principal annual test for dedicated lift gear is a load test. Frequently, the lift gear is load-tested by attaching it to the package in its normal configuration and lifting the package (100% of rated load). Additional load (up to an additional 50% of rated load) is applied to the lift gear by "jacking" between the top of the package and the underside of the lift gear. Using the package as the primary load eliminates the inconvenience of fabricating and using a separate test fixture to mate with the lift gear. The configuration of the hydraulic jack or other device interposed between the package and the lift gear to increase the load must be established by the designer.

Note that the ANSI N14.6 Standard does not require that a load test be performed annually. A fabrication-acceptance load test, followed by an annual nondestructive examination is sufficient to meet the requirements of the standard. However, an annual load test is recommended primarily because it is the preferred "documentation" of most users, who already must perform annual load tests on shackles, slings, cranes, and any other lifting devices in order to meet Occupational Safety and Health Administration (OSHA) requirements.

Following the load test, the lift gear must be functionally tested, visually inspected, and nondestructively examined to verify the absence of defects or to identify evidence of defects in the load-bearing path of the lift gear and, in particular, in any welds that are in the load path. The nondestructive examination method selected must be compatible with the materials under test. Defects found in the load-bearing path of the lift gear is cause for its withdrawal from service until the defect is remedied. The lift gear should also be visually inspected for the following:

- Weld integrity
- Cracks
- Unusual or unexpected wear at bearing surfaces
- Distortion, bending, or other evidence of overload
- Joint fit-up
- Missing components (nuts, bolts, pins, etc.)
- Condition and operation of components
- Locking-device operation

Note that the above items are also candidate elements for a per-use, lift-gear, visual-inspection check list.

The nondestructive examination of welds is somewhat time-consuming and requires that the lift fixture be decontaminated and, for a painted lift gear, stripped of the paint. Bolted connections are easier to inspect, and damaged items, such as bolts, are easier to replace. Heat-treated materials, such as high-strength aluminum, can be damaged by rough handling or by impact energy if the heat-treated piece is struck by another object. Such damage can be difficult to detect and can lead to catastrophic failure of the lift gear under load. Machining, welding, or repairing heat-treated material often requires special equipment and skills that may not be readily available.

Because of these requirements, the designer should avoid the use of welds and heat-treated materials in the load path of the lift gear unless the high strength of heat-treated material is required.

The vast majority of packages do not use dedicated lift gear. They are handled using common wire rope, fiberglass slings and shackles, and forklift drum clamps and occasionally are moved by hand. Where common, nondedicated lift gear is used. The gear is typically maintained under a generic facility program that provides for periodic load testing and for marking or labeling the gear to indicate that the load test is current. This lift gear is considered to be "facility specific," not "package specific." The package owner should not provide generic lift gear to use with the package. This allows the user to rely upon its own systems and records rather than upon those of the package owner. (It also relieves the owner of purchasing, qualifying, documenting, decontaminating, packaging, shipping, and handling common lift equipment.) An exception can be found in the handling of large spent fuel shipping casks. These packages frequently rely on slings of a unique length or configuration for lifting the cavity closure, a shield plug, large covers or the like. Even so, it is recommended that the designer rely on common slings in the package design by specifying the length(s) of sling(s) that the package user should have available.

Other items of ancillary equipment are also usually highly package specific. As the equipment is developed or its need is identified, the designer should consider the maintenance requirements of each item. The maintenance requirement may be no more than a visual inspection, with the inspector disposing of any item that is nonconforming or dysfunctional. Some items, like leak detectors, may contain no user-serviceable parts other than external probes, hoses, and valves. The designer must record the ancillary items that the package user must have in order to handle and load the package. Some of those items will require maintenance; most will likely not. For those that do, the designer should specify the maintenance requirement, at least in the form of a narrative statement.

Procedures for inspection and maintenance of ancillary equipment are sometimes captured in a maintenance manual. A maintenance manual can come about because, in addition to maintenance procedures for items developed by the designer, some "off the shelf" equipment is typically specified, such as pressure or vacuum pumps, leak detectors, slings, and hardware. The possible contents of a maintenance manual are described more fully in Sect. 14.5.

To the greatest extent possible, the designer should eliminate reliance on ancillary equipment and should not specify the use of unique items. The designer should rely on common, standard tools and resist the urge to design and provide special items. There are several reasons for using this approach. Required ancillary equipment must be shipped with the package for the package to be used, which often results in additional shipping costs, equipment packing and unpacking, additional equipment-container handling, and additional radiation-exposure burden for technicians who must handle the equipment in a radiation area and decontaminate the equipment at the conclusion of (and frequently during) its use. In addition, if the equipment is required for package use, then the overall reliability of the package becomes a function of the reliability of the ancillary equipment. Failure of any required ancillary equipment means that the package cannot be used until repairs are made. Finally, the required use of special equipment can complicate the package-handling process. The more complicated the package-handling process, the higher the likelihood for procedural errors to occur.

Before an item of ancillary equipment is specified and before a package is designed so that a particular item is required for proper operation, the designer should determine if that item reduces

the radioactive-exposure burden of the user, reduces the chance for procedural errors, or shortens package-handling time. If it does none of these, the use of the item should be reconsidered.

14.4.3 Transporter

The transporter is the trailer or railcar used to carry the package in the public domain, and for most packages, the transporter is not a dedicated vehicle. The package is prepared for transport by securing it to a pallet, which is then loaded onto a flatbed truck or an enclosed trailer van or a railcar. The package may also be handled individually or secured along with other identical packages inside an overpack container. Nondedicated transporters do not have designated features to use with any particular package, and the designer does not specify the maintenance program for the transporter. The designer (and the later users) rely upon the inspection programs of the carrier that owns or operates the provided transporter. This section focuses on those packages that use a transporter which is dedicated for use with a specific package model; such dedicated transporters are typically used for larger packages [above about 5,000 kg (11,000 lb)].

A dedicated transporter has fixed components that are used to support, secure, tie down, or otherwise restrain the package when the package is in its proper position on the transporter. The tiedowns, supports, or other structures may be addressed in the SARP for the package. (Note that the package designer specifies what constitutes the "package." The package is always defined in the first chapter of the SARP.) Those components may include package-rotation fixtures; a personnel barrier, if a barrier is required by the thermal analysis or other considerations; and perhaps the method of attaching package supports to the transporter frame or deck.

Usually, none of the other transporter features, such as the frame, running gear, brake lines, etc., are analyzed in the SARP. Nevertheless, the transporter requires both periodic maintenance and per-use inspections.

Railcar and trailer maintenance is typically performed by qualified personnel who usually work for external organizations. Such personnel might work for a railroad company or an over-the-road transport or service company which is qualified to perform inspections on the equipment. The package user often pays for this inspection service. Rail carriers, in particular, do not accept the results of third-party railcar inspections before a railcar is accepted for the transport of radioactive materials. Loaded railcars are typically moved by the originating carrier to the first available switchyard for inspection.

14.4.3.1 Railcar Maintenance

Railcars are used to move packages that weigh above about 32,000 kg (70,000 lb) and to move accumulations of smaller packages that aggregate about the same, or higher, weight. Railcar maintenance is usually performed by a railroad that is a member of the Association for American Railroads (AAR). No specific periodic maintenance is required for the railcar or its running gear. The interchange inspection (an inspection performed at the first rail yard after pickup of the railcar) suffices to detect such things as problems in wheels, brake shoes, couplers, air-brake hoses, and glad-hands and problems with safety appliances such as hand holds. An annual inspection of the rolling gear, brakes, and safety devices is completed at a railroad-operated facility.

The entire air-braking system of a railcar must be tested and inspected every 10 years (120 months). AAR rule 88.1.d requires railcars carrying radioactive materials (whether dedicated or not) to be equipped with draft-and-buff cushion couplers. These couplers incorporate hardware that resists uncoupling during normal transport and minor accident events that result in coupler impacts or misalignment between connected railcars and normally require periodic lubrication. No other periodic lubrication is required for the railcar.

The railcar deck or frame attachment points must be inspected periodically to verify that they will still meet both NRC and AAR package retention requirements. This inspection can be done by using nondestructive examination techniques.

Main structural members of railcars include the center sills and side sills. These sills should be visually inspected for damage or other defects before each shipment by the package user.

14.4.3.2 Truck Trailer Maintenance

The fixed components of the truck trailer that have been designed as part of the package are always maintained according to the requirements specified in Sect. 8.2 of the SARP. This section provides some guidance on the periodic maintenance requirements that can be applied to the reminder of the trailer.

There is somewhat greater flexibility in the performance of inspections of highway transport trailers than there is for railcars in the sense that required inspections can be performed at any number of highway truck stops and garages. Some inspections and minor repairs (e.g., running-light replacement, brake adjustments, etc.) can be, and typically are, made by the motor carrier driver.

The components of the trailer that should be inspected by a qualified person (who could include the driver) can be placed in the following groups:

- Trailer frame and weldments
- Tires and wheels
- Brakes and adjustors
- Lights and wiring
- Hose and connections
- Running gear
- Landing gear
- Lubrication

Each of these elements is discussed briefly below. The package owner should use diagrams and checklists to facilitate the inspections and to help ensure that the inspections are correctly completed.

Trailer frame and weldments. Inspect main support members and lateral (cross) members for evidence of cracks, fatigue, and unusual wear along the length of the member, and at dimension transition points. Inspect weld locations at attachment of lateral members to the main support members, and at the king pin and king pin mounting plate (sometimes called a "fifth wheel." Inspect mounting welds for appurtenances such as spare-tire racks, placard holders, lights and lines, mud flats, running-gear equipment, air tanks, fender mounts, etc.

Tires and wheels. Inspect for irregular wear, air pressure, valve caps and stem locks, and remaining tread depth. Tread depth cannot be less than 2.4 mm (3/32 in.). Inspect wheels for cracks, curb damage, and loose or missing lugs. Record mileage if hub odometers are used. (Hub odometers are used to record trailer mileage because any number of tractors can typically be used to tow the trailer.) Verify that the dual wheels are aligned properly.

Brakes and adjustors. Inspect hoses and fittings for leaks, wear, and chaffing. Verify air pressure as well as brake operation on loss of pressure. Inspect brake drums and shoes. Verify that no screws or rivets penetrate the brake lining. Inspect antilock brake and air-ride systems. Verify brake shoe clearance and adjust clearance as necessary.

Lights and wiring. Inspect all lights and verify operation; replace cracked or missing lenses. Inspect for loose or frayed wiring and for loose connections. Inspect plug connector to tractor.

Hose and connections. Inspect air lines for slow leaks, fraying, or chaffing and for loose attachment points. Inspect glad-hand connectors and replace glad-hand seals. Inspect interconnections of air lines at antilock brake and air-ride devices.

Running gear. Inspect king pin and fifth-wheel plate for wear. Verify proper operation of the trailer relay valve. Verify that the air tank is drained of water. Inspect mud flaps, license plate, and operating stickers for damage. Verify that the bottom edge of the mud flap is the correct height above the road surface. Inspect springs and U-bolts for damage. Inspect splash pan for attachment, rust, and cracks. Inspect remaining cross members and under-structure, catwalks and fenders for damage.

Landing gear. Inspect leg braces, supports, and landing wheels or pads and pad swivels for damage. Inspect crank handle and handle-retaining clip for damage. Verify that the gear box is filled with gear oil to the correct level.

Lubrication. Check for loose and noisy bearings. Verify that the lubrication reservoir is full with the correct lubricant. Check for proper oil level at wheels.

In each case, items that are inspected or checked must be repaired or replaced if the item is noncompliant to inspection procedure requirements or to regulatory requirements as appropriate.

The actual periodic inspections performed depend on the design of the transport trailer. The cask owner should ensure that the inspections specified result in an adequate and satisfactory check of the trailer integrity.

Finally, many of the items inspected at the time of the periodic inspection are candidates for inspection on a per-use basis. In addition to verifying that the load is secure, some items must be checked by the driver before departure according to DOT requirements. Shipments that have a high local visibility may also be checked by a state inspector before departure. Currently, predeparture inspection checklists are being developed by the Commercial Safety Vehicle Alliance (CVSA) for use with planned truck shipments of transuranic (TRU) waste shipments from the various DOE sites to the Waste Isolation Pilot Plant (WIPP) facility near Carlsbad, New Mexico. A goal of the CVSA inspection program is to eliminate detailed inspections of the transport equipment in each state that the shipment passes through. It is a reasonable expectation that if this inspection program proves successful, it would be applied to future shipments of high-level radioactive wastes, particularly to those made for U.S. government agencies.

14.5 MAINTENANCE MANUAL

A maintenance manual is frequently developed for the package system, especially if more than one package is to be in service and if there are multiple users. The manual usually describes maintenance requirements of the package and items or components that are not part of the certificated system, such as maintenance and testing of the lifting yoke; the operation of a special item of equipment used with the package, such as a vacuum pump; and requirements for convenience and appearance items. It also provides ordering information for off-the-shelf parts used in the package ancillary equipment and for those parts needed to "interface" the package with the facility. These parts include quick disconnects, slings, shackles, eyebolts, hoses, etc. The information included usually consists of vendor catalog pages or copies of special-equipment operating manuals published by the equipment manufacturer. Providing the ordering information (descriptions, part numbers, and vendors) simplifies procurement of parts either for use or as spares and assists in ensuring that the correct parts are obtained for use. Drawings, figures, exploded views, tables, and diagrams should be used as needed to supplement the narrative steps.

The manual also should contain (1) maintenance information for the package transporter, and (2) copies of the checklists to be completed for each shipment and for other periodic inspections.

The designer should specify the periodic maintenance procedures in Sect. 8.2 of the SARP. These procedures will be reviewed and approved by the regulator. The procedures must be in sufficient detail to permit the regulator to follow the procedural process, understand the intent, and reach a conclusion about the adequacy of the procedure. If the maintenance procedures for the package are provided in the manual, the regulator will include the manual by reference in the package certificate. A disadvantage of this procedure is that changes that are made to any part of the manual could require an amendment to the package certificate. Package maintenance procedures included in a manual are (or should be) much more detailed than those presented in Sect. 8.2 of the SARP, providing information on (1) how to perform an inspection or test on the specific package, (2) increased frequency of maintenance, etc.

An advantage of a maintenance manual is that it provides a place to document procedures for the package, transporter, or ancillary equipment that are developed, but little used. Examples of these kinds of procedures include thread insert replacement, repair of components that are usually replaced, surface preparation for painting, decontamination procedures, special tool repair, repair of damaged surfaces or damaged components, etc.

The package designer or package owner has some flexibility in the form and content of what is included in (or excluded from) the manual because it is not intended for routine use by the package registered users.

14.6 UNSCHEDULED MAINTENANCE AND REPAIR

Thus far, the discussion has centered on scheduled, periodic maintenance, as described in Sect. 8.2 of the SARP. It is important to recognize that a maintenance requirement can arise at any time during use of the package. Damage or defects can be found at any point of package handling or use that precludes the package from meeting its performance standards as specified in the package SARP.

For the purpose of this discussion, this unscheduled maintenance (with a few exceptions described in the following) is called *repair*. Three important elements must be considered in the performance of repair.

The first is that the completed repair must return the package to the condition described by the SARP and established by the package certificate. This means that the repair must be made using the criteria for procedures, materials, and qualifications as were applied to the original fabrication of the package.

The second is that careful documentation must be kept of the repair so that it can be demonstrated from the records that the completed repair was performed according to an approved procedure and using correct materials and was carried out by qualified personnel. If the records do not reflect the correct repair, then the repaired package is unacceptable for use regardless of the quality of the repair. The records of the repair are part of the package permanent documentation.

Third, a damaged package cannot generally be moved to another location until the repair is completed. The reason for this is that for the package to move in the public domain, it must be in conformance with its certificate, and a damaged package is not in conformance. Consequently, damaged or unserviceable packages must often be repaired at the location where the damage is found, which can sometimes be inconvenient. In some cases, it may be possible to obtain an exemption to the certificate from the U.S. Department of Transportation (DOT) (49 CFR Part 107, Subpart B). The request for exemption must be based on the implementation of compensatory measures to ameliorate the consequences of moving the damaged package. The compensatory measure(s) must provide a level of safety equivalent to that presented by the uncompromised package feature.

An example of a compensatory measure applied to the shipment of a spent fuel cask that at a user's facility cannot be sealed (e.g., because of catastrophic damage to a seal surface) would be to remove the spent fuel and basket and decontaminate the inside and outside of the cask to DOT transport limits. This measure means that in normal and accident conditions of transport, there would be no releases of radioactive material from the cask that exceed regulatory limits. Even though this compensatory measure sounds somewhat simple to implement, in fact, it is very difficult to decontaminate the interior cavity of a spent fuel shipping cask because of accumulated residual contamination that has a very high radiation level. Often, it is easier to repair the damage at the location where it is found than it is to move a package through the exemption process.

A complication of the repair activity at a user facility is the need to have a procedure for the repair approved by the facility management. The procedure approval process can be somewhat time-consuming because the facility personnel must understand the nature of the repair and risks to the facility and its personnel during repair.

The regulatory agency must be notified of any significant defect found during an inspection or identified during use. It is up to the package designer to determine what constitutes a significant defect. Certainly, defects such as structural failures of base material or welds; loss of a thermal, shielding, or criticality-control capability; puncture of an outer or inner shell material of the package or impact limiter; or similar fault should be reported. Lesser defects should be repaired, and the repair should be documented.

As noted previously, there are some minor exceptions to a repair requirement when a package component fails in service. The exceptions relate to items that fail, but whose failure should not

be unanticipated by the package designer. These include such items as quick-disconnect and other types of fittings and thread inserts. Quick-disconnects can become inoperable because of debris that lodges inside the disconnect and prevents its normal operation. A threaded insert can "back out" of its seat during bolt or screw removal and break in the process. Even though correcting each is technically a repair to the package, it is more appropriately considered maintenance that is incidental to the use of the package. The replacement of these items should be described in appropriate procedures (which may be no more than the manufacturer's instructions in the case of threaded inserts) that provide documentation of the work done. Either item can be considered to be within the skill level of persons reasonably expected to handle a package.

14.7 RECORDS

Completed maintenance documentation is part of the permanent record of the package regardless of whether the maintenance was scheduled or unscheduled. Maintenance records for multiple packages of the same model must be maintained by package serial number.

The maintenance records must serve three purposes: (1) to record that maintenance has been performed as required by the package certificate and SARP; (2) to record the satisfactory and acceptable completion of the inspection or test; (3) to provide an audit trail that can stand reasonable tests regarding the satisfactory performance of past maintenance. Failure to have or to maintain these records, or to maintain records that do not show the correct performance of maintenance could result in a package's being removed from service and the imposition of civil penalty as provided for by the regulations.

There are several aspects of acceptable maintenance records: (1) an approved procedure that specifies the maintenance requirements; (2) each step of the maintenance procedure is signed by a qualified technician as it is completed; (3) steps involving devices (such as pressure gauges and leak detectors) are signed by a second qualified person, typically a QA/quality control inspector; and (4) serial numbers and calibration dates are recorded for tools, equipment, and devices used. The procedure used to document maintenance should incorporate space to record the required signatures or other data. If equipment is used that generates a permanent record of a test or inspection, then that record must be signed, dated, and retained as part of the maintenance documentation.

Because maintenance tasks must be performed by persons who have the requisite skills and training to perform these activities, the completed maintenance procedure must be supported by the training and qualification records of the persons performing the work. Even though these records need not accompany or be appended to the completed maintenance procedure document, they must be available for audit. Also note that some other types of records must be available for audit, including calibration records and procurement and shelf-life records for installed parts. Training, calibration, and procurement records may be resident in other parts of the organization. However, these records, and the maintenance procedure records, must be available for review and audit by users of the package and by the regulatory agency. Needless to say, this implies that the completed and signed maintenance procedure must contain information sufficient to allow the recovery of the original supporting records.

In addition to these records, procurement and stores records must be available that show that parts used on the package during maintenance conform to the package requirements. This rule means that purchase requisitions must show that the parts ordered meet the specifications established by

the package drawings or other documents and that the vendor supplying the respective part also supplies the documentation needed to show conformance of the part to the requisition. The documentation could consist of heat-treat information, chemical analysis, sample test results, or affirmations that the part delivered conforms to the requisition. These records are also subject to review and audit by others.

Finally, the records that are required to demonstrate conformance to the package maintenance requirements should be generated by the application of the QA program that applies to the package.

Similar records are required for ancillary equipment that is designated as "safety related," such as a package lift fixture, and for the package tiedown devices that are not an integral part of the package. The tiedown devices may consist of a cradle, or rotation frame, that supports the package on the bed of the transporter. These components are normally inspected and tested at the same time as the package; information from the inspection, and results from the nondestructive load test should be retained for at least 3 years. As previously noted, these components are not part of the certified system; however, it is standard practice to apply the package QA program requirements to these components.

Maintenance of the transporter is not generally performed by the package owner, even if the transporter (truck trailer or railcar) is a dedicated vehicle. The principal reason is that the package owner is usually not (also) in the transportation business. The carriage of the package is usually provided by a commercial carrier, and the carrier typically provides the expertise required to conduct a qualified inspection of the transporter. This situation is especially true for railcars because the carrier often insists on conducting both periodic and per-use inspections. There are a few exceptions, especially among low-level-waste transporters, who provide the package, site services, truck transport, and disposal services. To the greatest extent possible, the package owner may wish to obtain and retain records of inspections and tests of the package transporter for completeness. However, these records may not be routinely available.

Appendix A: A Survey of Materials and Procedures for Design of Impact Limiters in Radioactive Material Transport

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A.1 INTRODUCTION

The objective of this appendix is to provide information for the design of impact limiters that may be used to protect nuclear shipping casks from a postulated free fall of 9 m (30 ft) onto a horizontal, unyielding surface. Various analytical methods applicable to the design of impact absorbers were studied to identify the most appropriate methods for particular materials. Design procedures, ranging from volumetric energy absorption allowing manual calculation to nonlinear finite-element analysis using a computer, were considered. Each of the different procedures requires specific characteristics of the material. Material property data, which are necessary to carry out the design methods, were gathered from a variety of sources and evaluated for utility in the design process.

The 9-m (30-ft) free fall and subsequent impact are together the first of a sequence of accident conditions imposed on the design of nuclear material shipping casks by federal regulation.[1] Other tests in the sequence representing accident conditions also influence the choice of materials for impact limiters. The cask must strike the surface in the position expected to produce maximum damage; but because it is usually difficult to predict conclusively the critical position, the cask is dropped at least three times: on its end, on its side, and on a corner directly below the center of mass. A single test cask may be used for all drop tests by orienting it so that relatively undamaged portions are exposed during each drop. The remaining sequence of tests are performed on the same cask or casks that have been dropped.

Although the impact limiter may undergo extensive deformation and may be consumed in fire during the accident test sequence, the limiter must be designed to tolerate a specified operating environment, one that includes water spray and free drops of between 0.3 and 1 m (1 and 4 ft) (depending on cask weight), without consequential damage. Experience indicates that only minor provisions are required to meet operating conditions; therefore, only a brief mention of the relative

advantages or disadvantages of the various materials in the operating environment is given in this appendix.

This appendix extends the collection of impact limiter design concepts found in two previous reports. The *Cask Designer's Guide,*[2] published at Oak Ridge National Laboratory (ORNL), presents background information on design requirements and recommended design practice. R. E. Berry et al.[3] report on the development of criteria for selecting energy absorber concepts and materials, and they evaluate a large number of options. The contents of several papers and documents that present particular aspects of impact limiter design are referenced at the appropriate points.

This appendix consists of five technical sections and the references. Section 1, the introduction, discusses general analytical methods. The remaining sections each are concerned with a specific class of materials. Section 2 discusses foamed materials: polymer, ceramic, or metal. Section 3 discusses wood impact absorbers. Honeycomb absorbers, which are constructed of either aluminum or stainless steel, are covered in Sect. 4. A wide variety of configurations of metallic energy absorbers including fins, tubes, and frames, are discussed in Sect. 5.

Each of the sections on impact-absorbing materials includes the following topics: material characteristics, examples of successful use, sources for design data, and specific design procedures. Material characteristics included are stress-strain curves, strain rate sensitivity, and environmental sensitivity. Shipping casks that have received U.S. Nuclear Regulatory Commission (NRC) certification are indicated as examples of successful use. Appropriate design data and design technique are closely related. The data required for design are contingent on the selection of a design technique. A conservative sizing calculation does not require as complete a set of material property data as does a dynamic, nonlinear, finite-element model. Conversely, if dynamic material properties are not available, it is not practical to use a sophisticated dynamic analysis.

A.1.1 Methods of Analyzing Impact Limiters

The type of impact limiter chosen dictates the range of possible analytical methods and the complexity of analysis. When the shipping package can be separated into a containment part and an impact-limiter part and when these two parts are structurally dissimilar (e.g., when the impact material is less stiff and less dense than the containments), then many analytical techniques are valid, and this variety can be exploited to control the cost of analysis. If parts of a shipping package are structurally dissimilar, coupling between the parts is simplified, and the parts can be analyzed separately and then recombined only through interface forces. Typically, the separate analyses are less complex than those analyzing the complete structure. In the case of an impact limiter that is structurally similar to the containment, an analyst may be limited to modeling the combined structure or determining response experimentally.

Complexity of analysis also varies with the spatial dimensions necessary to describe the geometry of the impact limiter as it undergoes deformation. The simplest deformation state is uniform crush over a volume which corresponds to a zero-dimension state. A one-dimensional (1-D) solution occurs when the cross-sectional area of the impact limiter varies as it crushes in a predictable way. Two-dimensional (2-D) solutions are necessary for cases in which a 2-D stress state is important, such as one in which a shear stress limits the impact-limiter material involved in crushing. A full three-dimensional (3-D) analysis is required for material with a nontrivial shear strength, such as

steel, impacting in a skew orientation. The general analytical techniques discussed in the next sections are ordered, respectively, by the spatial dimension of the solution.

A.1.1.1 Uniform Crush Analysis

Some impact-limiter materials can be analyzed accurately by assuming uniform crush of the material. Manufacturers' literature recommends this simple analysis for the sizing of package protection systems,[4,5] but the assumptions used are not explicit. The impact-limiter materials are assumed to be ideally massless and insensitive to strain rate. Materials such as honeycomb, balsa, and redwood exhibit these characteristics. In these relatively low-mass materials, inertial forces and dynamic behavior can be ignored. A nominally static analysis of the impact-limiter material can then be performed.

The boundary conditions on the impact limiter must not change during a uniform crush. Surfaces which start out stress-free must remain so throughout, and other surfaces that conform to a prescribed shape, such as a cask end or the impact plane, must retain the original shape. These conditions hold true for an impact limiter that conforms to the shape of the end of a package on one side and opposite that side is flat against the ground, as shown in Fig. A.1. At every instant of time, the stress in the impact limiter must be uniform or representable as uniform. For high-shear-strength materials, uniform stress occurs when the impact limiter has a constant cross section in the direction of crush. Difficulty arises with high-shear-strength absorbers that are impacted over a small area at one end while supported over a larger area on the opposite surface. The compressive stress decreases away from the small contact area as the effective cross-section area increases. Materials with very low shear strength relative to compressive strength, such as balsa

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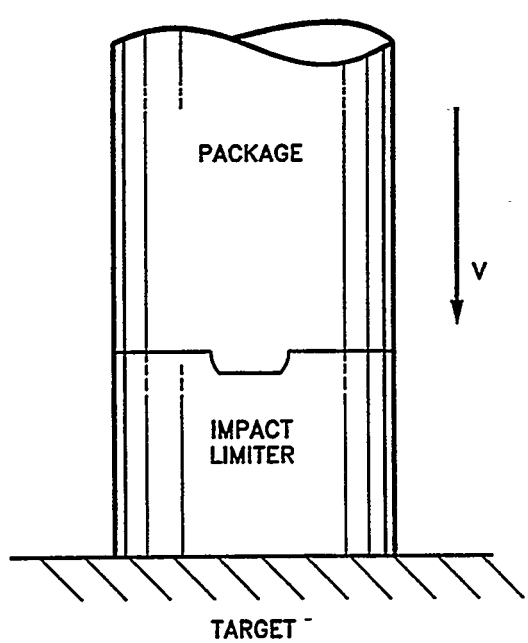

Fig. A.1. Geometry appropriate for a uniform crush analysis.

and some rigid foams, crush as though only the volume of material under the smaller area of contact is effective. Some low-shear-strength materials require the lateral restraint that surrounding material provides so that shear failure does not occur inside the crush volume and then cause chunks of material to fly out during an impact.

An analysis for the case of uniform crush is presented here for an impact-limiter material with the strictly increasing stress-strain curve (Fig. A.2). The analysis begins at the time the impact limiter contacts the ground. The assumption of zero mass for the limiter allows the limiter portion of the package to stop instantaneously. Classically, time would be incremented, but here velocity, acceleration, strain, or energy may be used, decrementing or incrementing as appropriate. Step size must be chosen to allow an accurate integration of the stress-strain curve, and the form of the approximation to the curve will guide that choice. Strain is the best independent variable for analysis without using computers. At the beginning of an impact cycle, the kinetic energy, KE_1 of the cask is

$$KE_1 = 1/2\ m\ V^2\ , \tag{A.1}$$

where

m = cask mass, kg,
V = cask velocity, m/s.

During the cycle, the center of mass of the cask moves downward by a distance h (in meters). The work done by gravity over a cycle is

$$W_g = mgh\ . \tag{A.2}$$

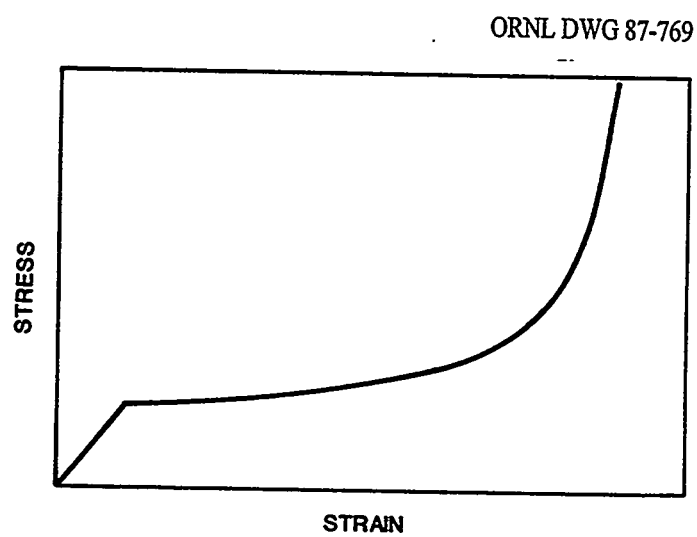

Fig. A.2. Strictly increasing stress-strain curve.

where

g = acceleration of gravity, 9.807 m/s^2 .

With a constant area, A (in square meters), the force developed in the impact limiter retarding the cask is

$$F = \sigma(\varepsilon)A \ , \tag{A.3}$$

with stress, $\sigma(\varepsilon)$, in Pascals, expressed as a function of strain, ε, assuming that strain rate is not important. The strain can also be written

$$\varepsilon = h/L \ , \tag{A.4}$$

where L is the original length of the impact limiter.

During a calculation cycle the work performed in crushing the impact limiter is

$$W_c = -\int F(h)dh \ , \tag{A.5}$$

which can be written as

$$W_c = AL\int \sigma(\varepsilon)d\varepsilon \ .$$

The kinetic energy at the end of the cycle is written

$$1/2\ mV_2^2 = 1/2\ mV_1^2 + mgh - AL \int \sigma(\varepsilon)d\varepsilon \ . \tag{A.6}$$

If $\sigma(\varepsilon)$ is analytically integrable, the equation can be solved for the crush distance, h, to stop the cask with $V_2 = 0$. Some readily integrable stress-strain functions to deal with are straight-line segments, polynomials, or combinations of the two.

Materials with stress-strain curves that have a flat center, as shown in Fig. A.3, are also straightforward to integrate when the stress is uniform throughout the impact limiter. A yield stress, σ_y, and strain, ε_y, and a lockup strain, ε_l, where the stress resumes increasing, are evident from the figure. These materials do not produce uniform strain throughout the limiter when loaded, but they undergo a predictable pattern of deformation which simplifies analysis. In this case, as the load on the limiter is increased, the stress is uniform until the weakest section yields, or alternatively the section of highest load, if the fiction of zero mass can be ignored briefly, yields. This section strains as the work done by the load increases, but the load cannot increase until lockup strain is reached. Then at a slightly higher load, the next weakest section yields and so forth until the entire impact limiter has reached lockup strain and a uniform condition prevails. In this material, between first yield and lock-up of all of the material, the stress is essentially constant and assumed to be independent of strain rate.

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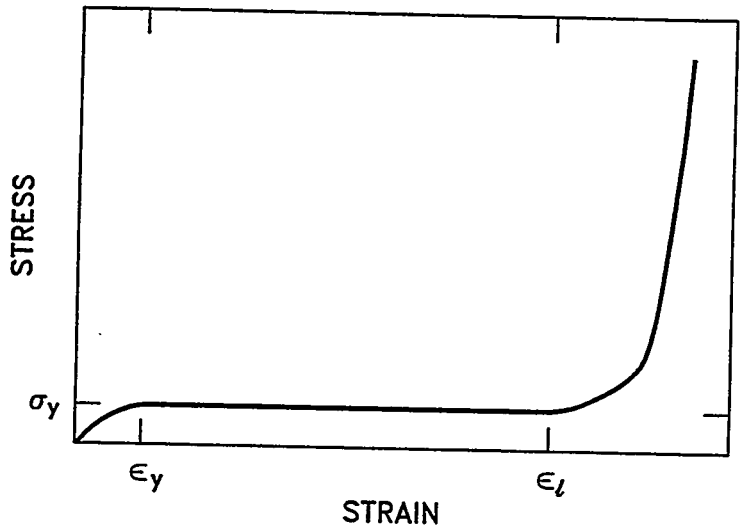

Fig. A.3. Flat-center stress-strain curve.

An impact limiter of this material produces a constant and controllable deceleration rate on the containment portion of a cask in a drop test after yield stress is reached and before the entire limiter locks up. Up to the point of yield,

$$1/2\ mV_2^2 = 1/2\ mV_1^2 + mg\varepsilon_y L - 1/2\ AL(\sigma_y)(\varepsilon_y)\ , \tag{A.7}$$

where $\varepsilon_y L$ is the change in length of the impact limiter up to yield. The square of the velocity at the yield point is

$$V_2^2 = V_1^2 + 2gh_1 - AL(\sigma_y)(\varepsilon_y)/m\ . \tag{A.8}$$

Then the constant force on the containment by the limiter and by gravity are

$$F = \sigma_y A - mg\ . \tag{A.9}$$

Using Newton's second law of motion, we find that the acceleration of the cask, a, is

$$F = ma = \sigma_y A - mg\ . \tag{A.10}$$

Solving for a,

$$a = \sigma_y A/m - g\ . \tag{A.11}$$

The change in velocity of the containment part of the cask during the constant acceleration is

$$V_3^2 - V_2^2 = 2(-ah_2) \ , \tag{A.12}$$

or

$$V_3^2 - V_2^2 = 2(g - \sigma_y A/m)h_2 \ , \tag{A.13}$$

where

V_3 = final velocity, m/s,
h_2 = crush since yield, m.

Substituting the velocity at yield for V_2^2 and setting V_3^2 to zero yields

$$-V_1 - 2g\varepsilon_y L + AL(\sigma_y)(\varepsilon_y)/m = 2(g - \sigma_y A/m)h_2 \ , \tag{A.14}$$

and

$$h_2 = [1/2\ V_1^2 + g\varepsilon_y L - 1/2\ AL(\sigma_y)(\varepsilon_y)/m]\ (\sigma_y A/m - g) \ , \tag{A.15}$$

to stop the cask. The total crush to stop the cask is $\varepsilon_y L + h_2$. If the impact limiter locks up before the cask is stopped, the force and acceleration on the cask will increase.

A.1.1.2 Linear Deformation Analysis

A variation in the cross-sectional area of the impact limiter along the direction of crush produces a condition of nonuniform stress as well as nonuniform strain that can be analyzed by breaking the limiter into elements of constant area. The number of elements to use depends on the rate of change of area. Many divisions are needed when area changes rapidly during the crush, but only one element is needed for a length of constant area. The adequacy of a model can be conclusively demonstrated by showing that going from a large number of elements to a larger number produces only a trivial change in the calculated answer.

Axial force is constant throughout a massless impact limiter, but the stress $\sigma(y_i)$ will vary according to the relationship,

$$\sigma(y_i) = F/A(y_i) \ , \tag{A.16}$$

where

F = axial force at any instant of time, and
$A(y_i)$ = cross-sectional area expressed as a function of y_i, the axial coordinate for the limiter.

With the ideal massless, strain-rate-independent materials, the strain can be read from the stress-strain curve for each element of material. For materials with stress-strain relationships resembling Fig. A.3, it is clear that elements of relatively small cross-sectional areas will experience strains

beyond the lockup strain to produce the same force as found in larger elements. The overall crush thickness, e, of an element of original thickness L and total engineering strain of ε is

$$e = \varepsilon L \ . \tag{A.17}$$

The total crush of the impact limiter is found by adding the crush of each element. Similarly, the total internal energy dissipated by the crush of the impact limiter is found by summing the energy in each element which is calculated by integrating under the stress-strain curve up to the maximum strain in each element.

Manual calculation is difficult because of the large number of integrals to be evaluated, but a computer program can be developed to carry out all of the calculations. The force in the impact limiter can be used for the independent variable and incremented. Choosing an increment size is difficult because elements with a small cross-sectional area develop high strains with small forces, but these elements contribute very little to the energy absorbed. In analyses of this type, the choice of increment must be tested by demonstrating insensitivity to a reduction in step size. However, this procedure is valid only when the original increment is small. The force is assumed to be constant and the stresses, strains, internal energy, and crush are determined over the calculation cycle. The change in kinetic energy over each cycle is

$$1/2\ mV_2^2 - 1/2\ mV_1^2 = mgH - E \ , \tag{A.18}$$

where

V_2 = final velocity of cask over cycle, m/s;
V_1 = initial velocity of cask over cycle, m/s;
H = total crush of limiter over cycle, m;
E = total internal energy dissipated, J.

At the end of each cycle the velocity is tested to see if the cask has stopped; if not, the force should be incremented, and the calculations should be redone. It is convenient to design the computer program so that the number of elements and the force increment can be easily changed to allow verification of the results.

The linear deformation analysis can be used for the side or corner impact so long as the absorbing material is essentially isotropic and has a low shear strength relative to its compressive strength. If the shear strength is low, the cask will punch into the limiter material. Then only the volume under the cask will absorb energy. This volume is called the *effective volume*. Effective volumes for the side and corner drops are shown in Fig. A.4.

When a falling cask with an impact limiter first contacts a target, the area of contact between the limiter and the ground is generally much smaller than the area between the cask and the limiter. Stress on the impact limiter is higher over the smaller area, and crushing will start at the point of contact. Because shear stresses are ignored, only the effective volume of the limiter is considered. The area of contact of the material being crushed increases until either the cask stops or the contact area is equal to the horizontal cross section of the effective volume. Once the entire horizontal area of the effective volume is engaged, the crush area becomes constant and a uniform crush analysis can be applied. To determine the energy absorbed in the limiter before the entire

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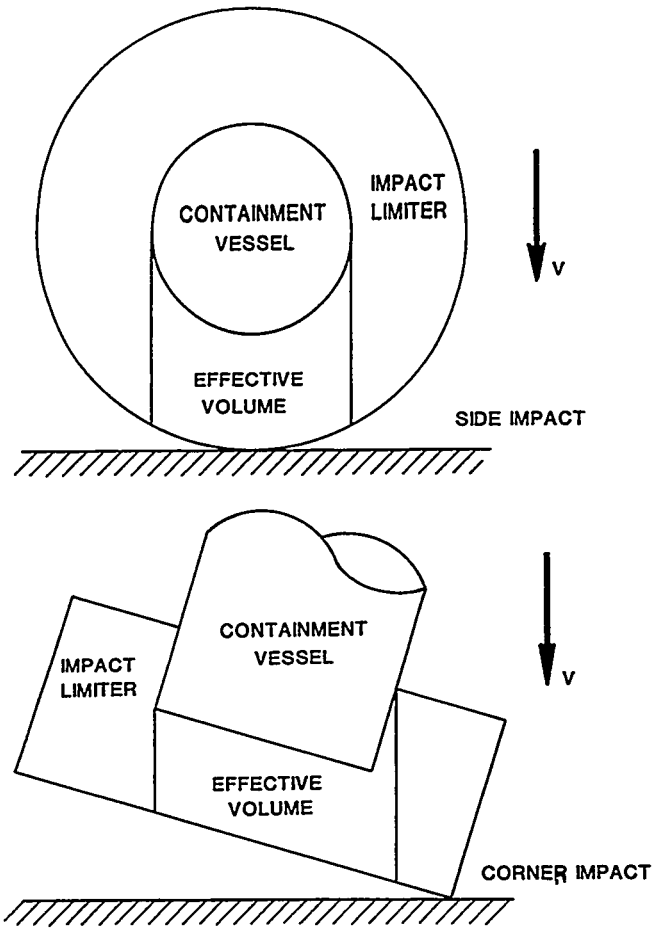

Fig. A.4. Effective impact-limiter volumes in side and corner impacts.

effective volume is engaged, an incremental approach must be employed. C. R. Adams et al.[6] discuss the geometric change in crush area during the side and corner drops. N. M. Meinert and A. H. Wells[7] and D. J. Nolan et al.[8] each describe a dynamic analysis of a cask, including a calculation scheme similar to that outlined here for determining energy absorbed by impact limiters.

A.1.1.3 Two-Dimensional Analysis

A number of complications can invalidate one-dimensional (1-D) solutions, but two-dimensional (2-D), dynamic, structural analysis programs are available to provide results. Two surveys of computer programs suitable for dynamic analysis of a nonlinear material with a thinner cladding like the cover over typical foam, wood, or honeycomb impact limiters have evaluated five programs: ABAQUS™, ANSYS™, HONDO, MARC™, and PISCES.[9,10] One of these programs, HONDO (now in version III), contains a material subroutine to model crushable foam behavior as well as elastic-plastic behavior.[11] This group of programs is representative of a larger group that

is capable of analyzing ductile-metal behavior. Two-dimensional treatment of shipping cask impact is appropriate for the end drop, which is axisymmetric; the side drop, which is approximately planar; and a plane-strain deformation of a single ductile fin. A corner drop or the deformation of a group of unaligned fins, however, requires a three-dimensional (3-D) analysis.

The material properties required for each program or material subroutine differ, but in general they require an elastic modulus, a yield stress, a plastic modulus, Poisson's ratio, a hardening parameter, and density. The special foam model in HONDO III uses bulk pressure versus dilatation, a bulk-unloading modulus, three yield-function constants, an elastic shear modulus, and the pressure cutoff for tensile fracture. Although the bulk data are rare in the literature, bulk properties can be derived from uniaxial test results.

A.1.1.4 Three-Dimensional Analysis

Some of the computer programs listed in Sect. A.1.1.3 can model stress and strain behavior in three dimensions. The elastic-plastic material subroutines in those programs are identical for 2- or 3-D analyses, and the same material properties are required. G. W. Wellman[12] discusses 3-D analysis of a cask design and identifies one three-dimensional analysis program, DYNA3D, which contains the crushable foam model that appears in HONDO III.

A.2 FOAMED MATERIALS

Three categories of foamed materials are considered in this section: flexible foams, rigid polymer (plastic) or metal foams, and brittle foams. All of these materials behave similarly under load, and all can be analyzed using the same methods. The rigid polymer foams are the most commonly used for impact protection, and considerably more information is available describing their impact properties.

A.2.1 General Material Characteristics

Among the materials used for absorbing energy during impact and thereby limiting the accelerations experienced by a shipping cask are flexible, rigid, and brittle foams. Foams have characteristics that make them attractive choices for impact-limiter applications. Foams can be manufactured so that they are essentially isotropic and as such do not necessitate consideration of the direction in which the impact load is applied. When used to protect a cask in a drop test, the effect of an energy absorber will be to lengthen the duration of the acceleration and reduce the acceleration experienced by the cask in contrast to the acceleration experienced by a cask with no limiter attached. The reduction in the magnitude of the acceleration will in turn reduce the maximum inertia loads on the components of the cask and its contents. 10 CFR Part 71 requires that the orientation of a cask during the 9-m (30-ft) drop test be such as to cause maximum damage. In an accident, the orientation of the cask will be unknown. Consequently, the energy-absorbing structure used must be able to withstand the application of an impact load from any direction. The isotropy of foam materials in part meets that requirement.

Foams can absorb energy and limit the transfer of force to the shipping cask to which it is connected. The energy absorbed during impact is equal to the work done by the impact force as

it crushes the foam material. The work done by the impact force is determined (1) by computing the area under the plot of impact force versus displacement or (2) by integrating the specific energy absorption in the foam over the volume of material which is active during the impact process. The ideal form for the force-versus-deflection curve is a flat rectangle, indicating a constant and minimum force throughout the impact process. Most foams, when loaded in compression, display a similar stress-strain curve having a small initial elastic region up to about 5% strain terminated by yielding.[13] This is followed by a region of relatively constant stress with increasing strain. This region extends to strains of 50 to 70%, depending on the material density, after which the foam locks up and the stress rises rapidly with increasing strain.

A typical stress-strain curve for a polymethacrylimid foam is shown in Fig. A.5. The relatively flat portion of the curve extending from about 5 to 60% strain exhibits the characteristic of constant stress over a large strain range. The average stress value corresponding to this "plateau" region increases directly with the material effective density. However, the higher the plateau stress, the more narrow the plateau strain range. The rapid rise of stress for large strain beyond the plateau corresponds to the lockup region, which is encountered when the foam cells have collapsed.

The tensile strengths of foams are relatively constant in the temperature range –45 to 52°C (–50 to 125°F).[14] The tensile strength is near its maximum at room temperature and decreases for higher and lower temperatures. The elongation departs from its room temperature value more rapidly with change in temperature, rising rapidly above room temperature. The compressive strength has its maximum at low temperature and decreases continuously for higher temperatures. For example, the compressive strength of a low-density foam can vary from 120 to 60% of the room-temperature value over a –185 to 150°C (–300 to 300°F) range.

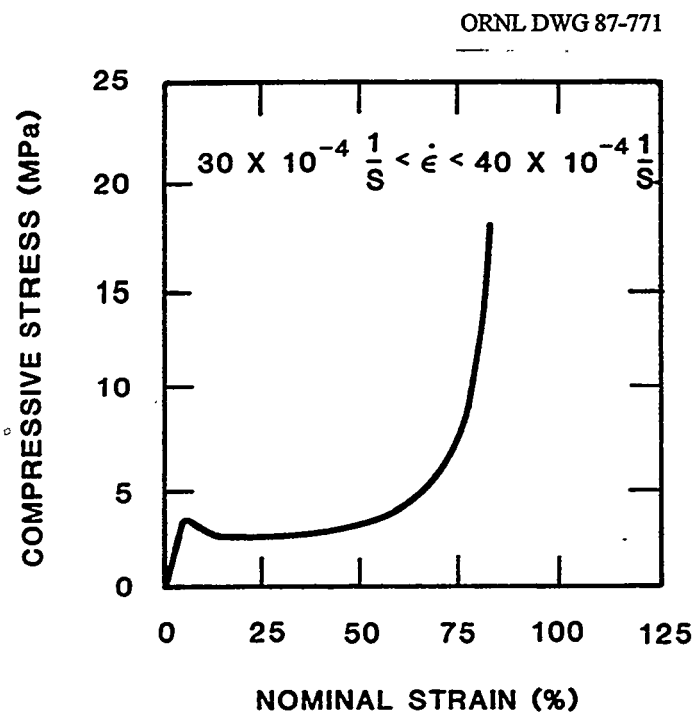

Fig. A.5. Stress-strain curve for polymethacrylimid foam. *Source:* ref. 13, p. 1965 (reprinted with kind permission from Elsevier Science, Ltd.).

Some foams used for limiting impacts are of the closed-cell type. In these materials, cells are filled with blowing agents during the manufacture. Consequently, when exposed to changes in temperature, significant changes in dimensions can occur. This poor dimensional stability decreases as the relative density of the foam increases. Changes in dimension can be as great as 30% at elevated temperatures. Shrinkage will occur at lower temperatures as the blowing agent condenses. Because the strength of the foam material generally increases at very low temperatures, the worst condition for foams is in the range of –30 to 17°C (–20 to 0°F).

Impact limiters constructed of foam are equipped typically with a thin, metal cover to protect the foam from the environment or from operational loads. Because most of the foams considered in this section retain some permeability and can absorb water or other chemicals to which they are exposed, these foams should be protected by a cover. Some of the polymer foams degrade after long exposure to sunlight. Although fire-resistant polymer foams are available, a cover may help contain toxic or flammable gases generated by burning of the foam. Certainly, the cover helps protect the foam from local damage caused by minor impacts.

R. Shield and C. Covington[15] provided data relating to the strain-rate sensitivity of foam materials. They report no clear trend in response of rigid urethane foam as impact velocity went from 11 to 23 m/s (37 to 74 ft/s), but no description of their experimental procedure was provided. W. E. Woolam[16] found a small strain rate effect for some of the foams he tested. T. K. Hill and W. W. Joseph[17] ranked a variety of materials, including polyurethane foams of various densities using both static and dynamic tests. There was some increase in specific energy absorbed in dynamic tests over static tests, especially for low densities. All specimens in these tests were confined laterally by steel tubes, so the dynamics of gas release during the crushing process may have affected the specific energy values found. None of the other literature reviewed discussed the effects of strain rate on energy absorption.

Reference 13 presents a compilation of data that relate Young's modulus to foam density for a wide range of foams. The data are presented in Fig. A.6 as the ratio of Young's moduli for the cellular solid to the cell-wall material vs the density ratio for the same materials.

The magnitude of the plateau stress for different foam materials increases directly with the foam-material density. Figure A.7 shows the dependence of plateau stress on density typical of different types of foamed materials. As shown by the data represented in this figure, the brittle foams exhibit a wide range of variability, while the plastic and elastomeric foams show a monotonically increasing plateau stress with increasing density. Poisson's ratio is approximately one-third in the initial elastic region for most foamed materials. In the plateau region, however, the ratio is effectively zero.

A.2.1.1 Flexible Foams

Foams that maintain their elasticity when subjected to loads are termed *elastomeric* or *flexible* foams. This category of materials includes polyethylene and polyurethane. The plateau stress for polyethylene foams is typically much greater than that for polyurethane. Figure A.8 shows the plateau stress vs foam density for flexible foams. Manufacturers of flexible foams for package cushioning display the characteristics of the foam in plots of "static stress" vs peak acceleration

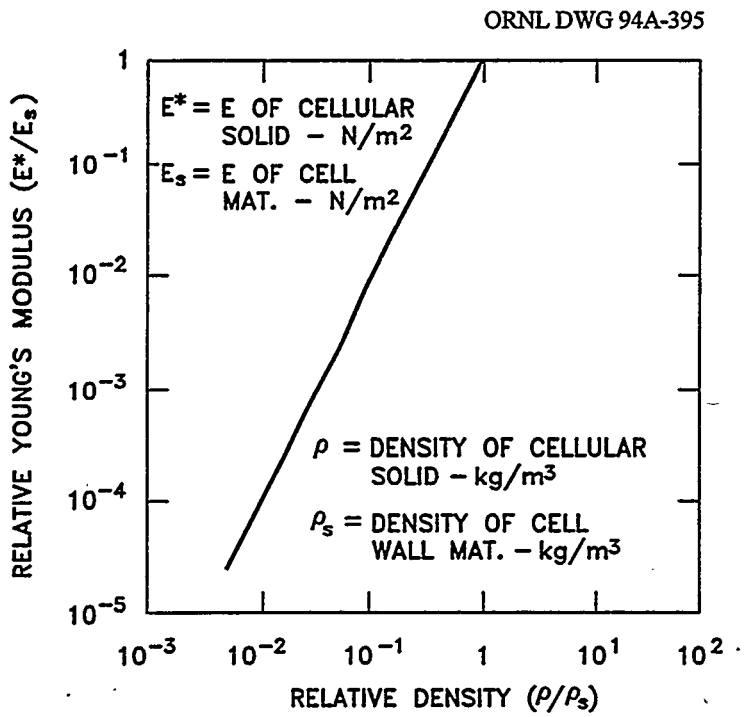

Fig. A.6. Variation of Young's modulus with density. *Source:* ref. 13, p. 1967 (reprinted with kind permission from Elsevier Science, Ltd.).

corresponding to a 0.75-m (30-in.) drop test for different thicknesses of foam. The static stress indicates the weight per unit area of the mass that is being cushioned. A typical display for a low-density polyurethane foam[18] is shown in Fig. A.9.

A.2.1.2 Plastic Foams

Plastic (or rigid) foams are characterized by the failure mechanisms exhibited by the cells of the foam under increasing load.[19] Failure occurs in these foams when the cell walls yield and then collapse when fully plastic hinges develop. The yield or plateau stress for plastic foams is significantly higher than for flexible foams. The plateau stress vs foam density is shown in Fig. A.10 for a polymethacrylimid foam. The variation of compressive strength of a low-density, rigid urethane foam with foam density is shown in Fig. A.11.

A.2.1.3 Brittle Foams

Brittle foams are characterized by an initial linear stress-strain relationship followed by an irregular but approximately horizontal region.[13] The mechanism that governs the latter region is the random fracturing of the cell walls. This fracturing accounts for the erratic nature of the stress-strain curve in this region. The location of the yield-stress plateau is also very irregular, thus making the behavior of brittle foams hard to predict and the use of these foams unattractive as impact absorbers in critical applications.

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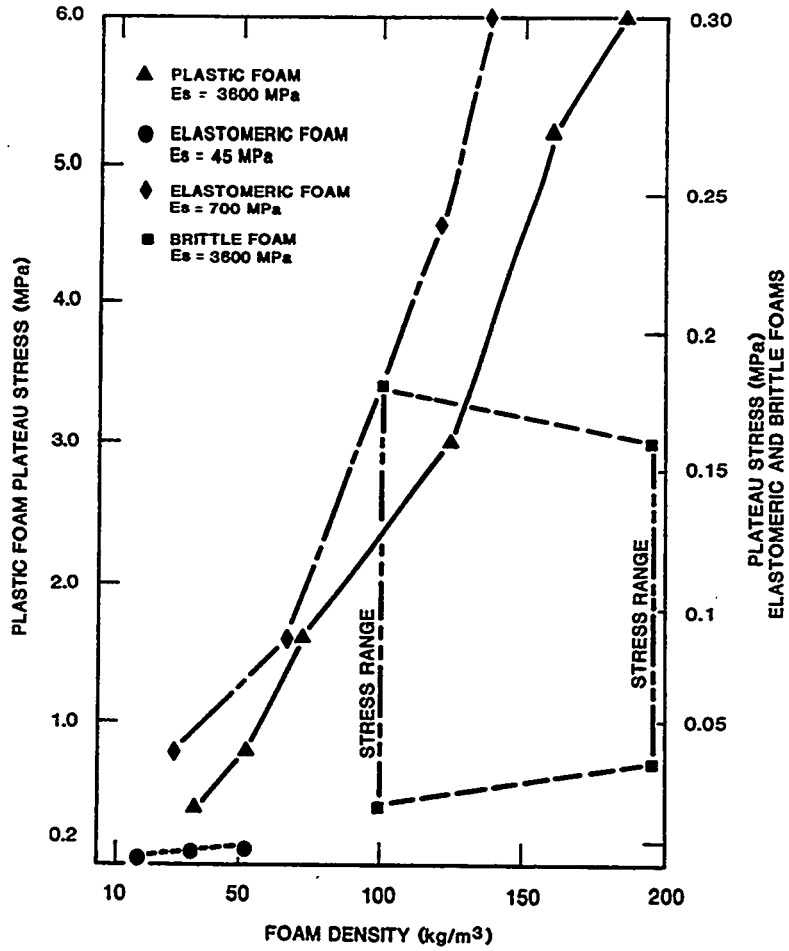

Fig. A.7. Plateau stress vs density of elastomeric and brittle foams. *Source:* ref. 13 (data adapted with kind permission from Elsevier Science, Ltd.).

A.2.2 Examples of Successful Use

No cask design using foam energy absorbers was found that has recently obtained an NRC license. However, one cask in use has passed the 9-m (30-ft) drop test: the Paducah Tiger Overpack.[20] A fuel pin shipping cask,[21] NuPac Model – T3, has been certified by the NRC as USA/9132/B(M)F.

A.2.3 Sources of Material Property Data

Foams are made from a variety of materials which determine their properties. Manufacturer's data provide complete information for the analysis of rigid polyurethane foams, including temperature effects and variability of material properties. The only data on other foamed materials were found in ref. 13, in which stress-strain curves for a flexible polyethylene foam, a rigid polymethacrylimid foam, and a brittle ceramic foam are displayed.

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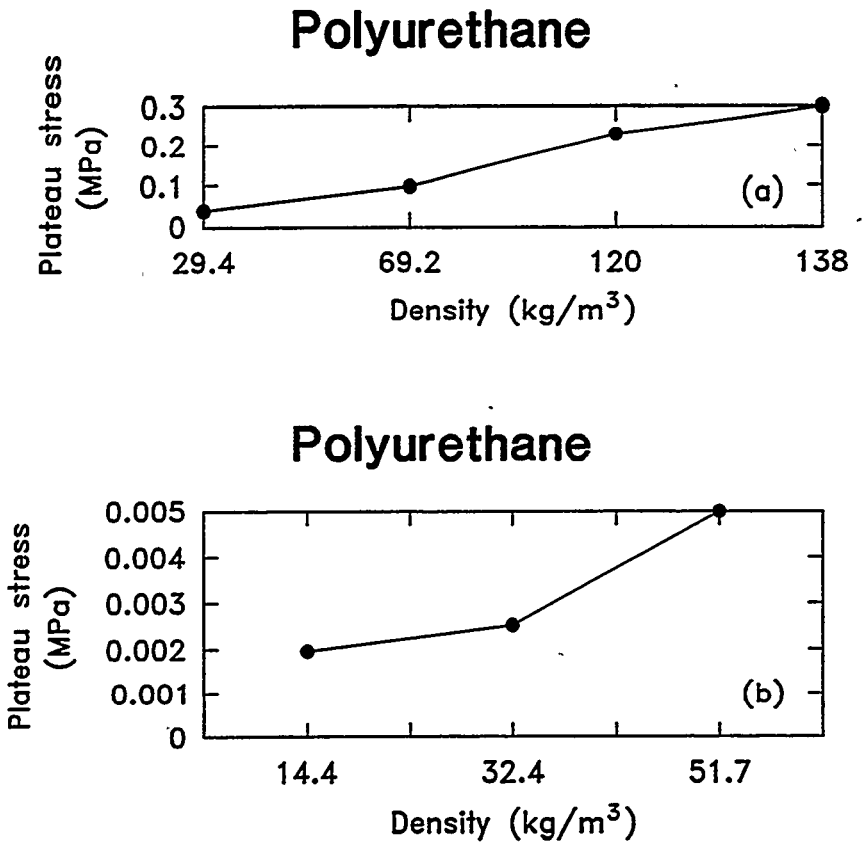

Fig. A.8. Plateau stress vs density of selected flexible foams. *Source:* ref. 13, (data adapted with kind permission from Elsevier Science, Ltd.).

A.2.4 Design Optimization Using Foam Energy Absorbers

Studies have been made of the design of energy absorbers using foams.[16,22] Similitude analyses have shown that an optimum specific energy capacity of the material will result in the minimum accelerations. The governing relationship is shown by a family of curves which plot the Janssen factor vs the energy absorbed per unit volume. The Janssen factor is the ratio of the peak acceleration during impact to the acceleration of an ideal absorber dissipating the same amount of energy. This family of curves are concave upward U-shaped curves. The optimum design is then the combination of foam properties which minimize the peak acceleration. The parameter that identifies different members of this set of curves is the ratio of impact velocity to foam thickness, V/h. This parameter is the initial engineering strain rate for the foam at impact. A typical family of curves is shown in Fig. A.12. The interpretation of the form of this family of curves is straightforward for a given weight to be protected, a given drop height, and a given thickness of foam. Some foams are too stiff and therefore stop the weight before the ultimate strain of the foam is attained, whereas other foams are so soft that the weight does not stop until the foam reaches lockup. Between these two extremes, the optimum design occurs in which maximum strain in the foam is achieved without entering the lockup region. This maximizes the use of the strain range for which the stress remains relatively constant, and consistent with minimum acceleration. Thus, optimizing the use of foam in energy absorbers requires specifying the contact area and

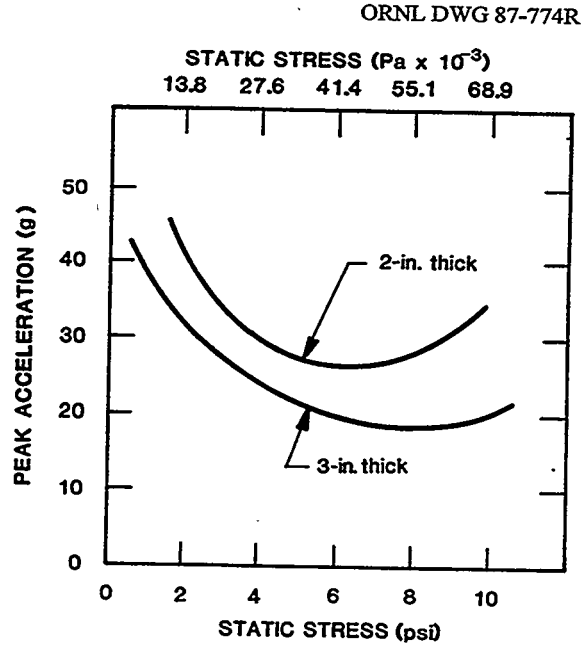

Fig. A.9. Peak acceleration vs static stress of low-density polyurethane foam (Olin Product data from ref. 18).

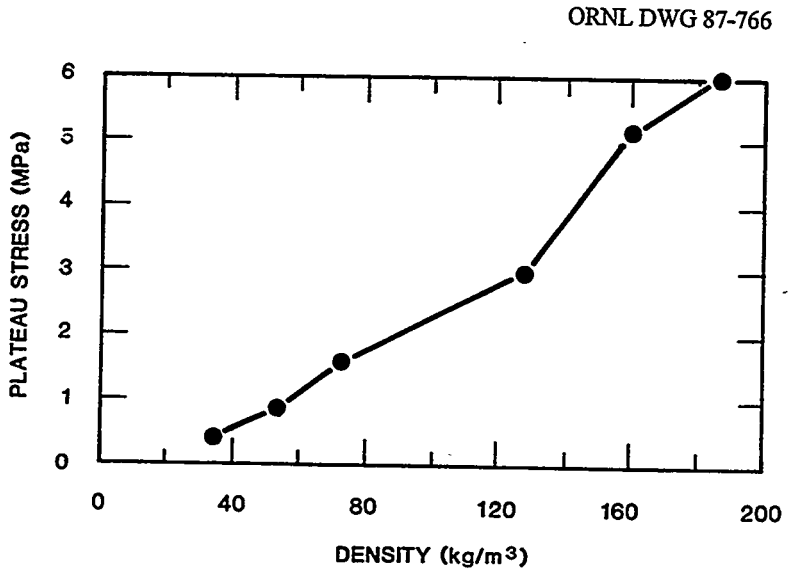

Fig. A.10. Plateau stress vs density for a polymethacrylimid. *Source*: ref. 13 (data adapted with kind permission from Elsevier Science, Ltd.).

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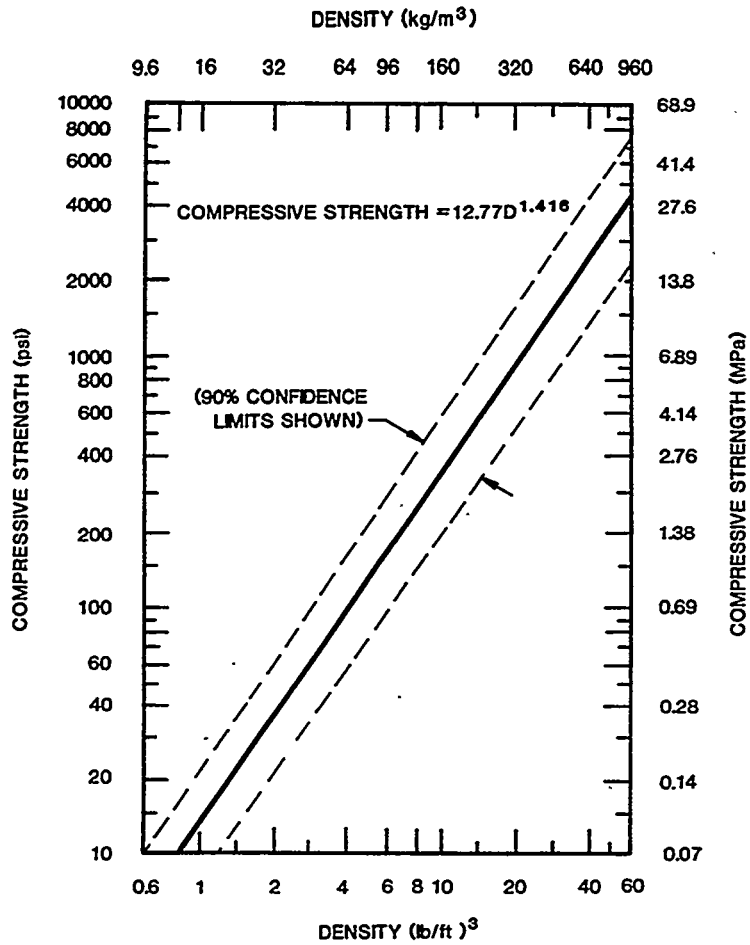

Fig. A.11. Compressive strength vs density for a low-density, rigid urethane foam. *Source:* ref. 14.

thickness of the foam as well as choosing the foam yield stress to achieve the best design. The steps in the design procedure are the following:

1. specification of the weight of the structure to be protected by cushioning,
2. specification of the height of free drop to be experienced or the equivalent velocity at impact of the structure to be protected by cushioning,
3. choosing a reasonable thickness for the foam, and
4. selecting the foam resulting in the minimum Janssen number according to a set of experimental curves for the selected range of foam materials.

The selection of the thickness and area of contact of the foam is equivalent to specifying the volume of foam that will be active in absorbing the impact energy. This feature of the design requires careful attention because the shape of the maximum acceleration curves increases on either side of the optimum design point. Even for drop heights that are less than the design height, the Janssen number would increase as the value of the abscissa decreased. However, the maximum acceleration would be less because it would be a larger fraction of the maximum "ideal"

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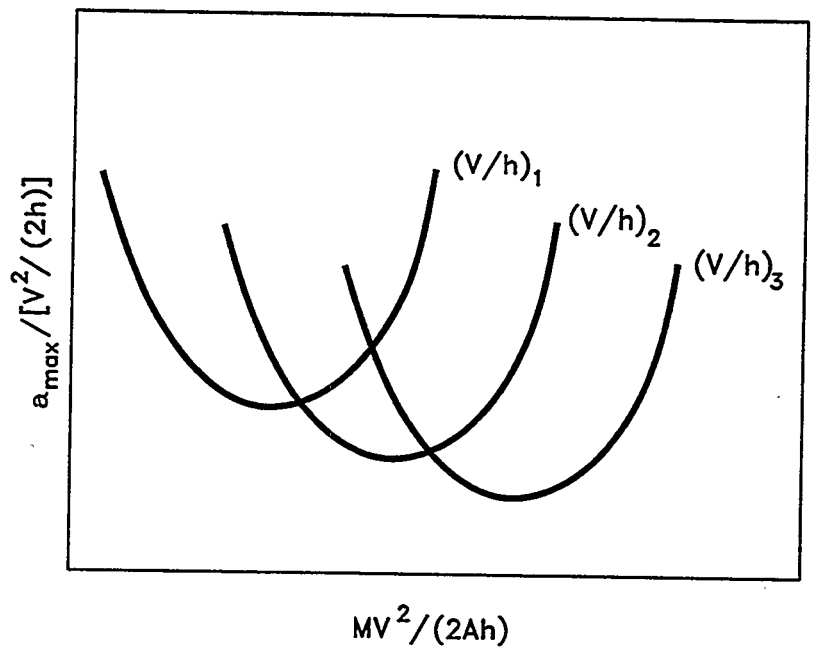

Fig. A.12. Janssen factor vs energy absorbed per unit volume. *Source:* ref. 22.

acceleration (i.e., $V^2/2h$) which would be smaller since it decreases directly with the decrease in drop height.

A.3 HONEYCOMB MATERIAL

A.3.1 Material Characteristics

Honeycomb impact absorbers are constructed by bonding corrugated strips of thin material together to form a relatively low-density grid full of prismatic voids. Honeycombs constructed of paper, fiber reinforced plastic, aluminum, or stainless steel all have nearly ideal load deformation curves for impact limiters; that is, the curves are flat up to about 75% strain in compression. New material shows a sharp rise of load, at impact above the plateau of load but this rise of load can be completely eliminated by slightly buckling the material or by simply dimpling the impact surface of the honeycomb. The voids are all parallel in basic honeycomb; consequently, the material properties are orthotropic. Orthotropic impact absorbers are positioned so the force is parallel to the holes. Special forms of honeycomb are available that have the layers of corrugations made of pieces of basic honeycomb bonded together in different directions. The variation of static crush strength of corrugated aluminum honeycomb with honeycomb density is shown in Fig. A.13. For honeycomb densities greater than approximately 70 kg/m^3, the variation of crush strength with density can be represented by the linear relationship,

$$\sigma_c = 90.74 \times 10^3 \rho_t - 4.69 \times 10^6 , \qquad (A.19)$$

where ρ_t is measured in kg/m^3 and σ_c is given in N/m^2.

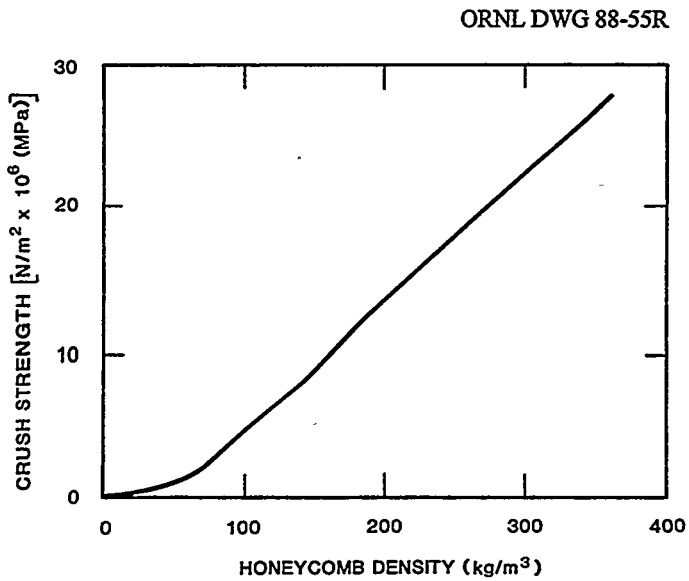

Fig. A.13. Static crush strength vs density of corrugated aluminum honeycomb. *Source:* ref. 23.

Assuming a useful crush of 70% of the honeycomb thickness, the absorbed energy per unit mass in N · m/kg is given by

$$e = 0.7\ \sigma_c/\rho_t\ . \tag{A.20}$$

Substituting for σ_c,

$$e = 63.5 \times 10^3 - 3.28 \times 10^6/\rho_t\ . \tag{A.21}$$

Note that e has units of N·m/kg and is useful for honeycombs whose densities are greater than about 70 kg/m^3.

The maximum specific energy absorbed as a function of density for corrugated aluminum honeycomb is presented in Fig. A.14. Typical mechanical properties of type-5052 aluminum honeycomb materials[23] are given in Table A.1.

Manufacturers' literature suggests that strain rate effects increase the crush strength of honeycomb by up to 30% in a dynamic test over the static value.[23] Experiments by J. M. Lewallen and E. A. Ripperger[24] show that peak crush strength is increased over the static value by 12% for an impact velocity of 15 m/s (50 ft/s) and is increased by 25% for an impact velocity of 30 m/s (100 ft/s), but there is almost no change in total energy absorbed with the change in impact velocity. These results suggest using a strain rate factor of 1.11 to match the impact velocity from a 9-m (30-ft) drop.

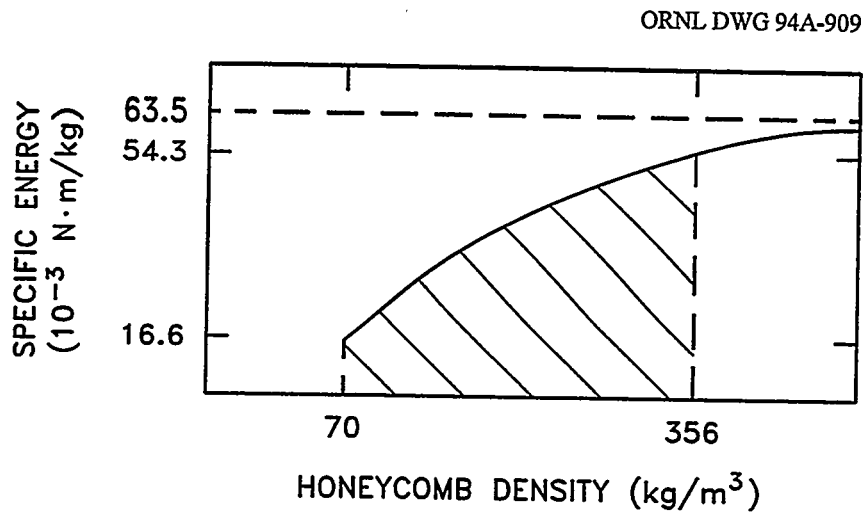

Fig. A.14. Specific energy absorbed vs density of corrugated aluminum honeycomb.

Table A.1. Mechanical properties of aluminum honeycomb

Cell size (10^{-3} m)	Foil thickness (10^{-6} m)	Normal density (kg/m^3)	Crush strength (MPa)	Compressive strength (Pa)
3.18	76.2	332.2	11.03	15.9
3.18	101.6	429.1	16.2	21.4
3.18	152.4	611.8	27.6	35.2
4.76	76.2	224.2	6.2	8.3
4.76	101.6	293.4	9.65	15.2
4.76	152.4	434.6	16.5	22.1
6.35	152.4	290.7	9.3	14.5

Source: ref. 23.

Metal honeycombs are usually enclosed because, although they are not affected by moisture, the metal used is typically 0.018 to 0.15 mm (0.0007 to 0.006 in.) thick and can be damaged easily or can be a safety hazard. Also manufacturers provide a special form of honeycomb with an outer wrapper to prevent reduced strength at the free edges of a honeycomb plate. An enclosure can provide the same strengthening effect. Protective coatings are available to protect against corrosion in salty, humid environments. The influence of temperature on yield strength is often included in manufacturers' literature[23] that describes the material from which the honeycomb is fabricated.

A.3.2 Examples of Successful Use

Honeycomb material is not used as the exclusive impact absorber in any cask licensed since 1970, but it is used to protect the closure of a strong inner package from impact by the innermost containment in the PAT-1,[25] a cask designed for air transport which has been successfully tested

with an impact velocity of 128 m/s (422 ft/s). The PAT-1 is cylindrical, about 1.22 m (4 ft) high and 0.61 m (2 ft) in diameter. The primary energy-absorbing material is redwood. The container is made up of four nested containers. The inner two are separated by a honeycomb cushion. The innermost container is a sealed can with radioactive material inside. This can is contained within another metal container with a bolted closure. The honeycomb protects the bolted joint from a secondary impact from the inner can.

A.3.3 Sources of Material Property Data

Manufacturers of honeycomb for impact absorption provide the static crush strength, a conservative bound on strain rate effect, and strength variation with temperature. These are the properties needed in a constant force type of analysis which applies very well to honeycomb materials so long as the honeycomb pad is not larger than the part of the cask that is indenting. Manufacturers do not supply the material properties required for a finite-element analysis, including the initial elastic modulus and the slope of the stress-strain curve as the material locks up. The information is available from published experimental data (e.g., refs. 3 and 24) for a few particular honeycomb designs. Off-axis crush strength data are also lacking.

A.3.4 Design Procedures

The honeycomb materials considered herein are ideal candidates for the uniform crush analysis, but lack of material property data makes finite-element analysis difficult without an experimental program. Strength and sizing calculations were adequate for the design of the honeycomb used in the PAT-1 cask because only uniaxial protection was desired and because the load limit to prevent yield in the critical closure was known. A pad of aluminum honeycomb is being used to mitigate the impact force produced by an inner container on the outer closure of a nuclear waste truck cask that was designed using a finite-element analysis.[26] The honeycomb was modeled as an elastic-plastic material (which was a conservative design according to the analyst) relative to using the crushable foam model. The elastic-plastic model was chosen because of the availability of appropriate material property data. However, no reference for the material property data was given in ref. 26.

A.4 WOOD IMPACT LIMITERS

A.4.1 Material Characteristics

Because wood ranks among the highest of all materials in specific energy absorption it is popular for protecting casks in spite of wood being orthotropic. According to ref. 3, of 45 materials tested statically, including foamed materials and honeycombs, the most desirable 10 were different samples of wood loaded—parallel to the grain. This ranking indicates that wood comes closer to being the ideal impact absorber; that is, it produces a wider flat stress-strain curve than do most impact-absorbing materials. All material properties of wood, however, vary depending on the angle from the direction of grain in the wood. Reference 3 listed wood loaded perpendicularly to the grain as one of the least desirable of the 45 energy absorbers tested. According to the *Wood Handbook: Wood* as an *Experimental Material*,[27] the variation in any strength property can be described by a Hankinson-type formula,

$$N = PQ/[P \sin^n\theta + Q \cos^n\theta] \ , \tag{A.22}$$

where

N = strength property,
θ = angle from the fiber direction,
P = strength parallel to the grain,
Q = strength perpendicular to the grain,
n = empirically determined constant.

Some values of n and Q/P are as follows:

Property	n	Q/P
Compressive strength	2–2.5	0.03–0.4
Modulus of elasticity	2	0.04–0.12

The grain of wood is not always parallel to the same direction throughout an entire piece of wood but may vary about a mean direction. Wood with a variation in grain angle, relative to the principal grain direction, greater than 7% should be avoided to prevent a reduction in compressive strength.

Mechanical properties are also affected by moisture content and temperature of the wood. Drier wood is usually stronger, and impact absorbers are made of dried wood enclosed in a protective cover. Like most materials, the properties of wood generally decrease as temperature is increased and improve as temperature is reduced. The average crushing stress for redwood plotted against temperature is shown in Fig. A.15.[28] All specimens had density of approximately 335 kg/m^3. The modulus of elasticity is down about 10% at 38°C (100°F) compared with 20°C (68°F) in wood with a 12% moisture content. The reduction in strength is much less (<2%) for completely dry wood. Long-term exposure to high temperature causes a permanent loss of strength.

Experimental stress/strain curves indicate that balsa and redwood, loaded parallel to the grain, have nearly ideal properties for use as impact absorbers; they have nearly flat response up to high strains. References 13 and 17 show that balsa is capable of strains over 80% before lock-up. W. A. Von Riesemann and T. R. Guess[28] studied redwood and measured strains up to 73% before lockup. Load-deflection curves for redwood for a range of temperatures, taken from ref. 28, are shown in Fig. A.16. Figure A.17 shows the maximum specific energy absorbed by redwood for a range of temperatures. The response of woods depends on species and density. Balsa, however, showed a variation of less than 10% in energy absorbed per weight of crushed material in tests of densities between 96 and 192 kg/m^3 (6 and 12 lb/ft^3). When dynamically tested, balsa often displays a peak stress at the yield point higher than the plateau stress found in the flat center section of the stress-strain curve and the difference in stresses does depend on density. The amount the peak stress exceeds the plateau stress ranges from 37% for low densities (105–118 kg/m^3 {6.5–7.4 lb/ft^3}) to 78% at high density (185–200 kg/m^3 {11.5–12.4 lb/ft^3}).[29] Even though stress and strain are in the flat region of the stress/strain curve, the balsa exhibited a Poisson's ratio of zero. The effectiveness of balsa as an impact absorber when loaded away from the axis of the grain can be seen in Fig. A.18, which is plot in which the peak deceleration (or

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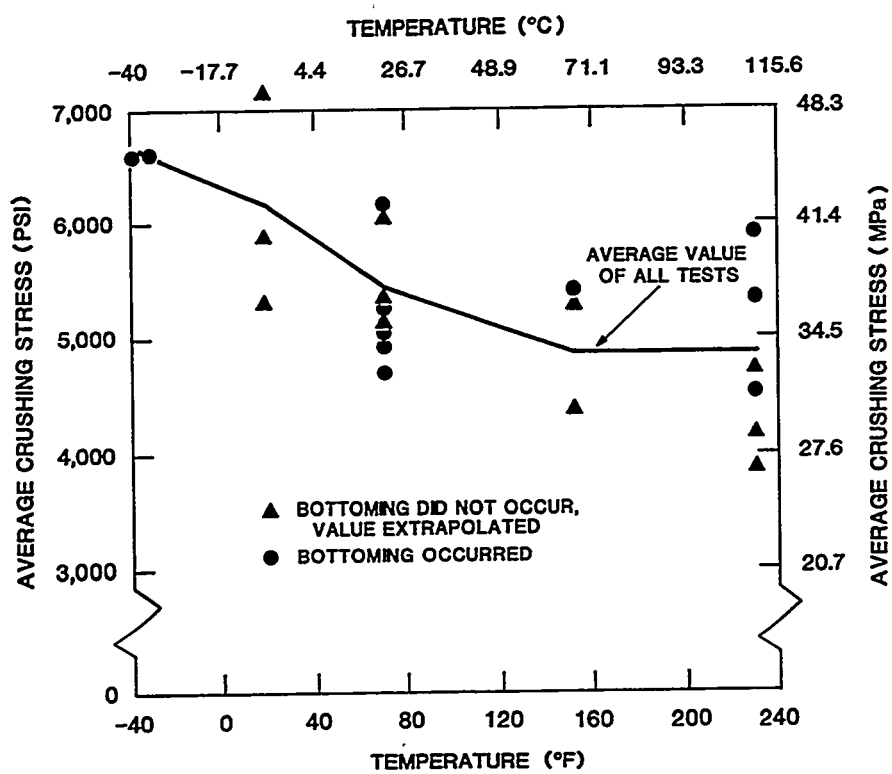

Fig. A.15. Average crushing stress vs temperature for redwood. *Source:* ref. 28.

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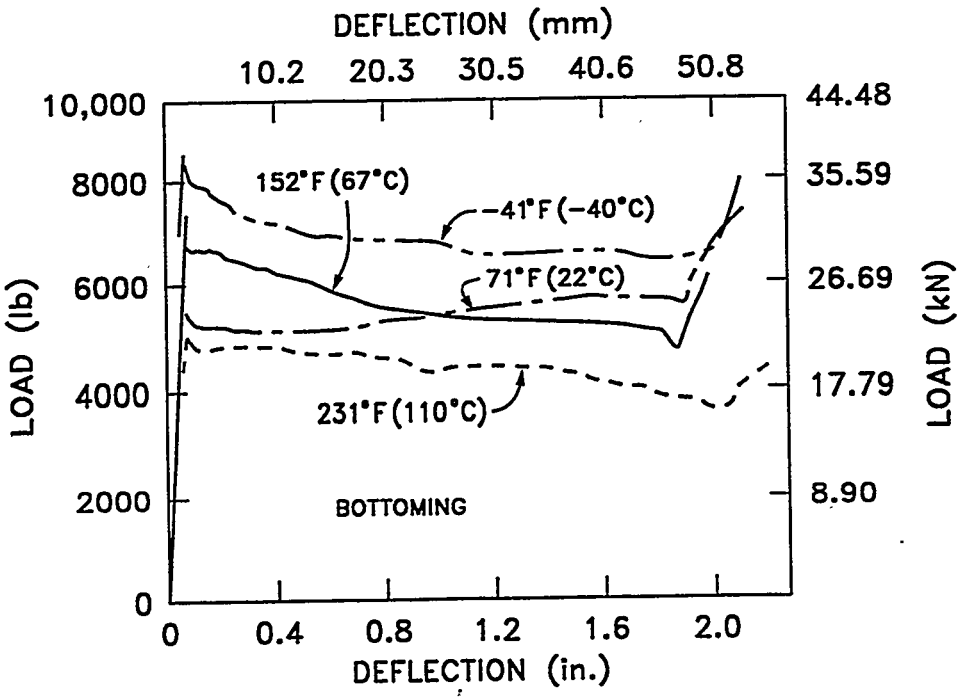

Fig. A.16. Load-deflection curves for redwood. *Source:* ref. 28.

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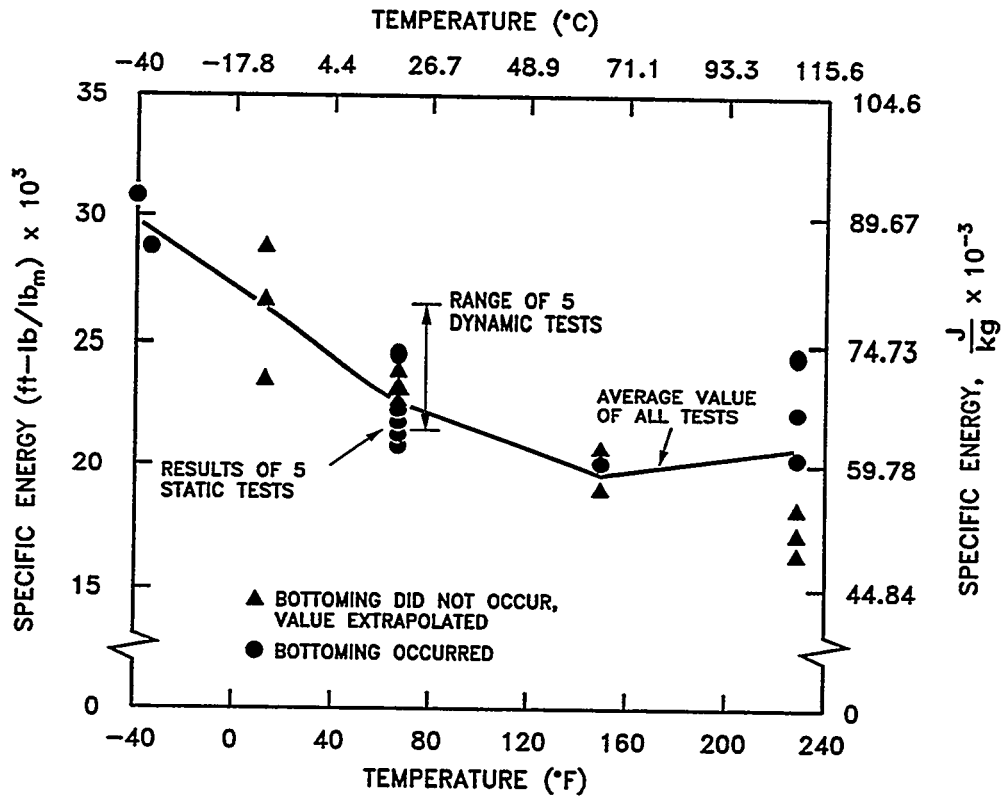

Fig. A.17. Maximum specific energy vs temperature for redwood. *Source:* ref. 28.

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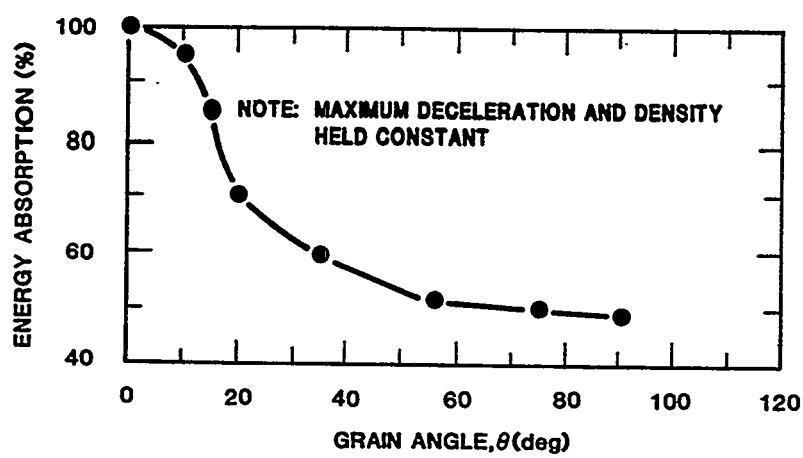

Fig. A.18. Influence of impact angle to grain direction on energy absorption in balsa. *Source:* ref. 29.

stress) is held constant. The plot underemphasizes the drop in effectiveness away from the grain direction because the test parallel to the grain was arbitrarily limited in maximum strain relative to the other orientations. Balsa that was constrained by crushing the center of a block absorbed 5 to 10% more specific energy than laterally unconstrained balsa. A. C. Knoell[30] reported a specific energy of balsa at 93°C (200°F), 20% less than at 20°C (68°F). In redwood, a temperature increase from 20 to 110°C (70 to 230°F) produces about 10% reduction in specific energy.

D. L. Daigle and J. O. Lonborg[29] found an increase in specific energy absorption of balsa for dynamic testing over static testing. A static load deflection curve for balsa is shown in Fig. A.19. Table A.2 gives the mechanical properties of balsa for a range of material density. Static-specific energy absorption was only 73% of the specific energy absorption measured dynamically. However, a variation in impact velocities between 6 and 15 m/s (20 and 50 ft/s) produced no noticeable effect on specific energy absorption.

A.4.2 Examples of Successful Use

Among nuclear spent fuel shipping casks licensed in this country since 1970, five use balsa impact absorbers: Nuclear Fuel Services, Inc., NFS-4; N. L. Industries NLI 1/2 and NLI 20/24; Transnuclear, Inc., TN-8 and TN-9; and National Fuel Services, Inc., NFS-5.6. The principal energy-absorbing material in the air-transported PAT-1 is redwood.[25]

A.4.3 Sources of Material Property Data

References 3, 17, 28, 29, and 30 give material properties for balsa and redwood that are useful for uniform crush analysis of those woods loaded parallel to the wood grain. Reference 6 uses

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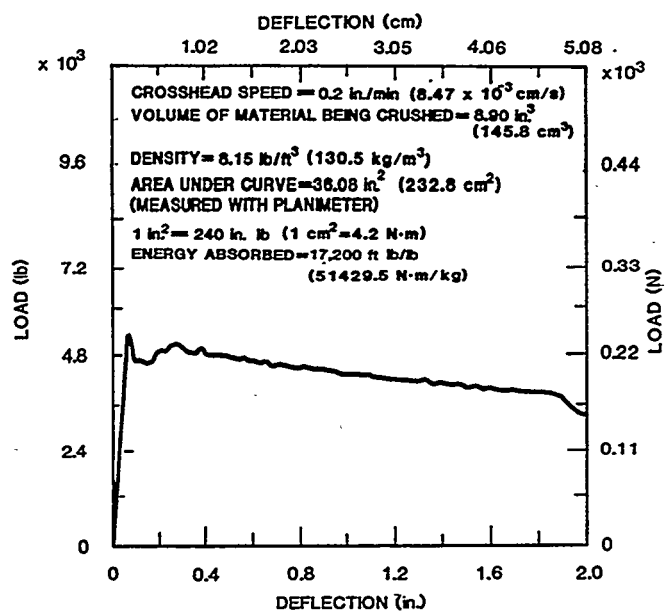

Fig. A.19. A static load deflection curve for balsa. *Source:* ref. 29.

Table A.2. Averages of peak and mean crushing strengths for various densities of balsa

Density range (kg/m^3)	Average of peak crushing strength, MPa	Average of mean crushing strength, MPa	Peak strength, % of mean strength	Number of tests
104–119	11.65	8.49	137.0	9
120–135	14.34	10.01	143.2	18
136–151	10.07[a]	6.65[a]	153.3	3
152–167	18.96	11.20	169.3	8
168–183	21.17	12.27	172.5	17
184–199	21.06[a]	11.82[a]	178.5	2

[a]Low values believed due to insufficient testing.
Source: ref. 29.

these same data to develop the material properties necessary to use the elastic-plastic and the crushable foam subroutines of the HONDO program. The effectiveness of wood impacted off the grain axis has not been fully explored in the dynamic regime.

A.4.4 Design Procedures

The standard techniques used to size foam impact limiters can be used for wood limiters as long as the wood is loaded parallel to the grain. The Janssen factors can be plotted for different densities or different species to find the optimum material. If even the lightest balsa wood is too stiff, holes can be drilled to reduce the area of the wood. Experiments indicate that wood has low shear strength relative to compressive strength and that for end impact the only wood effective is that in the cylindrical volume projected down from the contact surface between the limiter and the package. For corner drops, the effective cylinder of wood should be projected from the package-interface parallel to the grain of the wood. The strength of wood which is loaded off the grain axis should be reduced from the on axis value using a Hankinson-type formula, but the strength should not exceed the values in Fig. A.15.

A.5 STEEL ENERGY-ABSORBING DESIGNS

Because steel is as stiff as most materials used in packaging construction, an impact limiter must be shaped and positioned to allow the limiter to dissipate energy by distorting plastically. Steel limiters have been designed in the shape of frames made of welded tubing, shells of revolution, and fins. The fins are able to serve a dual purpose as an enhancement to heat transfer and as an impact limiter. Scale-model testing is a valid method of verifying designs of steel impact limiters, but this report is limited to analytical design procedures.

Manual calculation methods have been used to analyze steel-welded-tube-frame impact limiters,[31]

but usually nonlinear finite-element computer programs are used to predict behavior of steel impact limiters. Other metals such as aluminum are not considered. Aluminum is not as ductile as steel and does not maintain its properties at high temperatures.

A.5.1 Material Property Characteristics

Mild steel and stainless steel are common materials for the construction of impact limiters because of their relatively high ultimate strain: 30% for annealed American Society for Testing and Materials (ASTM) A285 carbon steel and 50% for type-304L stainless steel.[32] Stainless steels also possess excellent corrosion resistance. M. J. Rack and G. A. Knorovsky[33] collected available data on stainless steels that are potential shipping cask materials. The data indicated that elastic properties are similar among the group of austenitic stainless steels reviewed. Increasing temperature causes a decrease in elastic tensile and shear moduli but an increase in Poisson's ratio. Figure A.20 shows static properties for a range of test temperatures. Experiments show that while yield strength of type-304 stainless steel increases by a 48-MPa (7.0 ksi) per decade increase in strain rate, the overall stress-strain curve is not altered in shape. Carbon steels show a smaller strain rate effect with a flow stress increasing by 9.0 MPa (1.3 ksi) per decade increase in strain rate.[34]

Steels are considered isotropic, but manufacturing processes impart directional properties. Elongation and toughness are reduced in plates along directions transverse to the rolling or extrusion direction.[35] The same variation in properties must also occur in piping. The yield strength of some steels is anisotropic with respect to principal stresses. Biaxial yield strength is often larger than the uniaxial value, as shown in ref. 35.

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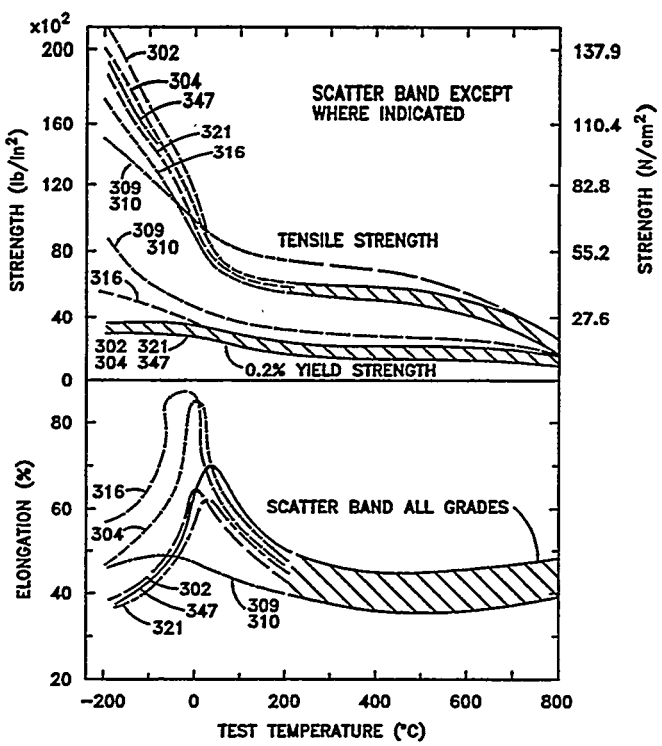

Fig. A.20. Static properties of stainless steels as a function of temperature. *Source:* ref. 33.

Steels follow a kinematic hardening rule,[36] in which the surface moves as the material yields but does not expand in size. However, for a monotonic increase in stress, an isotopic hardening model, which grows about a fixed center, produces the same result. The difference in hardening is important for reversed stresses such as those that occur after an edge-loaded fin buckles and bends.

R. J. Burian and E. C. Lusk considered metallic impact limiters for shipping containers consisting of tube clusters impacted longitudinally.[37] Figure A.21 shows their recommendations for choosing an optimum impact limiter configuration.

T. Kukkola investigated energy absorbers for impact loading for application as pipe restraints in nuclear power plants.[38] Force-deflection curves are given for a number of configurations of circular and square cross-section tubes. However, no indication of the steel chemistry is given.

P. Hernolsteen and L. C. Leblois performed a series of experiments to determine the dynamic material properties of steel bars, as well as copper bumpers and lightweight concrete.[39] Their results indicate that the dynamic true stress—true strain relationship for stainless steel is

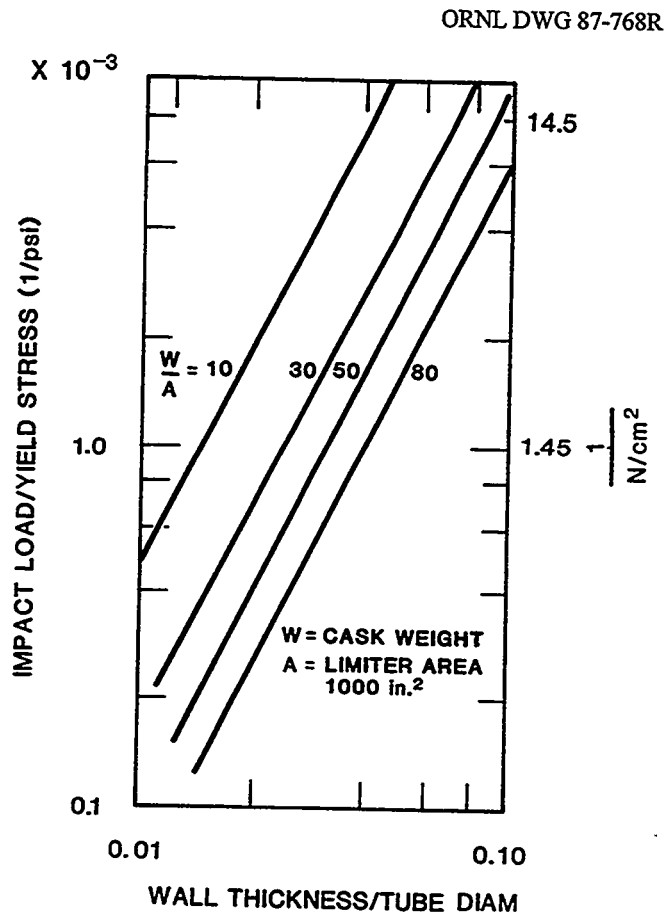

Fig. A.21. Tube cluster impact-limiter design configuration. *Source:* ref. 37.

$$\sigma = \sigma_0(1 + \lambda\varepsilon) , \tag{A.23}$$

where σ_o and λ are material parameters that have the following ranges, depending upon heat treatment and the test temperature and σ is the true strain:

$$235\ MPA \leq \sigma_0 \leq 645\ MPa$$
$$1.96 \leq \lambda \leq 6.29 .$$

Figure A.22 shows the variation of strain energy density vs strain for stainless steel rods tested in tension for a range of temperatures. Stability considerations mitigate against the use of tubes or rods in compression. Short members, which will avoid buckling do not provide sufficient volume of material for the required energy dissipation. K. R. F. Andrews et al[40] tested a large number of aluminum tubes in order to identify the failure modes for various geometries. Figure A.23 shows their results.

A.5.2 Examples of Successful Use

The General Electric Company IF 300 rail-transported spent fuel shipping cask is protected on its ends and side by stainless steel fins.[6] Reference 41 details the tube cluster impact limiter designed for the Peach Bottom No. 1 shipping cask.

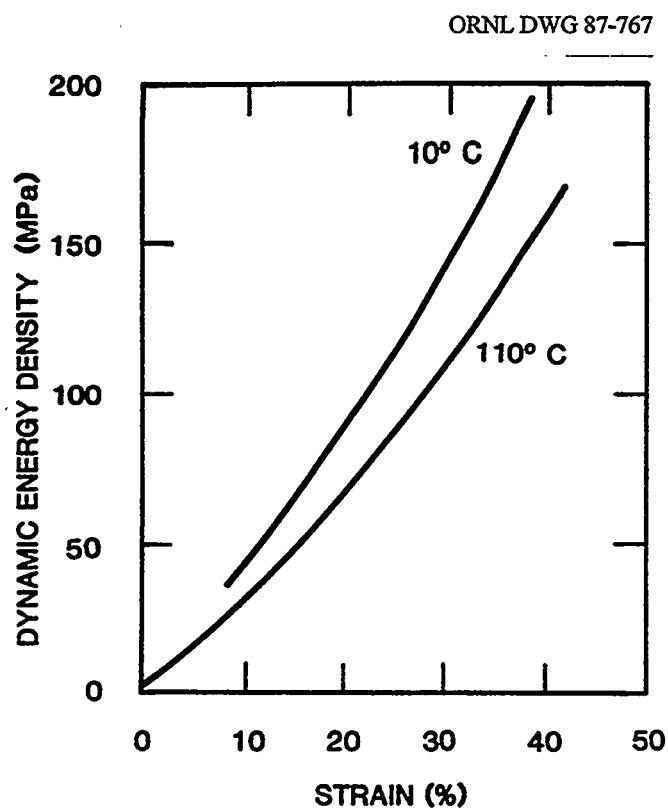

Fig. A.22. Strain energy density vs strain in stainless steel rods (ref. 39).

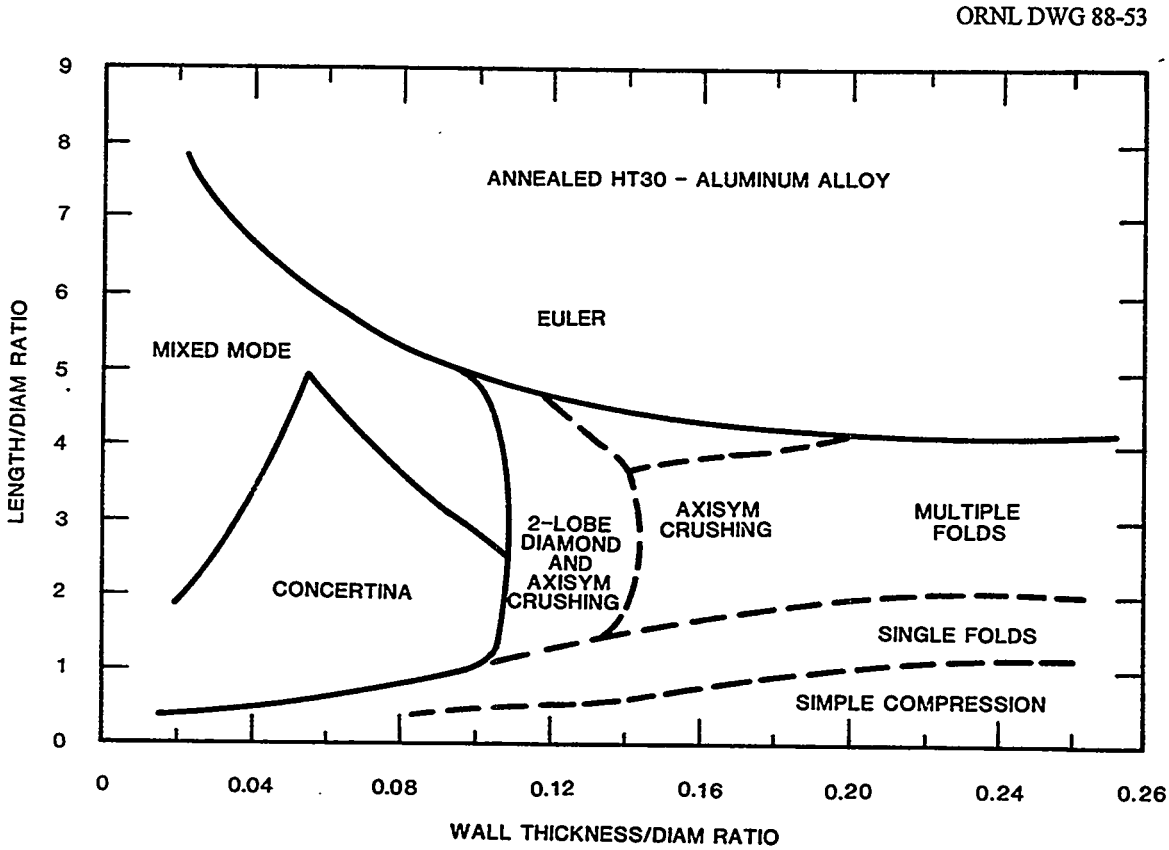

Fig. A.23. Failure modes of aluminum-tube impact limiters. *Source:* ref. 40.

A.5.3 Sources of Material Property Date

Individual sources of a complete set of data necessary for nonlinear dynamic analyses were not found, but two surveys provide a good starting point for collecting data: ref. 33 identifies sources for temperature and strain-rate-dependent data for stainless steels used for the construction of shipping casks, while ref. 34 includes strain-rate-dependence of other steels. A comprehensive set of data for inelastic but low strain-rate-analysis of 304 and 316 stainless steels is available in ref. 36. Information is lacking on multiaxial high-strain-rate behavior.

A.5.4 Design Procedures

Several published analyses demonstrate the capabilities of the finite-element method for nonlinear analysis of steel impact limiters in nuclear material transport. Adams et al[6] use HONDO II to analyze a single 304 stainless steel fin as it buckles and bends in a simulated impact following a 9-m (30-ft) free fall. Drop tests of a scale model of a cask with ellipsoidal shell end caps confirmed analyses with the DYNA 2-D and DYNA 3-D programs[41] but no discussion of the material properties used is given. The design for a nuclear waste truck cask, including a shell impact limiter undergoing a corner impact, is analyzed with the DYNA 3-D program in ref. 26.

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Appendix B: Thermal Code Benchmark Problems

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B.1 INTRODUCTION

An issue of significant concern in nuclear waste management is the capability of computer codes to simulate correctly the phenomena associated with the transport of radioactive materials. In particular, structural, thermal, criticality, and shielding codes must be able to simulate normal transport and hypothetical accident conditions and the corresponding package response in order to design and certify packages based on analysis.

The U.S. Department of Energy (DOE) has funded programs to provide analytical code benchmarks and to evaluate the many codes used in the design and certification of casks. This appendix contains those problems and the respective solutions developed for evaluating thermal codes.

This effort was in conjunction with code intercomparisons being performed under the auspices of the Organization for Economic Cooperation and Development, Nuclear Energy Agency Committee on Reactor Physics, and Committee on the Safety of Nuclear Installations.

The constraints on the development of the problems were that the package models should be cask-like in nature, be as geometrically simple as possible, and have imposed on them thermal conditions addressed in the Federal Regulations. Therefore, for this exercise, several thermal problems were proposed to simulate normal and accident conditions of transport. These problems were designed to simulate the hypothetic accident conditions given in Title 10 of the *Code of Federal Regulations,* Part 71 (10 CFR 71) while retaining simple calculation geometries. This produced a problem set that exercises the ability of the codes to model pertinent physical phenomena without requiring extensive use of computer resources. Specific conditions were imposed on three different package configurations (Models A, B, and C) to evaluate the ability of popular heat-transfer codes to simulate realistic physical phenomena. The problems are not all-inclusive. They do not address cask-specific thermal phenomena such as a phase change in materials of construction (e.g., a lead shield) with accompanying natural convective heat transfer. They do encompass thermal phenomena encountered in cask analysis such as conduction, radiation, and a specified convective boundary.

*This work performed at Sandia National Laboratories, Albuquerque, N.M., supported by the U.S. Department of Energy under Contract DE-AC04-76DP00789.

The solutions presented are consensus solutions based on computer analyses done by both national laboratories and industry in the United States, United Kingdom, France, Italy, Sweden, and Japan. They can provide code users with a set of standard thermal problems and solutions which can be used to evaluate individual codes. The solutions are representative of the codes currently available for these types of analyses. In all cases, the solutions have been reviewed by all participants, and only those agreed to have been included in this appendix. When a closed-form analytical solution is available, it is also presented.

B.2 THERMAL PROBLEMS

The thermal problems simulate normal transport and accident conditions specified in 10 CFR Part 71. These regulations define a thermal event as "exposure of the whole specimen for not less than 30 minutes to a heat flux not less than that of a radiation environment of 800°C (1475°F) with an emissivity coefficient of at least 0.9. For purposes of calculation, the surface absorptivity must be either that value which the package may be expected to possess if exposed to a fire or 0.8, whichever is greater."

To model completely the thermal event requires simulating conduction, convection, and radiation. The problems included in this appendix will exercise these facets of a code. Additional phenomena are not included in this problem set, such as phase change (lead melt) and boiling-water heat transfer.

The codes whose results are used here include COYOTE,[1] HEATING-5 (ref. 2), Q/TRAN,[3] TAC-2D,[4] TEMPEST,[5] TRUMP,[6] and HEATING-6 (ref. 7). DELFINE, which is sponsored by the Commissariat à L'Énergie Atomique in France, and TAU, which is used in the United Kingdom by the Atomic Energy Establishment, Winfrith and British Nuclear Fuels, PLC, are two other (unreferenced) codes that were also used to study the same thermal problems; the results of these latter codes also agreed with the initial results.

B.2.1 Model A

Model A, as shown in Fig. B.1, is a two-region cylinder. The interior region (Region I) contains a volumetric heat source representative of the internal decay heat load from irradiated fuel. This energy is dissipated at the outer surface of Region II by convection. This problem, which has a closed-form analytical solution, tests the ability of the codes to simulate conduction and a convective boundary condition.

The geometric and thermal characteristics of Model A are given in Table B.1. The ambient environment and initial cask temperature are each 54.4°C (130°F).

The steady-state results requested for this problem were (1) centerline temperature, (2) interface temperature (r_1), (3) outer edge temperature, and (4) plot of temperature vs normalized radial position.

The closed-form analytical and numerical solutions reported to 1°C accuracy are given in Table B.2, and the graphical solution reported to 0.1°C accuracy is given in Fig. B.2.

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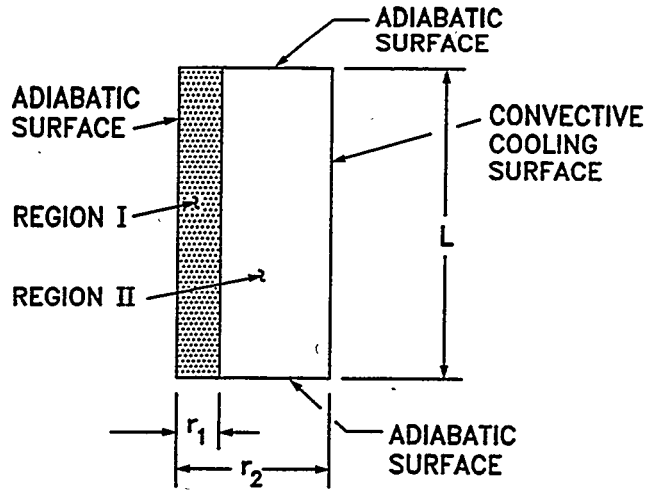

Fig. B.1. Model A—cylinder with internal heat generation.

Table B.1. Model A—cylinder with internal heat generation: characteristics

Characteristics	Region I	Region II
Radius, r	r_1 = 27.43 cm (10.8 in.)	r_2 = 91.44 cm (36 in.)
Length, L	475.2 cm (15 ft)	457.2 cm (15 ft)
Density, ρ	16.03 kg/m^3 (1 lb$_m$/ft^3)	16.02 kg/m^3 (1 lb$_m$/ft^3)
Specific heat, C_p	4.18 × 10^3 J/kg·K (1 Btu/lb$_m$, 36°F)	4.18 × 10^3 J/kg·K (1 Btu/lb$_m$ °F)
Conductivity, k	69.2 W/m·K (40 Btu/h ft °F)	34.6 W/m·K (20 Btu/h ft °F)
Heat source, Q	11.09 kW/m^3 (1071.5 Btu/h ft^3)	
Convective coefficient, h_c		5.67 x 10^{-3} kW/m^2 °C (1 Btu/h ft^2 °F)

Table B.2. Model A—cylinder with internal heat source: results

Solutions	Centerline (°C)	Interface (°C)	Outer edge (°C)
Exact (closed form)	152	149	135
Numerical[a]	152	149	135

[a]Numerical mean average of solutions provided.

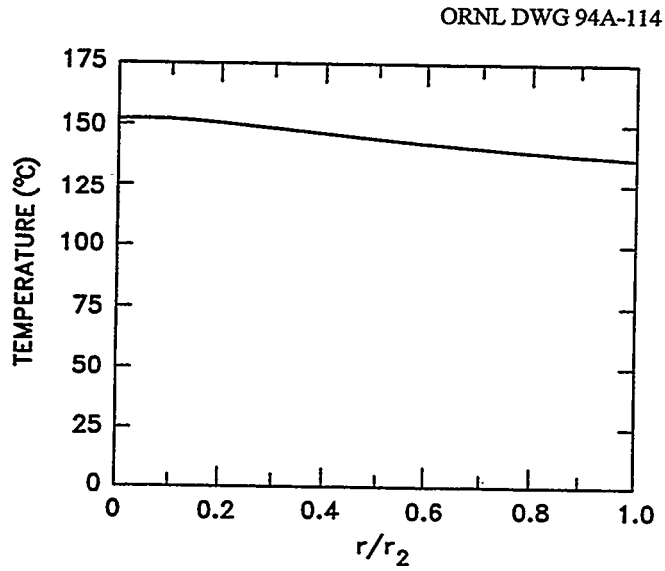

Fig. B.2. Model A—temperature vs normalized radial position.

B.2.2 Model B

Model B, which is shown in Fig. B.3, is based on a prototypic cask configuration consisting of several different annular regions. Region I contains a volumetric heat source simulating irradiated fuel decay heat. Region II is a monolithic cask wall, Region III is a voided neutron shield, and Region IV is the thermal radiation shield. Regions II and IV exchange heat solely by thermal radiation. Heat is also exchanged with the surrounding environment from Region IV by thermal radiation. For simplicity, all surfaces and the environment are assumed to be black (i.e., $\varepsilon = \alpha = 1$).

The following problem simulates exposure of Model B to the regulatory fire and involves three analysis. First, a steady-state solution, with radiant heat loss to a 54.4°C (130°F) environment, must be calculated. Second, using the steady-state solution for the initial conditions, the fire is simulated by an increase in the ambient temperature to 800°C (1475°F) for 30 min. The third analysis involves a cool-down period of 60 min following the fire as the ambient temperature is returned to 54.4°C (130°F). In each case, heat transfer between the cask and environment is solely by black-body radiation.

The geometric and thermal characteristics for Model B are given in Table B.3.

The results requested for this problem were temperatures at (1) the exterior surface, T_3; (2) the interface between Regions II and III, T_8; and (3) the interface between Regions I and II, T_4, as shown in Fig. B.3. The temperatures were requested in tabular form within 1°C accuracy at 0, 30, and 90 min in graphical form with temperature vs time. The tabular results are given in Table B.4, and the graphical results are presented in Figs. B.4 through B.6. The greatest temperature deviation from the mean values of the tabular data was 6°C at the interface between Regions II and III at 90 min.

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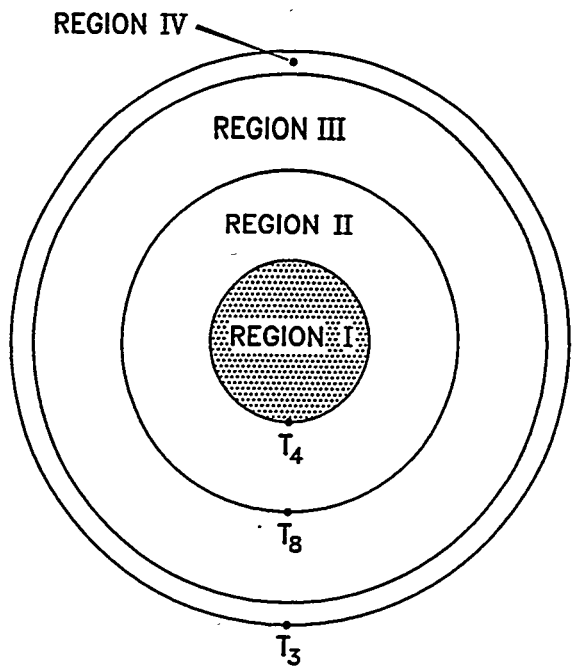

Fig. B.3. Model B—cask with annular regions.

Table B.3. Models B and C—cask with annular regions: characteristics

Characteristics	Region I	Region II
Radius, r	16.51 cm (6.9 in.)	38.74 cm (15.25 in.)
Density, ρ	2707 kg/m^3 (169 lb$_m$/ft^3)	7832.8 kg/m^3 (489 lb$_m$/ft^3)
Specific heat, C_p	0.89 × 10^3 J/kg·K (0.214 Btu/lb$_m$ °F)	0.47 × 10^3 J/kg·J (0.113 Btu/lb$_m$ °F)
Conductivity, k	242 W/m·K (139.7 Btu/h ft °F)	45 W/m·K (26 Btu/h ft °F)
Heat source, Q	38.32 kW/m^3 (3702.6 Btu/h ft^3)	
	Region III	**Region IV**
Radius, r	53.98 cm (21.25 in.)	54.61 cm (21.5 in.)
Density, ρ		7832 kg/m^3 (489 lb$_m$/ft^3)
Specific heat, C_p		0.47 × 10^3 J/kg·K (0.113 Btu/lb$_m$ °F)
Conductivity, k		45 W/m·K (26 Btu/h ft °F)
Heat source, Q		

Table B.4. Model B—cask with annular regions: results

Time (min)	T_3 (°C)	T_8 (°C)	T_4 (°C)
0	137	204	214
30	689	376	263
90	203	298	313

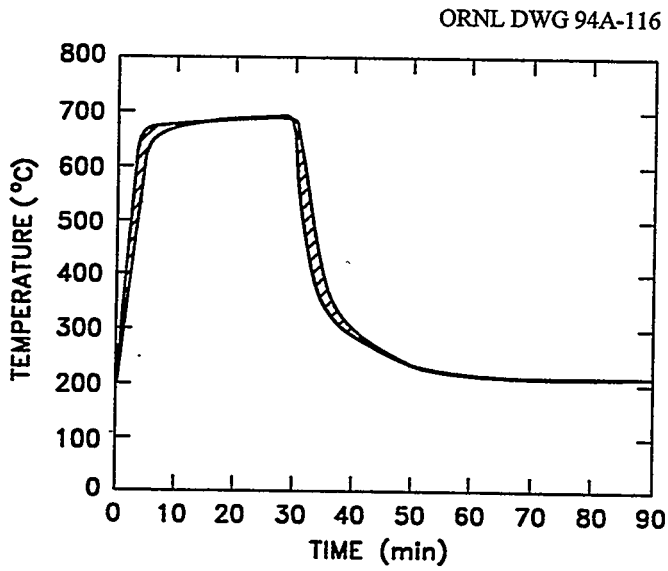

Fig. B.4. Model B—temperature T_3, at cask surface vs time.

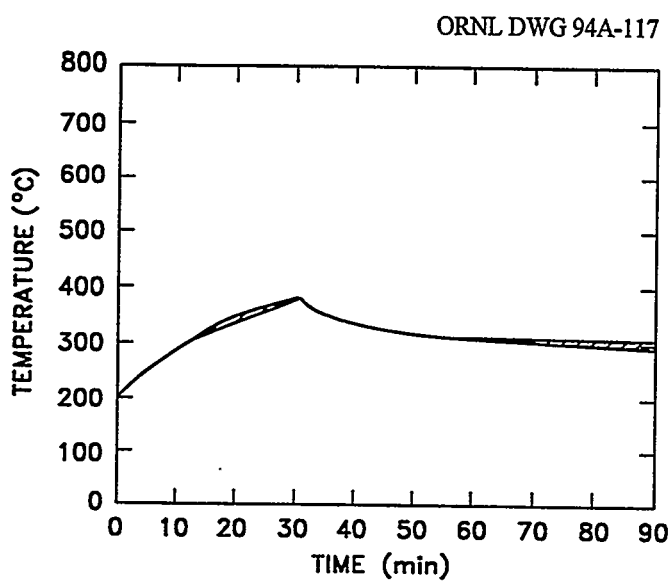

Fig. B.5. Model B—temperature, T_8, at interface between Regions II and III vs time.

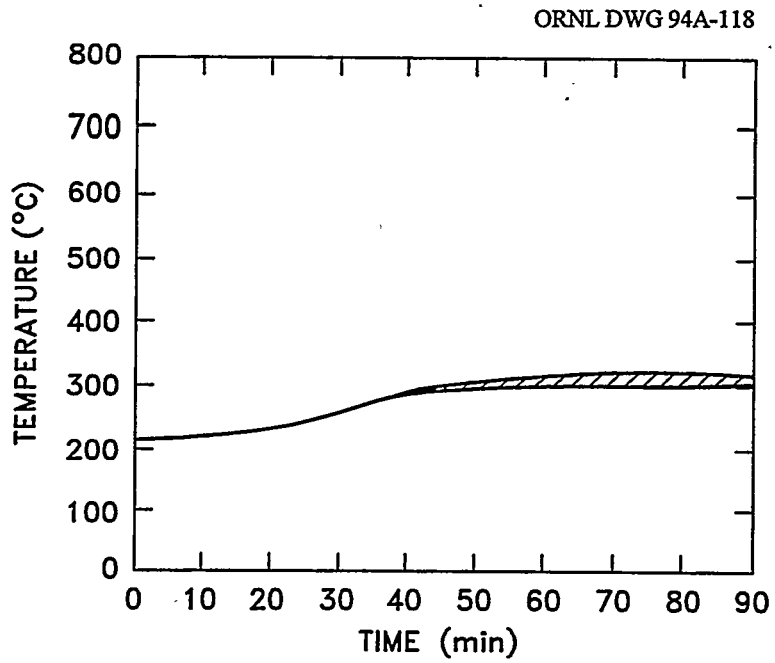

Fig. B.6. Model B—temperature, T_4, at interface between Regions I and II vs time.

B.2.3 Model C

Model C is a modification of Model B. For this problem, a flat plate has been added as shown in Fig. B.7. This plate simulates a truck bed, which acts as a thermal shield. The addition of this shield plate forces the code to simulate a two-dimensional thermal-radiation problem. Radiant heat transfer occurs between the cask and the environment and also between the cask and the shield plate. The remainder of the problem is as described for Model B. The properties of the shield plate are given in Table B.5.

The results requested were tabular data at (1) the center of the shield exposed to the environment, T_1; (2) the cask exterior facing the shield center, T_3; (3) the interface between Regions II and III, T_8; (4) the interface between Regions I and II, T_4; and (5) the exterior of the cask, T_6, as shown in Fig. B.7. These data were requested at 0, 30, and 90 min after the imposition of the simulated fire. The graphical data requested from the participants were plots of temperature vs time at these locations. Three of the codes (CINDA-3G, Q/TRAN, and TAC-2D) solve explicit two-dimensional thermal-radiation problems. The tabular results presented in Table B.6 and the graphical results shown in Figs. B.8 through B.12 represent results only these three codes.

The largest deviation from the mean temperatures was 4°C at location T_4 after 30 min.

This appendix contains thermal problems and the corresponding numerical solutions as obtained from analyses using several codes and code types. These results can be used in evaluating other thermal codes that might be employed in the design and certification of radioactive materials packagings in order to determine and validate, for example, the effect of mesh size, time steps, and material properties employed.

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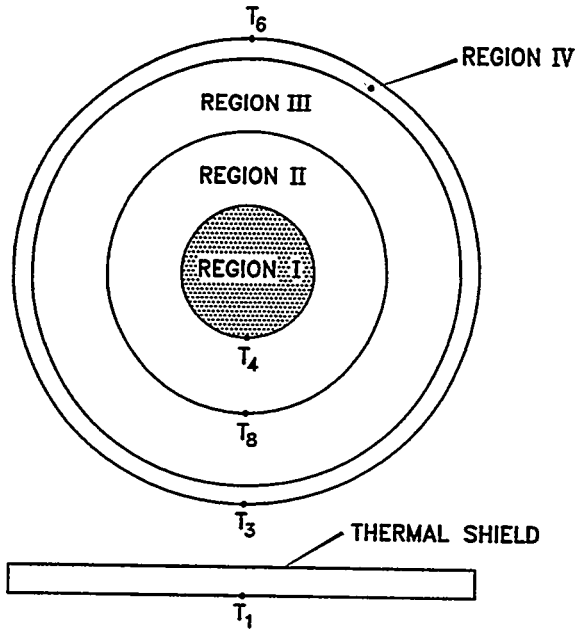

Fig. B.7. Model C—cask with annular regions and shield.

Table B.5. Model C—shield plate characteristics

Width, C	109.2 cm (43 in.)
Thickness, δ	2.54 cm (1 in.)
Distance from cask, D	30.48 cm (12 in.)
Density, ρ	7832.8 kg/m^3 (489 lb$_m$/ft^3)
Specific heat, C_p	0.47 × 10^3/kg·L (0.113 Btu/lb$_m$ °F)
Conductivity, k	45 W/m·K (26 Btu/h ft °F)

Table B.6. Model C—cask with annular regions and shield plate: results

Time (min)	T_1 (°C)	T_3 (°C)	T_4 (°C)	T_6 (°C)	T_8 (°C)
0	89	147	217	139	208
30	765	662	261	689	350
90	206	245	314	203	301

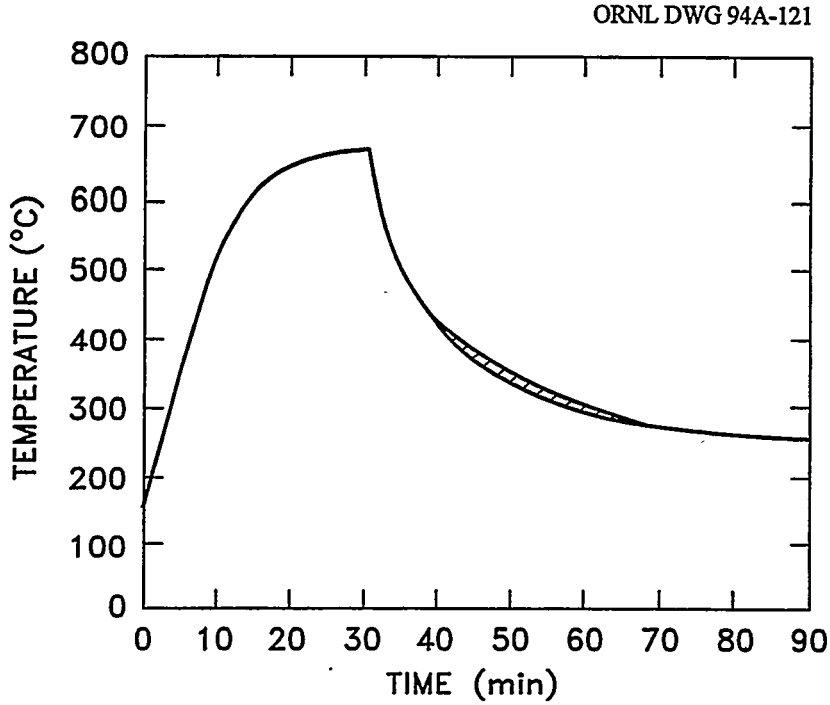

Fig. B.8. Model C—temperature at bottom of cask surface, T_3, vs time.

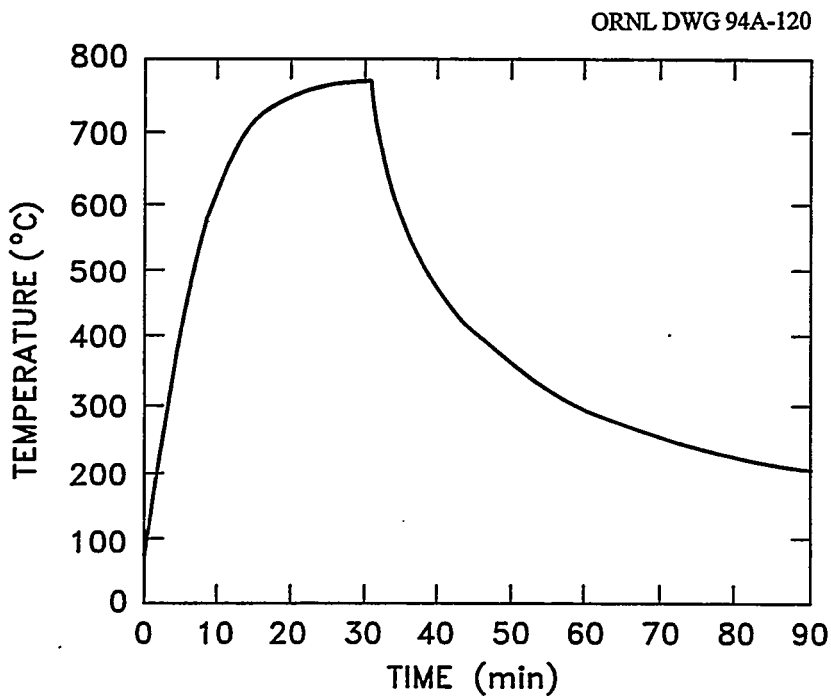

Fig. B.9. Model C—temperature at cask surface facing thermal shield, T_2, vs time.

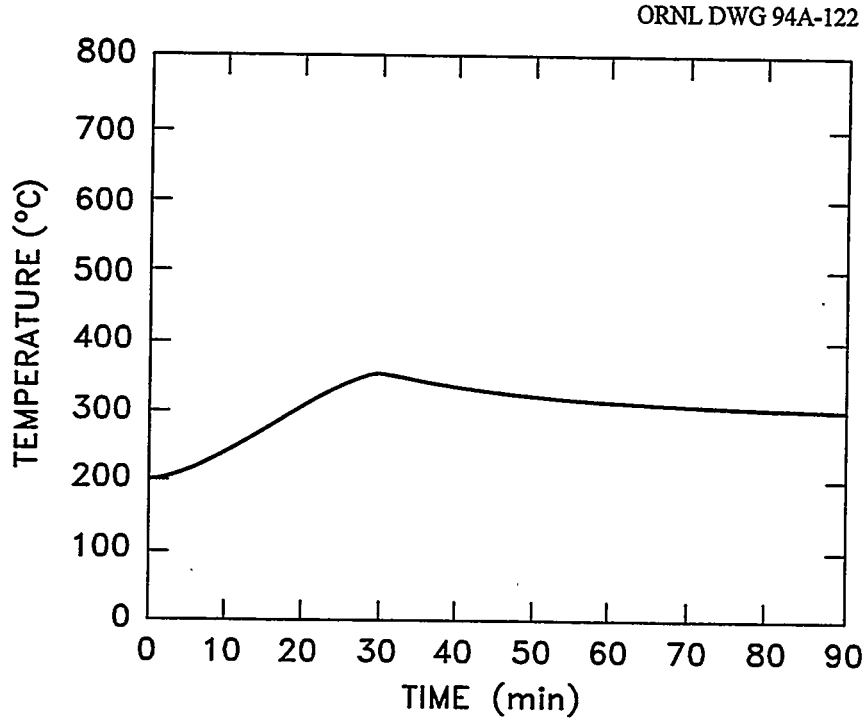

Fig. B.10. Model C—temperature at voided region interface, T_8, vs time.

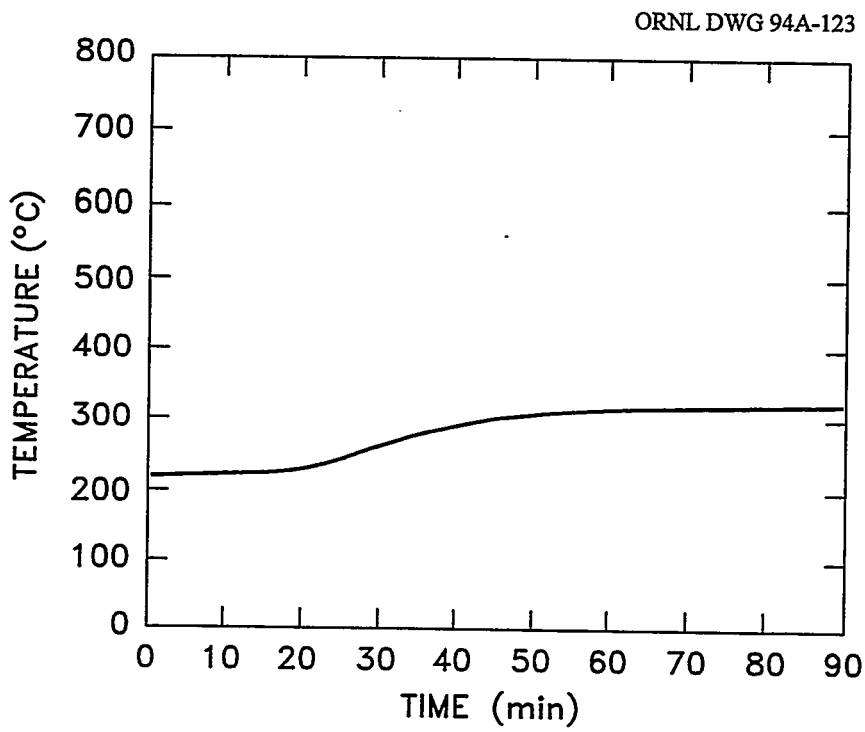

Fig. B.11. Model C—temperature at heat-generation-region interface, T_4, vs time.

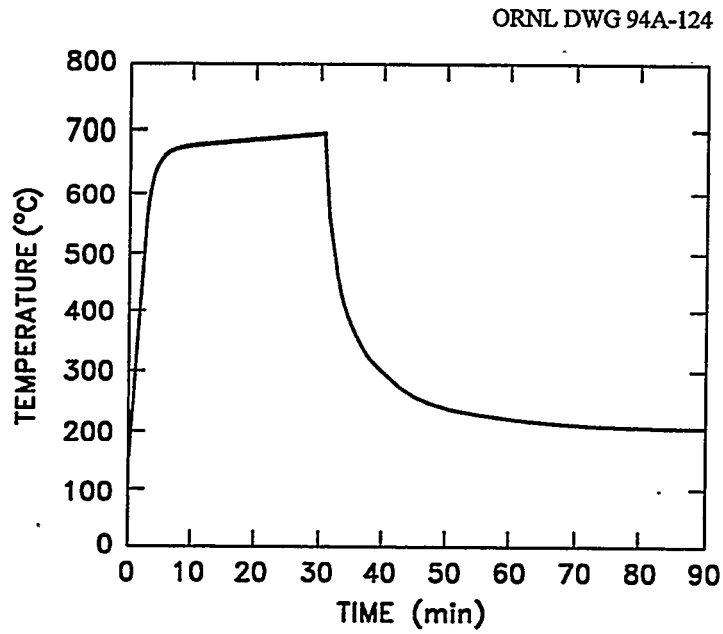

Fig. B.12. Model C—temperature at top of cask surface, T_6, vs time.

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Appendix C: A Statistical Technique for Determining Subcritical Limits

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C.1 DISCUSSION

In Sect. 10.3.2, a discussion is provided on the need for establishing a subcritical limit to be used in the criticality safety evaluation of a transportation package. In general, the recommended approach for establishing subcriticality based on the numerical calculation of the neutron multiplication factor, k_{eff}, is prescribed in Sect. 5.1 of ANSI/ANS-8.17.[1] The criteria to establish subcriticality requires that the calculated multiplication factor for a subcritical system, k_s, be less than or equal to an established maximum allowable multiplication factor based on benchmark calculations and uncertainty terms, that is,

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \,, \tag{C.1}$$

where

k_c = mean value of k_{eff} resulting from the calculation of benchmark criticality experiments using a specific calculational method and data;
Δk_s = uncertainty in the value of k_s;
Δk_c = allowance for uncertainties in the value of k_c (may include uncertainties in the calculational method and experimental data);
Δk_m = additional margin to ensure subcriticality.

For critical experiments, where $k_{eff} = 1$, the calculational bias (β) can be defined as $\beta = k_c - 1$ (such that overprediction of k results in a positive bias, and vice versa), and the uncertainty in the bias is identical to the uncertainty in k_c (i.e., $\Delta k_c = \Delta\beta$). Thus, the subcriticality condition may be rewritten as:

$$k_s + \Delta k_s + \Delta k_m - \beta + \Delta\beta \leq 1 \,. \tag{C.2}$$

The value of β is dependent on the set of specified experiments and the computational method and data that is applied. The value k_c and thus the bias β are not necessarily a constant over the range of a parameter of interest. If trends exist which cause the calculated values of k_{eff} to vary with one or more parameters (e.g., enrichment, moderation, etc.), then β can be estimated from a best fit for the calculated k_{eff} values, as a function of each of the parameters upon which it is dependent.

The uncertainty-$\Delta\beta$ is an accumulation of the uncertainty in the experiments, calculational method, and data. This uncertainty may include uncertainties in the critical experiments, statistical and/or convergence uncertainties in the benchmark calculations, uncertainties in the nuclear data, and uncertainties due to limitations or weaknesses in the geometrical or nuclear modeling of the critical experiments. Similarly, for a given subcritical system, there is an uncertainty Δk_s associated with the calculated k_{eff} value for the system, k_s. This uncertainty includes statistical/convergence and modeling uncertainties, as well as uncertainties in the material, fabrication, and dimensions of the system components.

The above discussion indicates that analysis of critical experiments can be used to determine an upper subcritical limit (USL) that defines a calculated subcritical limit for a specific analysis methodology within the range of applicable experiments. Thus, based on Eq. (2), the USL is the magnitude of the sum of the biases, uncertainties, and administrative and/or statistical margins applied to a set of critical benchmarks, such that with a high degree of confidence,

$$\text{USL} = 1 + \beta - \Delta\beta - \Delta k_m \,, \tag{C.3}$$

Because both β and $\Delta\beta$ can vary with a given system parameter, the USL can best be expressed as a function of the parameter within the parameter bounds that are defined by the set of critical experiments. For any given set of critical experiments, upper safety limits can be determined as a function of one or more key system parameters, such as the energy for the average lethergy causing fission (EALF), fuel enrichment, or fuel/moderator ratio. Extension of the USL outside the range of applicable parameters encompassed by the critical experiments would require consideration of additional uncertainties not provided in the discussion.

References 2–4 present a procedure for determining the USL based on the single-sided uniform width, closed-interval lower tolerance band, statistical approach of refs. 5 and 6. This approach is illustrated in Fig. C.1. In this figure, the upper line (solid) represents a linear regression fit $k_c(x)$ of the calculated k_{eff} results, for a series of critical experiments with varying values of the system parameter x (e.g., EALF, hydrogen-to-fissile ratio, etc.). As indicated in the figure, $\beta\,(x)$ is given as $k_c(x) - 1$. The middle line (long dashes) represents the lower confidence band for a single future calculation. The width of this band is determined statistically based on the existing data and a specified level of confidence; the greater the standard deviation in the data or the larger the confidence desired, the larger the band width will be. This confidence band, W, accounts for uncertainties in the experiments, the calculational approach, and the calculational data (e.g., cross sections). Thus W represents a statistical basis for $\Delta\beta$, the uncertainty in the value of β. The magnitude of W is defined for a confidence level of γ using the relationship:

$$W = \max\left\{ w(x)\,\big|_{x_{min},\,x_{max}} \right\} \,, \tag{C.4}$$

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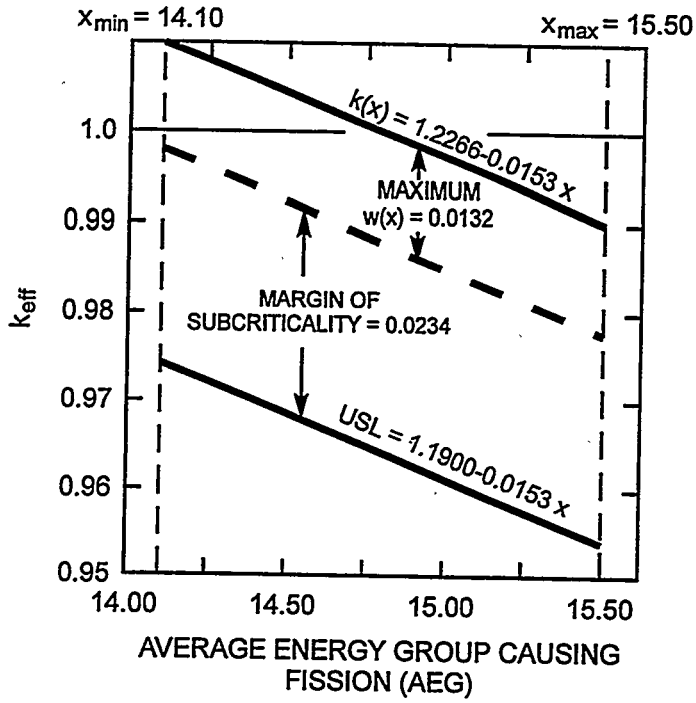

Fig. C.1. Confidence band and safety margin applied to a set of criticality calculations.

where

$$w(x) = t_\gamma s_p \left[1 + \frac{1}{n} + \frac{(x - \bar{x})^2}{\sum_{i=1,n} (x_i - \bar{x})^2} \right]^{\frac{1}{2}} \tag{6.5}$$

and

n = the number of critical calculations used in establishing $k_c(x)$;
t_γ = the Student-t statistic for a γ confidence and n - 2 degrees of freedom (i.e., data points);
$\bar{x}$ = the mean value of parameter x in the set of calculations; and
s_p = the pooled standard deviation for the set of criticality calculations.

The function $w(x)$ is a curvilinear function. For simplicity, it is desirable to obtain a constant width margin. Hence, for conservatism, the confidence band width W is defined as the maximum of $(w(x_{min}), w(x_{max}))$, where x_{min} and x_{max} are the minimum and maximum values of the independent parameter x, respectively. Typically, W is determined at a 95% confidence level.

The pooled standard deviation obtained from the pooled variance is given as:

$$s_p^2 = s_{k(x)}^2 + s_w^2 , \tag{C.6}$$

where $s_{k(x)}^2$ is the variance (or mean square error) of the regression fit, and is given by:

$$s_{k(x)}^2 = \frac{1}{(n-2)} \left[\sum_{i=1,n} (k_i - \bar{k})^2 - \frac{\left\{ \sum_{i=1,n} (x_i - \bar{x})(k_i - \bar{k}) \right\}^2}{\sum_{i=1,n} (x_i - \bar{x})^2} \right] , \tag{C.7}$$

with k = mean value of the k_i values obtained for the set of k_{eff} calculations, and s_w^2 is the within-variance of the data:

$$s_w^2 = \frac{1}{n} \sum_{i=1,n} \sigma_i^2 , \tag{C.8}$$

where σ_i is the standard deviation associated with k_i for a Monte Carlo calculation. For deterministic codes that do not have a standard deviation associated with a computed value, k_i, the standard deviation is zero. The term S_w^2 could also be used to incorporate known experimental uncertainties.

The bottom line (short dashes) in the Fig. C.1 represents the upper subcritical limit, and the additional margin of subcriticality provided by the distance between the middle and bottom line is the safety margin Δk_m defined by Eq. (1). Thus,

$$\mathrm{USL}(x) = k_c(x) - W - \Delta k_m . \tag{C.9}$$

As noted in Sect. 10.3.2, the value of Δk_m is typically assigned a fixed value of 0.05; however, a value for this margin can also be estimated statistically. This statistical estimation of Δk_m provides a quantitative measure to justify a Δk_m, rather than simply using an arbitrarily assumed value.

To obtain a statistics-based value for Δk_m, a method referred to as the lower tolerance band (LTB) approach is applied to determine a combined lower confidence band plus safety margin. The combined statistical lower bound is determined in terms of the product $C_{\alpha/\rho} \cdot s_p$, where s_p is the pooled variance of k_c described earlier, and $C_{\alpha/\rho}$ is a statistically determined multiplier. Thus, the upper subcritical limit is given as:

$$\mathrm{USL}(x) = 1.0 - (C_{\alpha/\rho} \cdot s_p) - \beta(x) . \tag{C.10}$$

$$= k_c(x) - (C_{\alpha,\rho} \cdot s_p)$$

The term $C_{\alpha/\rho} \cdot s_p$ provides a band for which there is a probability ρ that, with an α confidence, an additional calculation of k_{eff} for a critical system will lie within the band. For example, a $C_{95/99.5}$ multiplier produces a USL for which there is a 95% confidence that 995 out of 1000 future

calculations of critical systems will yield a value of k_{eff} above the USL. Calculation of the statistical multiplier $C_{\alpha/\rho}$ is described in refs. 5 and 6 and is beyond the scope of this discussion. In essence, this LTB approach provides a statistically based safety margin, Δk_m, which can be determined as the difference.

$$\Delta k_m = (C_{\alpha/\rho} \cdot s_p) - W . \tag{C.11}$$

A FORTRAN program is available that calculates $k_c(x)$, W, and $C_{\alpha/\rho} \cdot s_p$ for confidence and probability limits specified by the user and determines USL(x) based on Eq. (1). This code, called USLSTATS,[7] may be obtained through the following URL: http:www.cad.ornl.gov/cad_nea/text/scale-home.html. In criticality safety applications with a large number of applicable experiments, the USL(x) calculated by this program has generally provided a safety margin less than 0.05. However, remember that a minimum margin of subcriticality is needed that does not depend on the accuracy of the code or data. Since Δk_m would be zero if the analysis method had absolute accuracy, there is a need to set some minimum value for Δk_m. If only a fixed value for Δk_m is used, it is always advantageous to apply Eq. (C.11) in tandem to demonstrate that the fixed safety margin is conservative relative to the purely statistical basis that provides a quantitative measure of the confidence in the subcritical limits. This comparison is especially important when a limited number of data points are used in determination of $k_c(x)$, or when the calculated values have a large standard deviation.

As an example of this technique, assume that 29 critical experiments have been modeled and calculated. The calculated k_{eff} value, standard deviation, and the average energy group causing fission (AEG) are shown in Table C.1. The AEG is used here as the independent variable x. Table C.2 summarizes the various terms calculated to establish the USL and margin of subcriticality, with $\rho = 0.999$ and $\alpha = 0.95$. From these results, the USL is defined by the straight line USL = 1.1900 – 0.0153 x, and is statistically valid only between the range of AEG from 14.10 to 15.50 (see Fig. C.2). A system which is reasonably characterized by the experimental configurations and has a calculated $k_{eff} + 2\sigma$ that is below the USL should be adequately subcritical. To eliminate the effect of a calculated bias that raises the calculated k_{eff} value [e.g., $k_c(x) > 1$], the USL for values $x < 14.81$ [note $k_c(14.81) = 1.0$] could be set to the USL value at $x = 14.81$. In addition, if a minimum Δk_m of 0.05 is desired, the USL line should be dropped by 0.05 – 0.0234 = 0.0266 (i.e., USL = 1.1634 – 0.0153 x).

Note that ref. 4 recommends that positive biases [e.g., $k_c(x)>1$] be ignored as a prudent practice. If the package designer elects to include a positive bias, it is imperative that the reasons for the positive bias are well understood, the findings are consistent with other applications, and the system characteristics causing the positive bias exist in the actual package design.

Table C.1. Input data for example problem

k_{eff}	σ	AEG
0.99647	0.00337	14.82
0.99776	0.00326	14.81
1.00764	0.00311	14.83
0.99587	0.00365	14.44
0.99744	0.00327	14.28
1.00337	0.00335	14.84
0.99609	0.00395	14.73
1.00108	0.00378	15.08
0.99737	0.00325	15.20
0.98408	0.00342	15.31
0.98871	0.00361	15.26
0.99527	0.00292	15.50
0.98804	0.00273	15.49
1.01363	0.00401	14.36
1.01660	0.00445	14.36
1.00874	0.00485	14.36
1.01190	0.00479	14.38
1.00980	0.00498	14.35
1.00565	0.00397	14.10
1.01929	0.00407	14.12
1.00860	0.00411	14.10
0.99487	0.00462	15.04
0.99257	0.00382	14.90
1.00132	0.00450	14.90
0.99154	0.00420	14.90
1.00028	0.00374	15.43
0.99565	0.00413	15.44
0.98574	0.00415	15.43
0.98733	0.00416	15.43

Table C.2. Calculated terms for example problem

n	= 29
Linear regression, $k(x)$	= 1.226 – 0.015296 x
Minimum value of x	= 14.10
Maximum value of x	= 15.50
Average x (AEG),	= 14.8341
Average k_{eff},	= 0.99975
Variance of fit,	= 5.3718×10^{-5}
Within variance,	= 1.5455×10^{-5}
Pooled variance,	= 6.9174×10^{-5}
Pooled standard deviation, s_p	= 8.3171×10^{-3}
$C \cdot s_p$	= 0.0366
USL from Fig. (C.1)	= 1.1900 – 0.015296 x
Student-t @ $(n - 2)$ $(1 - \gamma)$	= 1.703
W	= 0.0132
Min. margin of subcriticality, $C \cdot s_p - W$	= 0.0234

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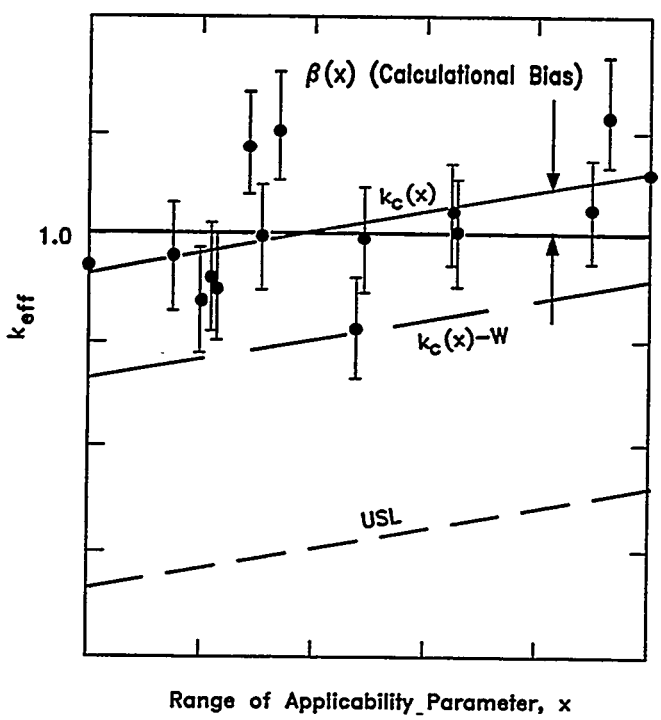

Fig. C.2. Typical results for the single-sided, uniform-width, closed-interval, LTB technique.

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Appendix D: A Partial Listing of Isotopic Source/Shield Design Compilation

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D.1 INTRODUCTION

This appendix presents a partial listing of an unpublished compilation of shielding data known as the Isotopic Source/Shield Design Compilation (IS/SDC). This compilation presented in Tables D.1 and D.2 lists radiation and thermal emission properties for some 30 isotopes and isotope pairs which have half-lives exceeding 10 d and is referenced in Sect. 9.4.1 of the shielding chapter. The various symbols listed in the table are defined as follows:

$\Gamma_\gamma(17,18)$	— the specific gamma-ray constant in rem-cm^2/(Ci-h) for photons with energies less than 100 keV
$\Gamma_\gamma(1\text{-}16)$	— the specific gamma-ray constant for photons with energies greater than 100 keV
Γ_x	— the specific gamma-ray constant for bremsstrahlung emission
E^γ_{eff}	— the effective gamma-ray, MeV
E^x_{eff}	— the effective bremsstrahlung energy, MeV
$E^{(x+\gamma)}_{eff}$	— the effective energy for gamma rays and bremsstrahlung emission combined
ϕ_{eff}/dis	— the effective number of gamma rays released per disintegration
MeV/dis	— the total gamma-ray energy released per disintegration
ϕ_{avg}/dis	— the average number of gamma rays released per disintegration
MeV/Ci-s	— the total gamma-ray energy released per curie second
x/β	— the function of the β particle energy released via bremsstrahlung
$\gamma + x$	— the effective number of photons released per disintegration for gamma ray and bremsstrahlung combined
Γ_n(SF)	— the neutron constant in rem-cm^2/(Ci-h) for neutrons produced by spontaneous fission
$\Gamma_n(\alpha,n)$Be	— the neutron constant in rem-cm^2/(Ci-h) for neutrons produced by (α,n) production in beryllium
Γ_γ(FP + SF)	— the specific gamma-ray constant in rem-cm^2/(Ci-h) for photons resulting from fission products and spontaneous fission.

Table D.1. Listing of shielding parameters of important isotopes: Part 1[a]

Isotope	Half-life	Specific activity Ci/g pure isotope or parent	Γ's (rem-cm²/(Ci-h))			E_{eff}^{γ}	$\frac{\phi_{eff}}{dis}$	γ emission rate			x/β	E_{eff}^{x}	$E_{eff}^{(x+\gamma)}$ γ+x (if important)		Shield thickness in cm Pb (tenth thickness) to meet shipping regs for 1000 Ci
			$\Gamma_\gamma(17,18)$	$\Gamma_\gamma(1\text{-}16)$	Γ_x			MeV/dis	$\frac{\phi_{avg}}{dis}$	$\frac{MeV}{Ci\text{-}s}$					
^{60}Co	5.271 year	1.131E+3		1.423E+4	2.3	1.263	1.984	2.5057	2.0	9.271E+10					21.4 (3.5)
^{22}Na	2.602 year	6.250E+3		1.2688E+4	5.6	1.102[b]	1.986	1.273[b]	1.0	4.71E+10[b]	3.04E−3	0.21			21.4 (3.72)
^{90}Sr–Y(Sr)	28.6 year	1.39E+2			12.3						1.093E−2	0.2			
(Y)	Equilibrium	1.39E+2			209.3						3.88E−2	0.93			11.5[c] (3.52)
^{99}Tc	2.13E+5 year	1.697E−2			5.7						7.75E−3	0.092			<0.5
^{106}Ru–Rh(Ru)	368.2 d	3.346E+3			Small										
(Rh)	Equilibrium	3.346E+3		1.354E+3	5.842E+2	0.941	0.211	0.590	0.338	7.38E+9	7.45E−2	1.40	1.22	0.216	17.7[c] (3.57)
^{110m}Ag	249.85 d	4.752E+3	3.2	1.669E+4	4.5	1.01	2.60	0.8205	3.193	9.69E+10	5.68E−3	0.088			16.0 (3.15)
^{125}Sb	2.77 year	1.033E+3	1.353E+3	3.142E+3	34.6	0.553	0.75	0.382	1.11	1.60E+10	3.06E−2	0.11			8.3 (1.6)
^{126}Sn–Sb(Sn)	1.0E+5 year	2.48E−2	1.464E+3		5.5	0.0815	0.693	0.0543	1.04	2.09E+9	7.00E−3	0.077			0.46 (0.085)
(Sb)	Equilibrium	2.48E−2	42.3	1.836E+4	75.8	0.799	3.459	0.639	4.32	7.41E+11	2.94E−2	0.35			14.45 (2.33)
^{126}Sb	12.4 d	8.358E+4	42.3	1.836E+4	75.8	0.799	3.459	0.639	4.32	7.41E+11	2.94E−2	0.35			14.45 (2.33)
^{134}Cs	2.062 year	1.295E+3	20.8	9.961E+3	12.0	0.89	1.748	0.6995	2.236	5.75E+10	1.06E−2	0.18			14.5 (2.72)
^{137}Cs–Ba(Cs)	30.17 year	86.56			25.6						1.38E−2	0.21			
(Ba)	Equilibrium	86.56	200	4.032E+3		0.6615	0.904	0.6082	0.983	2.21E+10					10.5[c] (2.38)
^{140}Ba–La(Ba)	12.789 d	7.294E+4	467	1.241E+3	51.7	0.5015	0.375	0.255	0.738	6.96E+9	2.00E−2	0.295			6.5
(La)	Equilibrium	7.294E+4	45.8	1.24E+4	153	1.63	1.419	1.067	2.167	8.55E+10	3.48E−2	0.60			28.1[c] (4.78)
^{144}Ce–Pr(Ce)	284.3 d	3.19E+3	268	122	6.72	0.1276	0.151	0.0828	0.232	7.11E+8	8.55E−3	0.062			
(Pr)	Equilibrium	3.19E+3		168.2	471	1.883	0.0168	1.239	0.0255	1.17E+9	7.25E−2	1.20	1.279	0.0863	20.0[c] (5.36)
^{152}Eu	13.6 year	1.731E+2	2.033E+3	6.728E+3	14.0	1.205	0.906	0.44	2.48	4.04E+10	1.95E−2	0.34			19.9 (3.5)
^{154}Eu	8.8 year	2.64E+2	7.045E+2	7.165E+3	33.0	1.065	1.17	0.64	1.95	4.62E+10	1.28E−2	0.30			18.4 (3.3)

[a]The table is extracted from an unpublished document entitled *Isotope Source/Shielding Design Compilation* by E. D. Arnold.
[b]Includes annihilation contributions.
[c]The indicated isotope from an isotope pair controls the contribution to the dose rate.

Table D.2. Listing of shielding parameters of important isotopes: Part 2[a]

Isotope	Half-life	Specific activity Ci/g pure isotope or parent	Γ's (rem-cm²/(Ci-h))			E^{γ}_{eff}	$\frac{\phi_{eff}}{dis}$	γ emission rate			x/β	E^{x}_{eff}	$E^{(x+\gamma)}_{eff}$ γ+x (if important)		Shield thickness in cm Pb (tenth thickness) to meet shipping regs for 1000 Ci
			$\Gamma_{\gamma}(17,18)$	$\Gamma_{\gamma}(1\text{-}16)$	Γ_{x}			MeV/dis	$\frac{\phi_{avg}}{dis}$	$\frac{MeV}{Ci\text{-}s}$					
^{144}Pm	363 d	2.498E+3	2.149E+3	1.047E+4		0.6444	2.415	0.466	3.34	5.76E+10					11.0 (1.94)
^{147}Pm	2.623 year	9.278E+2			38.75						4.93E-3	0.069			<0.5
^{146}Pm	2020 d	4.428E+2	1.349E+3	4.888E+3	28.7	0.646	1.18	0.435	1.728	2.78E+10	4.6E-3	0.185			11.0 (1.94)
^{148}Pm	41.3 d	2.137E+4	258	1.262E+4	18.5	0.703	2.83	0.597	3.33	7.36E+10	1.64E-2	0.11			12.0 (2.11)
^{192}Ir	74.02 d	9.189E+3	103	6.226E+3	38.3	0.4455	1.84	0.347	2.36	3.03E+10	3.39E-2	0.13			5.6 (1.17)
^{198}Au	2.696 d	2.446E+5	12.9	3.358E+3	86.4	0.419	0.967	0.402	1.008	1.15E+10	4.65E-2	0.23			5.3 (1.05)
^{194}Hg-Au[b]	1.90 year	9.707E+2	620	5.968E+3	168.9	1.292	0.84	0.443	2.44	4.0E+10	4.89E-2	0.47			20.8 (3.75)
^{208}Tl	3.053 mile	2.96E+8		1.6231E+5	769.1	2.508	1.46	1.523	2.38	1.35E+11	8.02E-2[c]	0.59			31.0 (5.09)
^{208}Tl[b]	3.62 d	5.78E+4		5.832E+4	276.1	2.508	1.46[b]	1.523	2.38[b]	4.847E+10	2.88E-2				28.7 (5.09)
^{224}Ra chain	3.62 d	1.611E+5	1.027E+3	7.8675E+4	614.1	2.385	1.79	1.648	2.59	6.10E+10					28.7 (5.09)
^{222}Rn chain	3.8235 d	1.539E+5	572	9.812E+3	190.4	1.60	1.101	1.6995	1.037	6.29E+10					25.1 (4.41)
^{238}Pu	87.75 year	1.713E+1	319.2			0.0136	0.116	0.0136	0.116	5.84E+7					
^{241}Am	458 year	3.2415	1.535E+3			0.0564	0.497	0.0344	0.814	1.04E+9					
^{242}Cm	163.2 d	3.307E+3	3.16E+3			0.0143	0.116	0.0143	0.116	6.08E+7					
^{244}Cm	18.11 year	80.97	283.5			0.0143	0.103	0.0143	0.103	5.45E+7					
^{252}Cf	2.639 year	5.38E+2	201.0			0.015	0.073	0.015	0.073	4.05E+7					26.3 (5.15)

[a]The table is extracted from an unpublished document entitled *Isotope Source/Shielding Design Compilation*, E. D. Arnold.
[b]Hg half-life with ^{194}Au activity data.
[c]Per Tl beta particle.
[d]Activity as part of ^{224} Ra decay chain.
[e]Particles per Tl decay chain.

Appendix E: The Standardized Computer Analysis for Licensing Evaluation (SCALE) Program

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E.1 BACKGROUND

The history of the Standardized Computer Analyses for Licensing Evaluation (SCALE) code system dates back to 1969 when what is now the Computational Physics and Engineering Division at Oak Ridge National Laboratory (ORNL) began providing the transportation package certification staff at the U.S. Atomic Energy Commission with computational support in the use of the new KENO code for performing criticality safety assessments with the statistical Monte Carlo method. From 1969 to 1976, the certification staff relied on the ORNL staff for assistance in the correct use of codes and data for criticality, shielding, and heat transfer analyses of transportation packages. However, the certification staff learned that, with only occasional use of the codes, it was difficult to become proficient in performing the calculations often needed for an independent safety review. Thus, shortly after the move of the certification staff to the U.S. Nuclear Regulatory Commission (NRC), the NRC staff proposed the development of an easy-to-use analysis system that provided the technical capabilities of the individual modules with which they were familiar. With this proposal, the concept of the SCALE code system was born.

The NRC staff provided ORNL with the following general development criteria for SCALE: (1) focus on applications related to nuclear fuel facilities and package designs, (2) use well-established computer codes and data libraries, (3) design an input format for the occasional or novice user, (4) prepare "standard" analysis sequences (control modules) that will automate the use of multiple codes (functional modules) and data to perform a system analysis, and (5) provide complete documentation and public availability. With these criteria, the ORNL staff laid out the framework for the SCALE system and began development efforts. Early papers on the development of SCALE from both NRC and ORNL perspectives are provided in refs. 1 and 2. The initial version of SCALE was released by the Radiation Safety Information Computational Center (RSICC) at ORNL in 1980. Since that time, the system has been considerably enhanced,[3–5] and the current release is Version 4.2 (ref. 6). Version 4.3 was released in the summer of 1995.

E.2 SYSTEM OVERVIEW

The original concept of SCALE was to provide standardized sequences in which the user had few options with regard to the geometry model employed, materials of construction used, and the type

of analysis carried out. Input for the control modules was designed to be free-form with extensive use of key words and engineering-type input requirements. The more flexible functional modules have a more difficult input logic and require the user to interface the data files necessary to run the modules in a stand-alone fashion. As the system has grown in popularity over the years and additional options have been requested, the control modules have been improved to provide experienced users additional access to the numerous capabilities within the functional modules. However, the most important features of SCALE remain its ability to accept simplified user input and to minimize the effort required to (1) prepare material mixtures and (2) perform adequate problem-dependent, cross-section processing.

The modules available in Version 0 of SCALE were for criticality safety analysis sequences (CSAS), which provided automated material and cross-section processing prior to a one-dimensional (1-D) or multidimensional criticality analysis. Since that version, the capabilities of the system have been significantly expanded to provide additional CSAS capabilities, new shielding analysis sequences (SAS) that also include depletion/decay capabilities for spent fuel characterization, and a heat transfer analysis sequence (HTAS). At the center of the CSAS and SAS sequences is the library of subroutines referred to as the Material Information Processor Library or MIPLIB. The purpose of MIPLIB is to allow users to specify problem materials using easily remembered and easily recognizable key words that are associated with mixtures, elements, and nuclides provided in the Standard Composition Library (SCL). MIPLIB also uses other key words and simple geometry input specifications to prepare input for the following modules that perform the problem-dependent, cross-section processing: BONAMI, NITAWL-II, and XSDRNPM. A key word supplied by the user selects the cross-section library from a standard set provided in SCALE or designates the reference to a user-supplied library. Several utility modules from AMPX[7] have been added to the SCALE system to provide users with the capability to edit the cross-section data and reformat user-supplied libraries for use in SCALE.

Over the history of the project, several modules have been removed from the system because they are no longer supported by the development staff at ORNL. Tables E.1 and E.2 provide a summary of the major applications of each of the control modules and functional modules in Version 4.3 of the SCALE code system. The control modules were designed to provide the system analysis capability originally requested by the NRC staff. The CSAS module contains eight sequences that enable general analysis of a 1-D system model or a multidimensional system model, capabilities to search on geometry spacing or nuclide concentration, and problem-dependent, cross-section processing for use in executing stand-alone functional modules. The SAS1 and SAS3 modules provide general 1-D deterministic and 3-D Monte Carlo analysis capabilities, respectively. The SAS2 module was originally developed to perform a depletion/decay calculation to obtain spent fuel radiation source terms that were subsequently input automatically to a 1-D, radial shielding analysis in a cylindrical geometry. Over time, the depletion/decay portion of the SAS2 module has been significantly enhanced, and interfacing to the other shielding modules has been provided. The SAS4 module enables automated particle biasing for a Monte Carlo analysis of a transportation package-type geometry. The HTAS1 module is the only heat-transfer control module; it uses the various capabilities of the HEATING code to perform different sequences of steady-state and transient analysis that enable the normal and accident conditions of a transportation package to be evaluated. Like SAS4, the HTAS1 module is limited to a package-type geometry.

Table E.1. Analysis capabilities summary of the SCALE control modules

Control module	Analysis function(s)	Functional modules executed
CSAS	1-D deterministic calculation of neutron multiplication 3-D Monte Carlo calculation of neutron multiplication Problem-dependent, cross-section processing Multiplication search on spacing	BONAMI NITAWL-II XSDRNPM KENO V.a ICE
CSAS6	3-D Monte Carlo calculation of neutron multiplication for complex geometries	BONAMI NITAWL-II XSDRNPM KENO-VI
SAS1	1-D deterministic calculation of radiation transport through shield and dose evaluation at a point Calculation of dose at detector based on leakage from critical volume	BONAMI NITAWL-II XSDRNPM XSDOSE
SAS2H	Point depletion/decay of nuclear fuel Spent fuel characterization Source-term and/or decay-heat generation 1-D radial shielding analysis in cylindrical geometry	BONAMI NITAWL-II XSDRNPM COUPLE ORIGEN-S XSDOSE
SAS3	Dose evaluation using MORSE 3-D Monte Carlo code	BONAMI NITAWL-II XSDRNPM MORSE-SGC
SAS4	Calculation of dose outside of transportation package using MORSE code and automated biasing techniques	BONAMI NITAWL-II XSDRNPM MORSE-SGC
QADS	3-D point-kernel gamma-ray shielding analysis	QAD-CGGP
HTAS1	R–Z steady-state and transient analyses of a transportation package	OCULAR HEATING 7.2

Version 4.3 of the SCALE system was developed to facilitate portability among various computing platforms. The system is maintained and enhanced at ORNL under quality assurance (QA) and configuration management plans. The system is routinely tested on an International Business Machine RS/6000 workstation. In addition, the system has been tested at ORNL on Digital Equipment Corporation (DEC), Sun Micro-Systems, Inc. (SUN), and Hewlett Packard (HP) workstations. Information needed to install SCALE on each of these systems is included with the software package distributed by the code center. A separate SCALE software package, designated

Table E.2. Analysis capabilities summary of the SCALE functional modules

Module	Function
BONAMI	Resonance self-shielding of cross sections with Bondarenko factors
NITAWL-II	Resonance self-shielding of cross sections with resolved resonance data
XSDRNPM	General 1-D, discrete-ordinates code for: · zone-weighting of cross sections · eigenvalue calculations for neutron multiplication · fixed-source calculation for shielding analysis · adjoint calculation for determining importance functions
XSDOSE	Module for calculation of dose at a point based on the 1-D leakage flux from a finite shield
COUPLE	Interface module for preparation of cross-section and spectral data for ORIGEN-S
ORIGEN-S	General-purpose point-depletion and decay code to calculate isotopic, decay heat, radiation source terms, and curie levels
ICE	Cross-section utility module for mixing cross sections
MORSE-SGC	Monte Carlo code with combinatorial and array geometry features used to perform radiation shielding analysis
HEATING7.2	Finite-volume, multidimensional code for conduction and radiation heat transfer
KENO V.a	Monte Carlo code for calculation of neutron multiplication factors
KENO-VI	Monte Carlo code for calculation of neutron multiplication factors for complex geometries
QAD-CGGP	3-D point-kernel gamma-ray shielding analysis
OCULAR	Calculation of radiation exchange factors

SCALE-PC, is also available for execution of the SCALE system on a personal computer (PC). SCALE-PC also includes a sequence called ORIGEN-ARP, which allows the stand-alone execution of ORIGEN-S for spent fuel depletion/decay using a full-screen input processor and an automatic interpolation on database of burnup-dependent, cross-section libraries (see Sect. F7.A of ref. 6).

E.3 DESCRIPTION OF MODULES

E.3.1 Functional Modules

The major functional modules in SCALE-4.3 are listed in Tables E.1 and E.2. The SCALE functional modules were selected on a basis of both their analytical capabilities and their proven records of reliable performance. A brief summary of the capabilities of each functional module is provided below.

- NITAWL-II applies the Nordheim Integral Technique to perform neutron cross-section processing in the resonance energy range. This technique involves a fine energy group calculation of the slowing-down flux across each resonance with subsequent flux weighting of the resonance cross sections. The major function of NITAWL is its conversion of cross-section libraries from a problem-independent to a problem-dependent form. However, NITAWL also assembles group-to-group transfer arrays from the elastic and inelastic scattering components and performs other tasks in producing the problem-dependent working library.

- BONAMI performs resonance shielding through the application of the Bondarenko shielding factor method. As input, the program requires the presence of shielding factor data on the AMPX master library interface. As output, BONAMI produces a problem-dependent master library. Thus, in the SCALE sequences, it is always used in conjunction with NITAWL-II which, even if no resonance processing is done, converts the master library into an AMPX working library.

- ICE is used primarily in SCALE to mix microscopic cross-section data from an AMPX working library and to correct the results to macroscopic data in one of several available formats.

- COUPLE is used in the SCALE analytic sequences to update ORIGEN-S libraries with flux-weighted cross sections provided by XSDRNPM. COUPLE uses the multigroup fluxes from XSDRNPM to calculate THERM, RES, and FAST, the spectral parameters used in ORIGEN-S. This calculation is performed in a manner that is consistent with the use of the data in the ORIGEN-S library for those nuclides that are not being updated with weighted cross sections from XSDRNPM. COUPLE can also be used to create and/or update binary libraries for ORIGEN-S.

- KENO V.a is a multigroup Monte Carlo code employed to determine effective multiplication factors (k_{eff}) for multidimensional systems. The basic geometrical bodies allowed for defining the model are cuboids, spheres, cylinders, hemispheres, and hemicylinders. KENO V.a differs from KENO IV in that it (1) has an enhanced geometry package that allows arrays to be defined and positioned throughout the model, (2) has a P_n-scattering treatment, (3) allows extended use of differential albedo reflection, (4) generates printer plots for checking the input model, (5) places energy-dependent data into supergroups, (6) has an improved restart capability, and (7) allows origin specifications to be made for spheres, cylinders, hemicylinders, and hemispheres.

- KENO-VI is a new version of the KENO Monte Carlo code designed for criticality analysis of more geometrically complex systems. It has the following features: intersecting geometry

regions; hexagonal as well as cuboidal arrays; regions, holes, arrays, and units rotated to any angle and truncated to any position; and the use of an array boundary that intersects the array. In KENO-VI, units can be constructed using both simple geometric shapes and tailored geometric shapes constructed using quadratic equations. Two-dimensional color plots of the geometry model can be obtained on workstation and PC platforms using Version 4.3.

- XSDRNPM is a 1-D, discrete-ordinates transport code for performing neutron or coupled-neutron-gamma calculations. The code has a variety of uses within SCALE: preparation of cell-averaged cross sections for subsequent system analysis, 1-D criticality safety and radiation shielding analysis, and generation of a neutron spectrum used to develop spectral parameters for ORIGEN-S (via COUPLE). The SAS4 control module also uses XSDRNPM to generate 1-D adjoint functions for use in preparing biasing parameters for MORSE-SGC.

- MORSE-SGC is the SCALE version of the MORSE family[9] of Monte Carlo programs for radiation shielding analysis. This version uses the MARS geometry system with combinatorial geometry. This geometry package allows ease in modeling multiple-array systems by using a repeating array feature along with an unlimited (except by computer core) array-nesting capability. MORSE-SGC incorporates supergroup cross-section storage and tracking.

- XSDOSE is a code used in conjunction with XSDRNPM to compute the neutron and gamma flux and the resulting dose at various points outside a finite cylinder or sphere. It may also be used to compute the flux or dose at various points on a finite rectangular surface source or a circular disc. The code assumes that the outgoing angular flux distribution on the rectangle, cylinder, sphere, or disc is independent of position and that the surrounding media is a void. The numerical technique employed in XSDOSE is suitable for points on, close to, or far from the source.

- ORIGEN-S is an updated version of the ORIGEN code[10] with flexible dimensioning and free-form input processing. One of the primary objectives in developing ORIGEN-S was that the calculations be able to use multienergy-group neutron flux and cross sections in any group structure. Use of the multigroup data is automated via the COUPLE code. ORIGEN-S performs point-depletion and decay analyses to obtain isotopic concentrations, decay heat source terms, and radiation source spectra and strengths for use in subsequent system analyses. A PC-based, full-screen input processor for depletion/decay of isotopes in spent fuel using existing cross-section libraries is available (ref. 11). This processor, combined with ORIGEN-S and a new procedure for interpolating between existing cross-section libraries make up the new ORIGEN-ARP depletion/decay sequence that is now available in SCALE-PC for easy and accurate stand-alone use of ORIGEN-S.

- HEATING7.2 is a 3-D, finite-volume code for solving general heat conduction problems. Steady-state or transient problems can be analyzed using a variety of boundary conditions and heat-generation specifications. HEATING7.2 provides for direct input of mode-to-mode connectors that allow multidimensional radiative heat transfer to be modeled. Plot files that interface with commercially available graphics packages are produced by HEATING7.2.

- QAD-CGGP is a 3-D, point-kernel code to calculate gamma-ray penetration through various shielding materials and configurations. It employs combinatorial geometry and geometric progression buildup factors.

- The PICTURE module provides a printed view of arbitrary 2-D slices through a MORSE or QAD-CGGP geometry model. By inspecting these pictures, one may determine if the geometry specified is the desired geometry.

- OCULAR is a radiation-exchange factor module compatible with the axisymmetric r–z geometry of HEATING. Gray-body exchange factors and geometric configuration factors may be generated by OCULAR for use in HEATING.

E.3.2 Control Modules

As noted in Sect. E.1, the modular structure of SCALE allows back-to-back execution of the functional modules to perform a system analysis. However, a variety of control modules have been developed that automate and standardize various analytic sequences. The control-module-input format has been designed to help minimize input errors. Upon processing the user-specified input, the SCALE system control modules immediately print an input check list in which the user (or reviewer) can easily establish that the input describes the system to be analyzed.

The control modules shown in Table E.1 are those developed to date. Although the control modules are typically designated by their principal analytic sequence, note that more than one sequence may exist within a control module.

- CSAS is the control module designed for executing all the criticality analysis sequences. The analytic sequence designated by =CSAS25 follows the execution path BONAMI-S, NITAWL-II, and KENO V.a. The sequence =CSAS4 follows the same execution path but repeats the KENO V.a. analysis as part of a search for the geometry or nuclide concentration that provides the maximum or specified k_{eff} value. The previous sequences are modified to include cell-weighting with XSDRNPM-S if =CSAS2X or =CSAS4X is designated. Other sequences are available to obtain k_{eff} via a 1-D XSDRNPM model or simply prepare problem-dependent, cross-section processing for subsequent use in analyses with individual codes. A PC-based, full-screen input processor for preparing CSAS input is available in SCALE-4.2 (see ref. 12).

- SAS1 is a control module for performing data processing, radiation-shielding analysis, and dose evaluation on systems which can be adequately modeled in 1-D geometries. The execution path includes BONAMI, NITAWL-II, XSDRNPM, and XSDOSE. Cell weighting is also allowed if =SAS1X is specified rather than =SAS1. A sequence to obtain the dose from a critical geometry is also available.

- SAS2H is a control module that processes fuel assembly cross sections, computes photon and neutron source spectra, and evaluates dose rates from spent fuel casks using a 1-D, transport-shielding analysis. The execution includes repeated passes through BONAMI, NITAWL-II, XSDRNPM, COUPLE, and ORIGEN-S for cross-section processing and fuel burnup; radiation source computations; radial shipping-cask-shielding analysis applying the calculated spent fuel composition and sources; and the final determination of dose rates by XSDOSE from the angular flux leakage. The neutron flux used to update the cross sections during the depletion is based on a 1-D model of the fuel assembly. Typically, SAS2H is used only for the depletion/decay portion of the analysis and SAS1 or SAS4 is used for the shielding evaluation.

- SAS3 performs data processing and radiation shielding analysis on systems for which the user specifies the radiation source distribution and which must be modeled in 3-D geometry. The execution path includes NITAWL-II, XSDRNPM (if lattice cell, cross-section processing is specified), ICE, and MORSE-SGC.

- SAS4 is a control module that allows calculation of radiation doses exterior to a transport/storage cask via a 3-D Monte Carlo analysis. The sequence execution path includes BONAMI, NITAWL-II, and XSDRNPM (if lattice-cell geometry is specified) for cross-section processing, and XSDRNPM for evaluation of the adjoint flux used by the sequence for automatic generation of Monte Carlo biasing parameters; and MORSE-SGC for the radiation transport and dose calculation.

- QADS is the control module designed to perform a 3-D, point-kernel estimation of gamma transport using QAD-CGGP. QADS was designed to capture the flexibility and power of the QAD technique for problems amenable to point-kernel solution while allowing for easy-to-use input interface. The SCALE Standard Composition Library is used to allow simplified input of materials. The problem geometry is coded using the well-known MORSE combinatorial geometry package. Finally, the dose portion of the input follows the XSDOSE input data format in SAS1. The combined use of the SCALE standard compositions and free-form input with the multidimensional geometry capabilities and generally short running time of point-kernel techniques produces a very powerful procedure for shielding analysis of a wide variety of problems.

- HTAS1 generates the necessary HEATING7.2 input and automatically manipulates the module to perform a 2-D (r–z) thermal analysis for a specific class of shipping containers during the normal, fire, and post-fire conditions specified by the licensing regulations.

E.3.3 Data Libraries

The SCALE code system includes several problem-independent, multigroup cross-section libraries. The most popular libraries have been the 27-group neutron library[13,14] and the $27n$–18γ library based on ENDF/B-IV[14]. More general fine-group and associated broad-group neutron libraries based on ENDF/B-V were released with SCALE-4.3 (refs. 15 and 16). Other data libraries included in SCALE are the ORIGEN-S nuclear data libraries (decay data, branching ratios, etc.), the ORIGEN-S card-image cross-section libraries, a thermal properties database, and a database of standard composition materials.

E.3.4 Other SCALE Components

Several SCALE components cannot be readily classified as functional or control modules. For example, the Material Information Processor is a library of subroutines assessed by all the control modules, except HTAS1. The processor reads the material-related information, accesses the SCALE system standard compositions library, and calculates isotopic atom densities. It also prepares the physics parameters required by the resolved resonance analysis in the NITAWL-II module. For example, the Material Information Processor includes a calculation of the Dancoff factor for problems in which a lattice-cell-geometry option has been specified. The MARS geometry package is also a set of subroutines developed for use with the MORSE-SGC code

system and to allow easy input specification of combinatorial geometry models containing multiple or nested arrays.

The Subroutine Library available in SCALE contains a number of useful subroutines that can be accessed by the functional or control modules. The SCALE free-form reading routines, a random number generator, and timing routines are examples of the routines in the library.

The system driver is the foundation upon which the modular features of SCALE are based. The driver facilitates module interfaces and ensures that minimum storage and computational times are used in the SCALE analyses.

Also included in the latest SCALE-4 release are a number of AMPX utility modules that perform invaluable operations for building and maintaining the SCALE cross-section data libraries.

E.4 VERIFICATION AND VALIDATION

Verification and validation are important aspects of the maintenance and update activities associated with the SCALE code system. Numerous ORNL studies have been performed to demonstrate proper operation of the system (ref. 5 provides a list of references associated with each technical area). More often than not, verification work is reported in internal ORNL documentation associated with configuration control of the system.

The SCALE code system has also been verified and validated by numerous organizations before they used the system to perform safety analyses on shipping packages. Early ORNL validation work comparing analysis results with critical experiments is documented in ref. 13. More recent validation work for criticality safety using ENDF/B-V data is reported in ref. 16 and 17. Recent validations of the shielding analysis sequences and the SAS2H sequence have been documented in refs. 18 and 19, respectively. Reports that give guidance on verification and validation problems that can be used during system installation have recently been published.[20-21]

E.5 AVAILABILITY

The SCALE code system and related software have been packaged by the Radiation Safety Information Computational Center (RSICC) and may be obtained by contacting

Radiation Safety Information Computational Center
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6362
Telephone: (423) 574-6176
FAX: (423) 574-6182

E.6 SUMMARY

As shown in this appendix, the SCALE system has a wide range of capabilities. The SCALE modules used for preparing problem-dependent, cross-section libraries and performing criticality

safety analyses are well established and in routine use by much of the United States and foreign criticality safety communities as the primary computational tool or as a backup/review tool. Applications extend well beyond cask analysis to the areas of fuel-reprocessing, -handling, and -storage facilities; critical experiment design and analysis; and selected reactor applications. The use of SCALE to characterize spent fuel and to perform heat transfer and shielding analyses has increased since the release of SCALE-4.0 in 1990. The improvement in system installation procedures and portability to a variety of system platforms will continue to improve the flexibility of SCALE and its usefulness by package designers.

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