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A REVISION OF THE *CASK DESIGNERS GUIDE* FOR THE '90s

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INTRODUCTION

The report *A Guide for the Design, Fabrication, and Operation of Shipping Casks for Nuclear Applications*, ORNL-NSIC-68, commonly called the *Cask Designers Guide*, is being revised at the request of the Transportation and Packaging Safety Division of the Department of Energy (DOE). The new document will be called the *Packaging Handbook*. The *Cask Designers Guide* was published in 1970 during the period when many radioactive materials packagings were being developed and many technical studies applicable to these packagings were being performed. Since that period, many improvements in packaging design have appeared, designers have improved their calculational techniques, and much effort has gone into applying Quality Assurance (QA) principles to cask development. Materials, and their limitations, have surfaced as a very important consideration in the licensing process. While the *Packaging Handbook* considers all Type B packages, most of the authors' experience lies in the technical areas found in the licensing of spent nuclear fuel (SNF) packagings and this is reflected in the document.

The *Packaging Handbook* has one primary goal: to provide sufficient information and guidance to improve the quality of Safety Analysis Reports for Packaging (SARPs) for Type B (including fissile) package designs that are submitted to DOE for certification. This is being accomplished by utilizing a group of experts that have contributed to the development of one or more SARPs and have submitted them to either DOE or the Nuclear Regulatory Commission (NRC) for certification or have recently interacted with others in the licensing process. Their experience should provide the best guidance to minimize the questions raised during the certification process and maximize the likelihood of successfully obtaining a certificate. The report also discusses information on package fabrication, quality assurance (QA), SARP preparation, certification, use, and maintenance, particularly as related to the licensing process.

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TOPICS OF THE HANDBOOK

The handbook will contain 13 technical chapters, and while most of these address the technical aspects of packaging design, several address related topics that should influence the design, or the way the design is evaluated.

One chapter, **Package Certification**, will examine the problem of obtaining a proper packaging to transport radioactive materials. The first activity is to determine if an existing licensed packaging can accommodate the material to be shipped. It is possible that such a package might technically be suitable but not certified for the material in question, in which case the licensing body could be petitioned to modify the certificate appropriately. If such packagings do not exist, then a second activity would involve defining the functional requirements of the container. The chapter further discusses the preparation of the SARP in accordance with the outline found in the NRC Regulatory Guide 7.9 and the process through which Certificates of Compliance are granted.

The chapter **Regulations and Standards** will discuss the regulatory philosophy applied to radioactive materials transport which requires that the packaging provide the primary protection, with minimal reliance on operational controls or human intervention. The regulations are based on a graded approach which has been applied to the level of performance required of radioactive materials packages. This level of performance is affected by the potential hazard presented by the contents. The chapter provides an overview of the regulations and describes the interactions among the regulatory bodies both within and external to the United States. While this chapter provides a summary of the national and international standards applicable to both Type A and Type B packagings, the development of Type B packages is emphasized.

The **Design and Demonstration Process**, as applied to packaging approval, is discussed in a separate chapter. Radioactive materials transportation packages are highly integrated systems with each component performing multiple functions involving strength, shielding, criticality control, heat transfer, etc. As such, the multiple disciplines participating in the design and demonstration process must be carefully controlled to ensure both rapid convergence of the design process and preservation of a logically consistent demonstration effort. This calls for a "top-down" design approach with both demonstration rationale and design criteria established, in a mutually supportive fashion, when the design project is initiated. Both criteria and demonstration rationale should initially focus upon the four primary performance requirements for radioactive materials packages, including (1) containment of the radioactive materials within the package itself; (2) criticality control of the contained radioactive materials; (3) external radiation controls to limit public exposure, and (4) heat management.

The chapter on **Materials** is important because in recent years materials has surfaced as a key topic in the certification process. NRC 10 CFR 71, Paragraph 71.33(a)(5) requires that information regarding materials of construction be furnished in sufficient detail to provide a basis for evaluation of the packaging. Furthermore, paragraph 71.37(b) requires, in part, that the applicant identify any established codes and standards proposed that are applied or utilized in the packaging 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 and approved by professional societies should be used [e.g., the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC)]. This ensures that the material properties that serve as a basis for the various safety analyses will be uniform. 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, in addition, the means by which quality is assured and the effect of fabrication processes on these properties should be addressed.

The transportation package components that function to ensure the containment of the contents are recommended to conform to Section III of the B&PVC, Subsection NB as noted in Table 1. All transportation package components that function to ensure that the package remains subcritical should conform to Section III, Subsection NG, which was originally prescribed for core support and internal structures of a nuclear reactor system. Those transportation packaging components that function to guarantee adequate shielding of the radiation emitted by the package contents and other safety-related functions should conform to Section III, Subsection NF or Section VIII, Division 1 of the B&PVC.

Fabrication Criteria Based on ASME B&PVC or Other National/International Codes	
Component Safety Designation	Applicable Code (Section)
Containment	ASME B&PVC (Section III, Subsection NB)
Criticality-related	ASME B&PVC (Section III, Subsection NG)
Safety-related	ASME B&PVC, Section III, Subsection NF or Section VIII, or AWS Structural Welding Codes, or ASME B31.1 Power Piping Code, or Military Specifications

Table 1. Summary of Criteria for Type B Transportation Packages

One of the main technical chapters of the Handbook titled **Structural Analysis** will deal with that topic as applied to packages. All packages must be analyzed under normal operating conditions to ensure that the package, cask supports, trunnions, and impact-limiter attachments will not be affected by normal operation loads and stresses. Cyclically applied loads or vibrations must also be analyzed.

The principal parts of a Type B packaging that must be analyzed from a structural perspective include:

Cask body (single or multiwall)
Closure lids and bolts
Valves and valve covers
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)

The structural analysis of Type B packages under accident conditions requires special attention to two areas: the loads developed in impacts and the strength and stability of the structure that resists those loads. As applied to spent fuel shipping casks, two types of loads are developed in impacts: (1) loads produced in the 9-m drop scenario as the package impacts a solid surface, or by deformation of impact limiters (shock absorbers) if the package is equipped with them; and (2) those produced in the 1-m pin-puncture scenario. These loads create stresses in the packaging and closure lid, and these stresses must be evaluated and shown by analysis to be less than those allowable for the material of the structure. The allowable stress limits must be adequately documented by codes or acceptable standards.

The analysis of packages using classical structural methodologies or computer codes allows the determination of the stresses, and resulting structural safety margins, in many locations of the cask and in many drop orientations. The regulations of 10 CFR 71 specify that the cask be dropped from 9-m in the most damaging orientation; and regulatory interpretation includes oblique angles that vary between corner and side. The stress locations of greatest interest in the cask are the closure lid, lid bolts, and the cask midbody, where the greatest bending moments are developed. The discontinuity stresses produced in multiwall casks, which transition from a relatively thin structural wall into a much thicker end, are of particular interest.

Package structures may also be evaluated by testing. In the case of a spent fuel cask, drop testing of a cask (either reduced- or full-scale) can confirm that the structural analyses of specific drop orientations that have been tested and are analyzed in the SARP are correct. It is also possible that the analyses of untested orientations may be shown to be valid based on extrapolation of data gathered from impact testing of models dropped in different orientations. A typical series includes 9-m drop tests of the model or prototype on its 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 Type B packages with simple geometries and large structural safety margins may rely heavily on tests in place of detailed analyses. More complex Type B packages, are normally licensed based upon the structural analyses plus confirmatory tests.

The chapter titled **Containment** will address the containment of the contents that are shipped in Type B packages because these packages are specifically limited in the quantity of radioactive materials that may be released under normal conditions of transport and under the hypothetical accident conditions. The regulatory requirement for containment of the contents of a Type B package is that no significant release of the contents of the package may occur in normal transport and only a very limited amount may be released as a consequence of the hypothetical accident conditions. A methodology for determining a package leak rate that is accepted by the regulatory authorities is described in the American National Standards Institute (ANSI) National Standard N 14.5 (ANSI 1987).

The package containment boundary consists primarily of the cavity in which the contents are placed, plus the closure lid, penetration closures, and the seals that are placed between metal surfaces to prevent leakage. Because the material and gases in the cavity cannot penetrate the metallic surface of the containment boundary to escape the package, the principal concern is the performance of the seals between the mating pieces of the cavity and the various penetrations that define the containment boundary. As a result, there are several important factors that must be considered by the designer with regard to specifying seal selection and configuration. These factors include seal performance, resiliency, permeability, expected life in use and storage, method of testing for acceptance (both in annual maintenance and routine use), and replacement during handling and maintenance of the package.

A chapter titled **Thermal Evaluation** will examine recent advances in predicting temperatures within packages under both normal and accident conditions; computer codes that are used to analyze the thermal conditions will be discussed. Also, recently a number of standard thermal problems were developed and presented to analysts in order to benchmark the many thermal codes that are currently used to predict temperatures in packages under a variety of conditions. The results of the benchmarking study will be included.

The shielding of a shipping package must be designed to maintain radiation dose rates external to package surfaces below established regulatory limits under defined normal and accident conditions. Allowable dose rates 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. The chapter titled **Shielding** will discuss the two forms of radiation that are of most concern in package design, gammas and neutrons. For example, in the case of spent fuel casks this chapter will note (1) how the radiation source can be characterized, (2) analysis methods that may result in a preliminary package design, and (3) calculational techniques that may be applied to a package design in order to predict external dose rates, particularly as applied to the shipment of SNF. Current 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; 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 (Bucholz 1987) has been developed to study simultaneously material selection (depleted uranium, lead, and steel gamma shields are considered), heat transfer, shield optimization, payload determination, and dose estimation. The CAPSIZE methodology includes a number of approximations but also allows the cask designer to make a number of preliminary trade-off and cask design decisions. Once the scoping work for preliminary cask designs has been completed, a more accurate code can be employed to carry out a more detailed analysis.

The chapter titled **Criticality** will discuss the methodologies used to ensure that spent fuel shipments will not produce a criticality incident even if the package is involved in an extremely severe transportation accident. With regard to spent fuel casks, most are designed with some type of basket in the cask cavity whose purpose it is to separate and support the assemblies, transfer heat from the assemblies to the inner cavity wall, and prevent a critical mass from forming. Subcriticality is achieved and maintained by incorporating sufficient nuclear poison in the basket separator plates to keep k_{eff} to well below 0.95, even after damage in the cavity is accounted for. Such nuclear poisons are likely to change the effective structural, as well as nuclear, properties of the basket (e.g., the ductility and thermal conductivity of borated stainless steel are affected by the amount of boron used). Consequently, the basket designed with this material must be examined for adequate structural or heat transfer properties. Its continued presence under accident conditions must be ensured. Similarly, the reflective worth of the cask body and shielding materials must be established for the single cask; and because regulations require the designer (under certain conditions) to consider an array of identical casks when evaluating the criticality safety of a package, the effectiveness of neutronically decoupling these materials from adjacent casks must be established.

The chapter on criticality also discusses the analytical methodology to be used in criticality analyses, which must be validated through the analysis of critical experiments: this validation requirement applies to both the neutron transport codes and the nuclear data. The predicted margin of subcriticality should incorporate the calculational bias as well as design and analytical uncertainties. Popular analytical methodologies involve the multigroup libraries and analytical sequences in the SCALE system (NRC 1983) based upon the KENO Monte Carlo code and the energy-pointwise Monte Carlo codes MCNP and MONK. Deterministic codes are useful for scoping analyses or for determining the reactivity worth of design uncertainties. However, the geometric capability inherent in the Monte Carlo method makes it the method of choice for analyzing the complex geometries of SNF shipping casks once the scoping studies have been completed.

Each radioactive material shipping package project is unique in terms of the fabrication challenges it presents. The project elements are dictated by the type of packaging, the packaging construction, the basis for procurement, the purchaser's and vendor's experience base, the project and purchasing rules with which the purchaser must comply, the project responsibility shared by both the purchaser and vendor, and many other considerations. Another specific chapter, titled **Package Fabrication and Acceptance Testing**, will outline some of the fundamental considerations of fabrication and acceptance testing of new packagings. It will identify those considerations that are strongly recommended in order to have a successful project, and it will also offer some experience-based suggestions that have been shown to facilitate project execution.

Another chapter **Package Operations**, will recommend examining the operation of the package during its design phase. The two most important aspects of the concept of repetitive operations with a fixed package design, aside from the actual steps required, are the dose received by operating personnel and operator performance in handling a package. To address these concerns effectively, the designer must specifically consider how the package must be handled in the user operating environment. The designer should consider how operator exposure could be reduced, above that achieved by the package shielding, by considering placement of components or process efficiencies. Package operational requirements, such as draining, and package fixtures, such as penetration valves, should be designed so as to reduce the potential for the operator to make an error in opening, closing, or otherwise using the package.

Efficient use of the package is typically supported by the use of ancillary equipment, such as a lifting fixture or special tools, specified by the package designer. To the extent possible, the package should be designed so that reliance on special tools or unique equipment is minimized. As the number of special tools or amount of unique equipment increases, the total system reliability decreases; operational complexity increases; and the transportation, handling, packaging, and decontamination requirements increase. Reducing (or eliminating) the reliance on unique equipment, tools, fixtures, or fasteners reduces the burden on all package users. Nevertheless, it is possible for the designer to identify special equipment that decreases operator dose and/or the potential for operator error. These items merit consideration for inclusion as part of the ancillary equipment that is provided to the package user.

Maintenance of a packaging will be discussed in a separate chapter. The requirement for maintenance inspections and tests arises from (1) a need to ensure the preservation of public health and safety in the continued use of the package, (2) the use of accepted standards in the design and fabrication of packages, and (3) a historical view that periodic inspection of any in-service equipment is good business practice. As the package designer develops maintenance inspection, test, or replacement requirements during design or fabrication, care must be exercised to see that the requirements are incorporated into the maintenance program.

Maintenance may be either scheduled or unscheduled. Scheduled maintenance is typically performed on a per-use basis, as well as on a periodic basis, usually annually. Other time or use periods may be selected by the package designer depending on the package configuration or other specific requirements. Unscheduled maintenance arises from defects found in the package after it was placed in service. In either case, maintenance performed to restore the package to (or demonstrate the package is in) the "as certified" condition must be carefully documented. The resulting documentation then becomes a part of the permanent record of the package.

Quality Assurance (QA) is a cornerstone of all packaging developmental programs and is discussed in a separate chapter. The information that will be provided in this chapter is intended to assist the package designer and the user in developing a comprehensive QA plan in order to demonstrate compliance with DOE Orders and federal regulatory QA requirements. The information developed must demonstrate that the necessary QA requirements are applied to design, procurement, fabrication, acceptance testing, operation, maintenance, modification, and repair of the packaging in order to ensure adequate control over all items and activities that are important to safety. The level of the QA effort as applied to the safety and performance function of each cask item (part or component) is determined by using a graded approach based on a three-step process that is discussed in detail. In any case, the cask designer must ensure that sufficient information is provided in the QA chapter of the SARP to develop an effective QA plan.

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