NUREG–1536

Standard Review Plan
for Dry Cask Storage Systems

Draft Report for Comment

U.S. Nuclear Regulatory Commission

Office of Nuclear Material Safety and Safeguards

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Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
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ABSTRACT

The Standard Review Plan (SRP) For Dry Cask Storage Systems provides guidance to the Nuclear Regulatory Commission staff in the Spent Fuel Project Office for performing safety reviews of dry cask storage systems. The SRP is intended to ensure the quality and uniformity of the staff reviews and present a basis for the review scope and requirements.

Part 72, Subpart B generally specifies the information needed in a license application for the independent storage of spent nuclear fuel and high level radioactive waste. Regulatory Guide 3.61 "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask" contains an outline of the specific information required by the staff. The SRP is divided into 14 sections which reflect the standard application format. Regulatory requirements, staff positions, industry codes and standards, acceptance criteria, and other information are discussed.

Comments on this draft, will be considered and incorporated into the SRP as appropriate. The SRP is scheduled for publication as an NRC NUREG document late in 1996. Comments, errors or omissions, and suggestions for improvement should be sent to the Director, Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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(to be added)

INTRODUCTION

"The Standard Review Plan for Dry Cask Storage Systems" (SRP for DCSS) is prepared for the guidance of staff reviewers in the Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, in performing safety reviews of applications for approval of spent fuel storage cask systems. The principal purpose of the DCSS SRP is to ensure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. Other purposes of the SRP are to make information about regulatory matters widely available, to improve communication, and to help interested members of the public and the nuclear power industry in better understanding of the staff review process.

The use of dry cask storage systems for the storage of spent nuclear fuel, may be conducted under either a site-specific or general license for operating an independent spent fuel storage installation (ISFSI). At present, any holder of an active reactor operating license under Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50), has the authority to construct and operate an ISFSI under the provisions of the general license. Requirements and restrictions for construction and pre-operational activities of such an ISFSI are discussed in 10 CFR Part 72, Subparts K and L. The requirements for pursuing a site specific ISFSI license is discussed in Subparts B and C of 10 CFR Part 72. In either case, the cask design that will be used in an ISFSI must be reviewed and approved for use by the NRC staff prior to actual loading of spent fuel. This Standard Review Plan describes the methods used by the NRC staff to conduct such a review.

The safety review of dry cask storage systems is primarily based on the information provided by an applicant, or cask vendor, in a Safety Analysis Report (SAR). Sections 72.24 and 72.230 of 10 CFR Part 72, requires that each application for a license to store spent nuclear fuel or for approval of spent fuel casks include an SAR. The SAR must be sufficiently detailed to permit the staff to determine whether the design can be fabricated and used without endangering the health and safety of the public. Before submitting an SAR, an applicant should have designed and analyzed the storage cask system in sufficient detail to conclude that it can be fabricated and operated safely. The SAR is the principal document in which the applicant provides the information needed to understand the basis upon which this conclusion has been reached.

Section 72.24 specifies, in general terms, the information to be supplied in a SAR. The specific information required by the staff for an evaluation of an application is identified in Regulatory Guide 3.61, "Standard Format and Content of Topical Safety Analysis Reports for a Spent Fuel Dry Storage Facility." The SRP sections are keyed to the standard format defined in Regulatory Guide 3.61. Similar information is also provided in Regulatory Guide 3.62, "Standard Format and Content for the Safety Analyses Report for On-site Storage of Spent Fuel Storage Casks." Review plans have not been prepared for SAR sections that consist of background or design data that are included for information or for use in the review of other SAR sections.

The SRP is written so as to cover a variety of site conditions and cask system designs. Each section is written to provide the complete review procedure and all current acceptance criteria for all of the areas of review pertinent to that section. However, for any given application, the staff reviewers may select and emphasize particular aspects of each SRP section as are appropriate for the application. In some cases a cask feature may be sufficiently similar to that of a previous cask so that a de novo review of the feature is not needed. For these and other similar reasons, the staff may not carry out in detail all of the review steps listed in each SRP section in the review of every application. Conversely, the staff may find the need to ask additional questions or probe areas in greater depth, to adequately review a particular design.

The individual SRP sections address, in detail, the matters that are reviewed, the basis for the review, how the review is accomplished, and the conclusions that are sought.

Each SRP section is organized into seven subsections, as follows:

I. Review Objective

This subsection contains a statement of the purpose and scope of the review.

II. Areas of Review

This subsection contains a description of the systems, components, analyses, data, or other information that is reviewed as part of the particular SAR section in question. It also contains a discussion of the information needed or coordination expected from reviewers of other SAR sections in order to complete the subject technical review.

III. Regulatory Requirements

This subsection summarizes, in general, the applicable sections of 10 CFR Part 72 pertaining to a specific SAR chapter. This list is not all inclusive (e.g., some parts of the regulations such as Part 20 are assumed to always apply).

IV. Acceptance Criteria

This subsection addresses the design criteria and analytical methods that the U.S. Nuclear Regulatory Commission (NRC) staff have found to be acceptable for meeting the regulatory requirements, called out in 10 CFR Part 72, that apply to the subsection being reviewed.

These subsections typically set forth the solutions and approaches that have previously been determined to be acceptable by the staff in dealing with a specific safety problem or design area that is important to safety. These solutions and approaches are discussed in the SRP so that staff reviewers can take uniform and well-understood positions as similar safety issues arise in future cases. Like Regulatory Guides, these resolutions and approaches are acceptable to the staff, but they are not required as the only possible solutions and approaches. However, applicants should recognize that, as in the case of Regulatory Guides, substantial staff time and effort has gone into the development of these acceptance criteria, and that a corresponding amount of time and effort may be required to review and accept new or different solutions and approaches. Thus, applicants proposing solutions and approaches to new safety issues or analytical techniques other than those described in the SRP should expect longer review times and more extensive questioning in these areas. An alternative is to propose new methods on a generic basis, apart from a specific license application. This approach could avoid the impact of the additional review time on individual cases; however, Agency priorities may impact the timeliness of the review of such issues.

V. Review Procedures

This subsection discusses how the review is accomplished. The section generally discuss the procedure that the reviewer goes through to provide reasonable verification that the applicable safety criteria have been met.

VI. Evaluation Findings

This subsection presents the type of conclusion that is sought for the particular review area. For each section, a conclusion of this type is included in the staff's Safety Evaluation Report in which the staff publishes the results of their review. The SER also contains a description of the review including such subjects as which aspects of the review were selected or emphasized; which matters were modified by the applicant, require additional information, will be resolved in the future, or remain unresolved; where the cask's design deviates from the criteria stated in the SRP; and the bases for any deviations from the SRP or exemptions from the regulations.

VII. References

This subsection lists the references commonly used in the review process.
The SRP and the Standard Format are directed toward storage cask systems designed for spent fuel with zircalloy cladding. Staff reviews will adapt the SRP for use in the reviews of other storage designs and spent fuel types as needed.

The SRP results from many years staff experience establishing and using regulatory requirements in evaluating the safety of spent fuel storage system designs and in reviewing SARs. A great deal of progress has been made in the methods of review and in the development of regulations, guides, and standards. This SRP may be considered a part of a continuing regulatory standards development activity that not only documents current methods of review but also provides the base of orderly modifications of the review process in the future.

The SRP will be revised and updated periodically as the need arises to clarify the content or correct errors and to incorporate modifications approved by the Director of the Office of Nuclear Material Safety and Safeguards.

Comments and suggestions for improvement will be considered and should be sent to the Director of Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear

Regulatory Commission, Washington, DC 20555. Notices of errors or omissions should also be sent to the same address.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN FOR STORAGE CASKS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

1.0 GENERAL DESCRIPTION

I. REVIEW OBJECTIVE

The purpose of the review of the cask description is to ensure that the applicant has provided a non-proprietary description of the dry cask storage system (DCSS) that is suitable for familiarizing the reviewers and other interested parties with its pertinent features.

The general description of the cask or cask system enables all reviewers, regardless of their specific review assignments, to obtain a basic understanding of the cask system, its components and the protections afforded for the health and safety of the public. Because much of the information relevant to this section is presented in more detail in follow-on sections, this initial review focuses on familiarization with the cask and consistency with the remaining sections of the Safety Analysis Report (SAR). Reg. Guide 3.61[1] provides general guidance on information that should be included in the cask general description.

II. AREAS OF REVIEW

1. "Cask Description and Operational Features"

2. "Drawings"

3. "Cask Contents"

4. "Qualifications of Applicant"

5. "Consideration of 10 CFR Part 71 (Transportation) Requirements"

III. REGULATORY REQUIREMENTS

1. General Description and Operational Features

A general description and discussion of the cask must be presented, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. [10 CFR Part 72.24(b)]

2. Drawings

Structures, systems, and components important to safety must be described in sufficient detail to enable an evaluation of their effectiveness. [10 CFR Part 72.24(c)(3)]

3. Cask Contents

Specifications must be provided for the spent fuel to be stored in the cask, such as, but not limited to, type of spent fuel (i.e., Boiling Water Reactor (BWR), Pressurized Water Reactor (PWR), both), maximum allowable enrichment of the fuel prior to any irradiation, burn-up (i.e., megawatt-days/Metric Ton Uranium), minimum acceptable cooling time of the spent fuel prior to storage in the cask (aged at least one year), maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), and the inert atmosphere requirements. [10 CFR Part 72.2(a)(1) and 10 CFR Part 72.236(a)]

4. Qualifications of Applicant

The technical qualifications, including training and experience, of the applicant to engage in the proposed activities, must be included. [10 CFR Part 72.24(j), 10 CFR Part 72.28(a)]

5. Consideration of 10 CFR Part 71 (Transportation) Requirements

If the cask system under consideration has previously been reviewed and certified for use as a transportation cask, a copy of the Certificate of Compliance issued for the cask under 10 CFR Part 71, including drawings and other documents referenced in the certificate, must be included with the application. [10 CFR 72.230(b)]

IV. ACCEPTANCE CRITERIA

1. Cask Description and Operational Features

A broad overview and general non-proprietary description of the cask system (including illustrations) are provided which clearly identify the functions of all cask system components and provides a list of those components classified as "important to safety".

2. Drawings

A non-proprietary drawing of the cask system is provided.

3. Cask Contents

The applicant has characterized the fuel and other radioactive wastes expected to be stored in the cask storage system. If the potential for storing degraded fuel exists, the SAR should contain a discussion of how the criticality and retrievability requirements will be maintained.

4. Qualifications of Applicant

The roles and interactions between the cask designer, or vendor, and other agents such as potential licensees, fabricators and contractors are clearly defined and established. Responsibilities of the applicant and staff are delineated and activities that will not be performed by the applicant are succinctly defined. A discussion of the applicability of an NRC approved Quality Assurance program should be included.

5. Consideration of 10 CFR Part 71 (Transportation) Requirements

If the cask system under review has previously been reviewed for use as a transportation cask, the Part 71 Certificate of Compliance and associated documents are included in the submittal.

V. REVIEW PROCEDURES

1. Cask Description and Operational Features

This section should provide a broad overview of the cask system design that is non-proprietary and which may be used as a tool to familiarize interested parties with the features of the cask system. The principal characteristics of the cask, including its dimensions, weight, and materials of construction, must be presented. Components that are considered as important to safety should be clearly noted. Functional features such as confinement, sampling ports, valves, lids, seals, closure mechanisms, shielding, and lifting devices must be identified and described. A clear definition of the primary confinement system is of particular importance. The heat-removal system, shielding, criticality poisons, instrumentation, and any other special design features of the cask must also be discussed. Sketches and diagrams, if presented in this section, should be compared with the detailed drawings presented elsewhere in the SAR.

In addition to information on a single cask, a description of any limitations on the arrangement of cask arrays must be presented. For some cask types this may consist of a minimum spacing between casks. However, for cask systems such as those with metal confinement vessels stored in a concrete vault, information on the configuration of vault compartments and horizontal/vertical arrangement is necessary.

2. Drawings

Drawings are usually presented in Section 1 of the SAR. All structures, systems, and components important to safety should be sufficiently detailed to enable an evaluation of their effectiveness. In addition, information on non-safety items may also be necessary to ensure that they do not impede the safety systems. Reviewers should note that any drawings that are relied on as the technical basis for adding the cask design to the "List of approved spent fuel storage casks."

contained in Subpart K of 10 CFR 72, become part of the public record. Such drawings will not be treated as proprietary and will be made available to the public (10 CFR 2.790(c)).

The level of detail needed on the drawings is generally assessed by each reviewer during the evaluation of follow-on sections of the SAR. Particular attention should be devoted to ensuring that dimensions, materials, and other information on drawings are consistent with those described in the text of the SAR. If reduction in size has rendered information unclear or illegible, larger or full-size drawings should be requested from the applicant.

Drawings applicable to the SAR review should be identified by number and revision in Section 12 of the SER.

3. Cask Contents

A top-level description of the fuel to be stored in the cask should be presented. Because a very detailed description of the fuel is typically specified in Section 2 ("Principal Design Criteria"), the information presented in the "General Description" is important only to the extent that it provides an overall familiarization with the cask. Key parameters include type of fuel (e.g., PWR, BWR), number of fuel assemblies, and condition of the fuel assemblies (intact or consolidated). Other characteristics such as burnup, enrichment, heat load, cooling time, assembly vendor/configuration (e.g., Westinghouse 17x17) are often also included in this section and repeated in the principal design criteria. In addition to the fuel itself, the cover gas should also be identified.

If the cask system is intended to be used to store fuel or non-fuel core components that will not have intact cladding or an integral confinement boundary when placed in the confinement cask, the possible conditions of those fuel or components must be stated. 10 CFR 72.122(h)(1) requires "canning" or use of acceptable other means for storage of fuel with cladding that is not or may not remain intact and for unconsolidated assemblies (without intact cladding). The basic requirements are: maintaining subcriticality, preventing unacceptable release of contained radioactive material, avoiding excessive radiation dose rates and doses, and maintaining the ready retrievability for the contained items.

If the requested approval is to cover the possible use of the cask system for storage of non-fuel core components, summary descriptions of those components should be presented. Also, if the components are degraded, e.g., the component does not provide a primary confinement boundary to contain radioactive gas or other dispersable radioactive materials, those possible conditions and alternative methods for confinement, if any, should be described.

4. Qualifications of Applicant

A clear designation of the applicant and the prime agents, consultants, and contractors, for design, fabrication, and testing of DCSS components should be identified. A clear division and assignment of responsibilities should be delineated. Although specific subcontractors may not be known at the time the SAR is submitted, activities that will not be performed by the applicant should be clearly identified. The technical qualifications, especially previous experience, of these organizations to engage in the proposed activities should also be included. The applicant should discuss what activities will be conducted under an NRC approved Quality Assurance program and how those activities of its consultants, contractors, and fabricators will be monitored.

5. Consideration of 10 CFR Part 71 (Transportation) Requirements

If the cask system under review for storage has previously been reviewed for use as a transportation cask, a copy of the Certificate of Compliance issued for the cask under 10 CFR Part 71,[2] if applicable, and drawings and other documents referenced in the certificate, must be included with the application or incorporated by reference to the NRC Part 72 docket number. Because applications for certification under 10 CFR Parts 71 and 72 are sometimes submitted concurrently, the final (approved) version of such documents may not be available at the time of initial DCSS SAR submission.

Applicable documentation of the Part 71 certification, including questions and responses from the Part 71 review, is generally provided to the Part 72 review team, as appropriate. Substantial coordination of the Part 71 and Part 72 reviews is necessary to ensure consistency and avoid duplication of efforts. The applicant is responsible for promptly informing the review team of any cask system design changes precipitated by any concurrent safety reviews. Provisions for communicating these changes should be addressed by and discussed with the applicant.

VI. EVALUATION FINDINGS

Review the Part 72 acceptance criteria and provide a summary statement for each similar to the following:

A general description and discussion of the cask is presented in Chapter(s) _____ of the SAR, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

Drawings for structures, systems, and components important to safety are presented in Chapter _____ of the SAR. A listing of those drawings that were relied upon as a basis for approval is provided in Section 12 of the Safety Evaluation Report (SER).

Specifications are provided in chapter ______ for the spent fuel to be stored in the cask. Additional details on these specifications are presented in Chapter 2 of the SAR and SER.

The technical qualifications of the applicant to engage in the proposed activities are included in Chapter _____ of the SAR.

The [cask designation] has been [is being/is not] certified for transportation under 10 CFR Part 71.

A copy of the Part 71 SAR and Certificate of Compliance issued under Part 71 are on file with NRC under Docket No. ________ [if applicable].

The staff concludes that the information presented in this section of the SAR satisfies the requirements of 10 CFR Part 72. This finding is based on a review which considered the regulation itself, Regulatory Guide 3.61, and accepted practices.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.61, "Standard Format and Content for a Safety Analysis Report for a Spent Fuel Storage Cask", February 1989.

2. U.S. Code of Federal Regulations, Title 10, Part 71, "Packaging and Transportation of Radioactive Material."

2.0 PRINCIPAL DESIGN CRITERIA

I. REVIEW OBJECTIVE

This section evaluates the principal design criteria for structures, systems, and components important to safety to ensure that they comply with the general design criteria of 10 CFR Part 72. The SAR may present these criteria in Section 2 or in any of its other sections. These design criteria include specifications of the fuel to be stored in the cask, the external conditions during normal and off-normal operations, accident conditions, and natural phenomena events. A detailed evaluation of the design criteria and the assessment of the compliance of the cask system design with these criteria are presented in Sections 3 to 14 of the SER.

II. AREAS OF REVIEW

The areas of review identified below are those that have been adopted during recent reviews of dry cask system designs by the NRC staff, and include those areas noted in Regulatory Guide 3.61,[1]

1. Structures, Systems, and Components Important to Safety

2. Design Bases for Structures, Systems, and Components Important to Safety

   a. "Spent Fuel Specifications"
   b. "External Conditions"

3. Design Criteria for Safety Protection Systems[a]

   a. "General"
   b. "Structural"
   c. "Thermal"
   d. "Shielding/Confinement/Radiation Protection"
   e. "Criticality
   f. "Operating Procedures"
   g. "Acceptance Tests and Maintenance"
   h. "Decommissioning"

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[a] Also may be stated as "Design Criteria for Structures, Systems, and Components Important to Safety".

III. REGULATORY REQUIREMENTS

1. Structures, Systems, and Components Important to Safety

Structures, systems, and components important to safety must be identified. [10 CFR 72.24(c)(3), 10 CFR Part 72.44(d)]

Design bases and design criteria must be provided for structures, systems, and components important to safety. [10 CFR Part 72.236(b); 10 CFR Part 72.24(c)(1); 10 CFR Part 72.24(c)(2); 10 CFR Part 72.120(a)]

2. Design Bases for Structures, Systems, and Components Important to Safety

a. Spent Fuel Specifications

Specifications must be provided for the spent fuel to be stored in the cask, such as, but not limited to, type of spent fuel (i.e., BWR, PWR, both), maximum allowable enrichment of the fuel before any irradiation; burn-up (i.e.,megawatt-days/mtu); minimum acceptable cooling time of the spent fuel prior to storage in the cask (aged at least one year); maximum heat designed to be dissipated, maximum spent fuel load; condition of the spent fuel (i.e., intact assembly or consolidated fuel rods); and the inerting atmosphere requirements. [10 CFR Part 72.2(a)(1) and 10 CFR Part 72.236(a)]

b. External Conditions

The design bases for structures, systems, and components important to safety must reflect appropriate consideration of environmental conditions associated with normal operations and appropriate combinations of normal and accident conditions and the effects of natural phenomena events. [10 CFR Part 72.122(b)]

3. Design Criteria for Safety Protection Systems

a. General

The cask must be designed to store the spent fuel safely for a minimum of 20 years and permit maintenance as required. [10 CFR Part 72.236(g)]

Structures, systems, and components important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR Part 72.122(a)]

Applicable codes and standards for structures, systems, and components important to safety must be identified. [10 CFR Part 72.24(c)(4)]

b. Structural

Structures, systems and components important to safety must be designed to accommodate the combined loads of normal, accident, and natural phenomena events with an adequate margin of safety. [10 CFR Part 72.122(b), 10 CFR Part 72.122(c), 10 CFR Part 72.24(c)(3)]

The design earthquake must be equivalent to or exceed the safe shutdown earthquake of a nuclear plant at sites evaluated under 10 CFR Part 100. [10 CFR Part 72.102(f)]

The cask must reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR Part 72.236(l)]

The cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions. [10 CFR Part 72.236(c); 10 CFR Part 72.124(a)]

The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. [10 CFR Part 72.122(h)(1)]

Storage systems must be designed to allow ready retrieval of spent fuel waste for further processing or disposal. [10 CFR Part 72.122(l)]

c. Thermal

Spent fuel storage or handling systems must be designed with a heat-removal capability having testability and reliability consistent with its importance to safety. [10 CFR Part 72.128(a)(4)]

The cask must be designed to provide adequate heat removal capacity without active cooling systems. [10 CFR Part 72.236(f)]

The spent fuel cladding must be protected against degradation that leads to gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. [10 CFR Part 72.122(h)(1)]

Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. [10 CFR Part 72.122(l)]

d. Shielding/Confinement/Radiation Protection

Radiation shielding and confinement features must be provided that are sufficient to meet the requirements of 10 CFR Part 72.104 and 72.106. [10 CFR Part 72.236(d); 10 CFR Part

72.128(a)(2); 10 CFR Part 72.128(a)(3); 10 CFR Part 72.126(a)]

During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to (1) planned discharges of radioactive materials, radon and its decay products excepted, to the general environment, (2) direct radiation from ISFSI or MRS operations, and (3) any other radiation from uranium fuel cycle operations within the region. [10 CFR Part 72.104(a); 10 CFR Part 72.236(d); 10 CFR Part 72.24(d)]

Any individual located on or beyond the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident. The minimum distance from the spent fuel handling and storage facilities to the nearest boundary of the controlled area shall be 100 meters. [10 CFR Part 72.106(b); 10 CFR Part 72.236(d); 10 CFR Part 72.24(m); 10 CFR Part 72.24(d)]

The cask must be designed to provide redundant sealing of confinement systems. [10 CFR Part 72.236(e)]

Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [10 CFR Part 72.122(h)(4); 10 CFR Part 72.128(a)(1)]

Instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal and off-normal operation. Those control systems that must remain operational under accident conditions must be identified. [10 CFR Part 72.122(i)]

The spent fuel cladding must be protected against degradation that leads to gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. [10 CFR Part 72.122(h)(1)]

e. Criticality

Spent fuel transfer and storage systems must be designed to be subcritical under all credible conditions. [10 CFR Part 72.124(a); 10 CFR Part 72.236(c)]

When practicable, the design must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design shall provide for positive means to verify their continued efficacy. [10 CFR Part 72.124(b)]

f. Operating Procedures

The cask must be compatible with wet or dry spent fuel loading and unloading procedures. [10 CFR Part 72.236(h)]

Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. [10 CFR Part 72.122(l)]

The cask must be designed to minimize the quantity of radioactive waste generated. [10 CFR Part 72.128(a)(5); 10 CFR Part 72.24(f)]

The equipment and processes used to maintain control of radioactive effluents must be described. [10 CFR Part 72.24(l)(2)]

The cask must be designed to facilitate decontamination to the extent practicable. [10 CFR Part 72.236(i)]

Operational restrictions must be established to meet Part 20 limits and as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI operations. [10 CFR Part 72.104(b); 10 CFR Part 72.24(e)]

g. <u>Acceptance Tests and Maintenance</u>

The cask design must permit testing and maintenance as required. [10 CFR Part 72.236(g)]

Structures, systems, and components important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR Part 72.122(a); 10 CFR Part 72.122(f); 10 CFR Part 72.128(a)(1); 10 CFR Part 72.24(c)]

h. <u>Decommissioning</u>

The cask must be compatible with wet or dry unloading facilities. [10 CFR Part 72.236(h)]

The cask must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment and minimize the quantity of radioactive wastes, contaminated equipment, and contaminated materials at the time the ISFSI is permanently decommissioned. [10 CFR Part 72.130; 10 CFR Part 72.24(f); 10 CFR Part 72.236(i)]

Information on proposed practices and procedures for the decontamination of the site and facilities and for disposal of residual radioactive materials after all spent fuel has been removed must be described in order to provide reasonable assurance that decontamination and decommissioning will provide adequate protection to the health and safety of the public. [10 CFR Part 72.30(a); 10 CFR Part 72.24(q)]

IV. ACCEPTANCE CRITERIA

1. Structures, Systems, and Components Important to Safety

The general configuration of the dry cask storage system should be discussed, including an overview of specific components and their intended functions. Those components deemed as "important to safety" should be defined and their safety function addressed in terms of how they meet the general design criteria and regulatory requirements discussed above. Additional information on specific functional requirements for individual cask component functions are addressed in the follow-on sections of this SRP.

2. Bases for Structures, Systems, and Components Important to Safety

Detailed descriptions of each of the items listed below are generally found in specific chapters of the SAR; however, a brief description of these areas, including an overview of the analytical techniques used in the design process, is also captured in Chapter 2 of the SAR. This overview provides the reviewers with an overall perspective of how specific cask components interact with each other in order to meet the regulatory requirements of 10 CFR Part 72. This discussion should be considered non-proprietary and may be used to familiarize interested persons with the design features and bounding conditions of operation for a given cask system.

a. Spent Fuel Specifications

The applicant should define the range and types of spent fuel or other radioactive materials that the cask system is designed to store. Specifications regarding fuel type, fuel age, burnup history, configuration requirements, maximum allowable heat generation rates and the maximum amount of fuel permitted for storage in the cask system should be clearly defined and stated. For cask systems that will be used to store radioactive materials other than spent fuel, the type and amount of radionuclides, and source strength and radiation energy spectra should be specified.

b. External Conditions

The SAR should define the bounding conditions in which the cask system is expected to operate. Such conditions include both normal and abnormal environmental conditions as well as bounding accident conditions. These bounding conditions may or may not be actual credible events. The effects of natural events such as tornadoes, earthquakes, floods, and lightning strikes should be considered. The effects of such events are addressed in individual sections of the SRP (e.g., the effects of an earthquake on the cask structural components are addressed in Section 3, "Structural Analysis").

3. Design Criteria for Safety Protection Systems

a. General

The SAR should define an expected design lifetime for the cask system. Although the approval period is 20 years, the expected design lifetime should include some provisions for extending the approval. Any rationale for cask storage system reapproval should be mentioned here. The Quality Assurance program and industry codes and standards that will be applied to the design, fabrication, construction, and operation of the dry cask storage system should be briefly described.

b. Structural

The SAR should define how the cask system structural components are designed to accommodate combined normal, off-normal, and accident loads, while protecting the cask contents from significant structural degradation, criticality, and loss of confinement, and at the same time preserving retrievability. This discussion is generally an overview of the analytical techniques and calculational results from the detailed analysis discussed in Chapter 3 of the SAR and may be used in a non-proprietary forum for the purposes of discussion.

c. Thermal

The applicant should provide a general discussion of the heat removal mechanisms, including the reliability and testability of the mechanism and any associated limitations. All heat removal mechanisms must be passive and independent of human actions under normal and abnormal conditions.

d. Shielding/Confinement/Radiation Protection

The applicant should describe those protection features of the cask system that protect occupational workers and members of the public against direct radiation dose, release of radioactive material, and which minimize the dose after any abnormal or accident conditions.

e. Criticality

The SAR should address the mechanisms and design features that enable the dry cask storage system to maintain spent fuel in a subcritical condition under normal, abnormal and accident conditions.

f. Operating Procedures

The applicant should provide general guidance, regarding the content of normal, abnormal, and accident response procedures, to potential licensees. Cautions regarding loading and unloading procedures should be mentioned here. Applicants may choose to provide mock procedures to be used as an aid for preparing detailed site-specific procedures.

g. Acceptance Tests and Maintenance

This section should identify the general commitments and industry codes and standards used to derive acceptance, maintenance, and periodic surveillance tests used to verify the capability of cask components to perform their designated functions. The methodology used to determine the need for such tests on specific components should also be discussed.

h. Decommissioning

Casks must be designed for ease of decontamination and eventual decommissioning. The applicant should describe the features of the design that support these two activities.

V. REVIEW PROCEDURES

Chapter 2 of the SAR should be reviewed by all members of the review team. Although Regulatory Guide 3.61 addresses the standard format and content for an SAR, it does not discuss the difference in level of detail for component design criteria discussed in Chapter 2 of the SAR and that addressed in the later sections of the SAR. Consequently, the reviewer for each chapter of the SAR must consider Chapter 2 in combination with additional details presented later in the SAR. In this SRP, design criteria applicable to each of the follow-on sections are generally discussed in detail in those sections, without regard to criteria that may have been presented in Chapter 2.

Although the design criteria presented in the SAR may be acceptable to the staff, the actual design may not meet either these criteria or the applicable regulatory requirements. It is also possible that the design criteria themselves, as presented in the SAR, may be unacceptable for application to a cask system design. As a result, the design may either be unacceptable in that it does not meet the regulatory requirements, or the design may satisfy alternate criteria that are not described in the SAR, that are, however, acceptable to the NRC staff. Reviewers should bring any of these inconsistencies to the immediate attention of both NRC management and the applicant.

1. Structures, Systems, and Components Important to Safety

Verify that structures, systems, and components important to safety (as defined by 10 CFR Part 72.3) have been clearly identified and that the rationale for this designation is explained. Such information may be included in tabular form. Review the general description of the cask presented in Section 1 of the SAR. Justification should be provided for excluding structures, systems, and components from this designation.

Pay particular attention to instrumentation and other equipment (e.g., lifting devices and transport vehicles). In general, the NRC staff accepts that monitoring systems need not be classified as important to safety. For example, a failure in the functioning of the pressure monitoring system does not itself result in a release of radionuclides. Additional justification for not considering such systems as important to safety may be presented in later sections of the SAR and summarized in Section 2.

Structures, systems, and components designated as important to safety should be included or referenced in Design Features of the Technical Specifications in Section 12 of the SAR.

2. Design Bases for Structures, Systems, and Components Important to Safety

The design bases for the cask system approval should identify the range of spent fuel configurations and characteristics, the enveloping conditions of use, and site characteristics. These determine the bounds for using the SAR as a reference, by the ISFSI owner, in place of providing additional proofs of suitability in the covered topics. Design bases that are less than the characteristics for actual use and site must be addressed by the ISFSI owner in the SAR for a license application.

a. Spent Fuel Specifications

Review the detailed specifications presented in Chapter 2 of the SAR for the spent fuel to be stored in the cask and ensure that they are consistent with those discussed in Chapter 1. The description of the spent fuel should include the type (PWR/BWR); configuration (e.g., 17x17, 15x15, 8x8; vendor; number of assemblies per cask; enrichment; specific power of assumed irradiation; burnup; cooling time; decay heat generation rate; type of cladding, dimensions, total weight per assembly, and uranium weight per assembly. In addition, if control assemblies are to be stored with the fuel, ensure that weight, dimensions, heat load, and other appropriate information (e.g., number per cask) are specified.

Examine any limitations on the condition of the spent fuel. If damage more severe than pinhole leaks or hairline cracks is allowed, the effects of such damage must be assessed in later chapters of the SAR. If damaged rods have been removed from a fuel assembly, determine if a requirement exists to replace them with dummy rods before loading into the cask. The presence of an additional moderator will need to be addressed in the criticality analysis in Section 6.

The release of fill and fission product gases from failed fuel rods increases the pressure in the cask cavity and the radiation released to the atmosphere in the event of confinement failure. Sufficient information should be provided regarding the fill/fission product gas volume to enable a calculation of the pressure in the cask cavity resulting from cladding failure during storage. NRC has accepted the following as assumptions for the minimum percentages of fuel rods that have failed (and released their gases) when calculating internal cask pressures: normal conditions, 1%; off-normal conditions, 10%; and, design basis (accident and extreme natural phenomena) conditions, 100%. The NRC accepts that a maximum of 30% of the significant radioactive gases within any failed fuel rod is available for release (see Section 7, paragraph III.3).

Pay particular attention to the specification of burnup, cooling time, and decay heat. These parameters are generally not independent and the manner in which they are specified can significantly affect the maximum allowed cladding temperature, as discussed in Section 4 of the SRP.

Note the specification of enrichment limits. As discussed in Section 5 of the SRP, the criticality evaluation is based on the highest enrichment (for a given fuel assembly), while the shielding source term, especially for neutrons, should be based on the lowest enrichment (for a given burnup).

The SAR will typically list various fuel assemblies that can be stored in the cask. In general, no one type of fuel assembly will be bounding for all analyses. Ensure that the applicant has justified which specifications are bounding for each of the evaluations to be presented in follow-on sections of the SAR. Specifications used in these analyses should also be clearly identified or referenced in Chapters 12 of the SAR and SER.

If the certificate is to cover storage of non-fuel core components in the cask system, review the detailed specifications, conditions, and constraints for these presented in the SAR. These should be as detailed as the applicable information presented for the fuel designs to be stored.

b. External Conditions

The SAR should identify those external conditions that significantly affect, or could potentially affect, the performance of the cask. These design-basis conditions will generally restrict either the sites at which the cask can be used for spent fuel storage or the manner in which the cask can be handled. For example, by selecting the design basis earthquake (DBE), the SAR limits the use of the cask system being reviewed to sites for which the safe shutdown earthquake does not exceed the cask system DBE. By establishing a design-basis drop, the SAR defines the maximum height to which a cask can be lifted without additional safety analysis by the licensee. The reviewer should note that movement of cask system components within a reactor building may not meet the NRC's criteria for movement of heavy loads within the reactor building. As such, coordination with the Office of Nuclear Reactor Regulation project manager should occur during the early stages of cask design review.

The NRC staff has generally addressed the conditions discussed below. Depending on specific details of cask design, other conditions may be relevant. The reviewer should pay particular attention to special design features and how these might be affected by other external conditions.

"Normal-conditions", including conditions of handling and transfer and the extreme ranges of normal-conditions, are to be assumed to exist during design basis accidents or design basis natural phenomena, with the exception of irrational or readily avoidable combinations. For example, an earthquake or tornado may occur at any time and with any "normal" conditions, whereas it can be presumed that transfer, loading, and unloading operations would not be conducted during a flood.

"Off-normal" conditions and events are assumed to occur concurrently with normal conditions that are not mutually exclusive. It is not required that the SAR analyze or that the system be designed for the simultaneous occurrence of independent off-normal-conditions or events, design basis accidents, or design basis natural phenomena. Conditions involving equipment or instrument failure or malfunction that could be latent (occur and remain undetected (e.g., a misreading instrument unless procedures provide for checking, or ventilation blockage if not checked, undetected damage from a prior design basis event or condition if there are no provisions for detection, recovery, or remediation) should be presumed to exist concurrently with other off-normal or design basis conditions and events.

If appropriate, these design bases should be included as operating controls and limits in Chapter 12 of the SAR and SER.

(1) Normal-conditions

The primary external conditions that affect cask performance are the ambient temperatures, solar insolance, and the operational environment experienced by the cask.

The NRC accepts as the maximum and minimum "normal" temperatures the highest and lowest temperatures recorded in each year averaged over the years of record. The SAR may select any design-basis temperatures as long as the restrictions they impose are acceptable to the applicant. If the cask is also designed for transportation, the temperature requirements of 10 CFR Part 71[2] could determine the design basis for storage.

The NRC staff accepts a treatment of solar insolance for storage casks similar to that prescribed in 10 CFR Part 71.71. The applicant may select another design approach, which must be justified in the SAR.

The operational environment experienced by the cask under normal-conditions includes the manner in which the cask is loaded, unloaded, and lifted. For example, the presence or absence of water from the spent fuel pool will affect dose rates received during closing of the cask, and drying operations will impact the temperatures of the fuel cladding. The manner in which the cask is lifted will determine the load on the trunnions and/or lifting yoke. The orientation of the cask (vertical/horizontal) and its height above ground during transport to the ISFSI will establish initial conditions for the drop accidents discussed below.

(2) Off-Normal Conditions

Several off-normal-conditions are generally addressed by the SAR. These should include: variations in temperatures beyond normal; failure of 10% of the fuel rods combined with off-normal temperatures; failure of one of the confinement boundaries; partial blockage of air vents; human error (simple negligence); out-of-tolerance equipment performance; equipment failure; and instrumentation failure or faulty calibration.

(3) Accidents

The staff has generally considered that the following accidents should be evaluated in the SAR, regardless of whether they are considered credible. These do not constitute the only accidents that should be addressed if the SAR is to serve as a reference for accidents for the site-specific application. Other accidents should be as derived from a hazard analysis. These may include accidents, for example, resulting from gross negligence, failure of instruments, lightning, and other occurrences. Accident situations that are not credible because of design features or proposed operating procedures should be identified as such in the SAR.

(a) Cask Drop

The SAR should identify the operating environment experienced by the cask and the drop events (end/side/corner) that could result. Generally the design basis is established either in terms of the maximum height to which the cask may be lifted when handled outside the reactor site spent fuel building or in terms of the maximum acceleration that the cask could experience in a drop.

(b) Cask Tip-Over

Although cask system supporting structures may be identified and constructed as being important to safety thereby effectively precluding cask tipovers, the NRC considers that cask tip-over events should be analyzed. In some cases, cask tip-over may be determined to be a credible hazard. The associated analysis would be based on the conditions (e.g., heights and velocities) associated with the hazard. In absence of an identified hazard, the NRC has accepted non-mechanistic cask tip-over about a lower corner onto a hard receiving surface from a position of balance with no initial velocity. The NRC has also accepted analysis of drop of a cask with longitudinal axis horizontal that, with analysis of a vertical axis drop, could bound a non-mechanistic tip-over case.

(c) Fire

The conditions for a fire should be specified in the SAR. The NRC staff accepts that these conditions may consider the limited availability of flammable material at an ISFSI (e.g., only that associated with vehicles transporting and lifting the cask). The fire condition stated in the SAR should provide an envelope for subsequent comparison with site-specific conditions. That envelope may be based on a hypothesized fire situation. Alternatively, the envelope may be based on the maximum fire conditions for which the inherent resistance of the cask system provides protection.

(d) Fuel Rod Rupture

The regulations require that the cask be designed to withstand the effects of accidents conditions and natural phenomena events without impairing its capability to perform safety functions. The NRC has considered that the cask analyses for conditions resulting from design basis accidents and natural phenomena should assume 100% of the fill/fission gases are released from the fuel rods.

(e) Leakage of Confinement Boundary

Casks are designed to provide the confinement safety function under all credible conditions. Nevertheless, NRC staff considers that, for assessment purposes and to demonstrate the overall safety of the storage cask system, leakage of the confinement boundary and the release of radionuclides based on failure of 100% of the fuel rods should be analyzed. The SAR should identify this as a bounding release and that it has been caused by a non-mechanistic event.

(f) Explosive Overpressure

The conditions under which an ISFSI may be exposed to the effects of an explosion varies greatly with individual sites. Generally the explosive overpressure is postulated to originate from an industrial accident as opposed to attack by a saboteur. Each licensee must establish detailed plans for security measures to physically protect the ISFSI from sabotage in accordance with the applicable portions of 10 CFR Part 73. The effects of various sabotage methods on cask systems are evaluated separately by the Division of Fuel Cycle Safety and Safeguards in developing appropriate regulations in 10 CFR Part 73. Therefore explosive overpressures from sabotage events are not considered in this SRP. The extent to which explosive overpressure is addressed in the SAR directly affects the degree of site-specific review required. The effects at the ISFSI are of principal concern in the SAR, as opposed to descriptions of hypothesized causes. Design parameters for blast or explosive overpressures should identify pressure levels as reflected ("side-on") overpressure, and provide an assumed pulse length and shape. This should provide sufficient information for licensees to determine if the effects of their site-specific hazards are bounded by the cask system design bases.

(g) Air Flow Blockage

For storage systems with internal air flow passages, the applicant must consider blockage of air inlets and outlets in an accident condition. NRC staff considers the assumption of total blockage of all vents to be appropriate in determining periodic inspection intervals for the cask installations.

(4) Natural Phenomena Events

(a) Flood

The SAR should establish a design-basis flood condition. This flood condition may be determined based on the criteria that the cask cannot tip over and that the yield strength of the cask material cannot be exceeded. Alternatively, the SAR can show that credible flooding conditions have negligible impact on the cask design. Further analysis of this event, if appropriate, can be treated on a site-specific basis. If the SAR establishes parameters of a design basis flood, all of the potential effects of flood water and ravine flood by-products should be recognized. The most serious flood consequences can involve effects such as blockage of ventilation ports by water and silting of air passages. Other potential effects include scouring below foundations and severe temperature gradients as may result from rapid cooling because of immersion.

(b) Tornado

NRC staff accepts the design basis tornado wind loading based on Reg. Guide 1.76 (Region 1)[3] and tornado missile impacts based on NUREG-0800, Section 3.5.1.4[4]. Design criteria should be established for the cask based on this wind loading and missile impact. The cask should not tip over and that the capability to perform the confinement safety function is not impaired. The NRC considers that tornados and tornado missiles could occur without warning. In general, the effects of a tornado missile bound that of a light general aviation airplane crashing on an ISFSI facility.

(c) Earthquake

The SAR must state the parameters of the DBE. For ISFSIs at reactor sites, this is equivalent to the "Safe Shutdown Earthquake" (SSE) used for analyses of nuclear facilities, under 10 CFR Part 50. An analysis for "Operating Basis Earthquake" (OBE) is not required for a 10 CFR Part 72 SAR. Cask tipover accidents are analyzed. Tip-over caused by an earthquake should not be a credible event.

(d) Burial under Debris

Debris that may result from natural phenomena or accidents that may affect cask system performance important to safety may be addressed in the SAR or may be left to the site-specific application. Such debris can result from floods, wind storms, and mud slides. The principal effect is typically on thermal performance.

(e) Lightning

Lightning typically has negligible effect on cask systems, however, the requirements of the Lightning Protection Code and National Electric Code should be applied to the design of the cask system structures. These codes should be cited as part of the general design criteria for the cask system (see Section 2, II.3.a, above). Lightning should also be addressed as a natural phenomenon in the SAR if cask system performance may be affected by lightning because of impairment of a component that is important to safety.

(f) Other

Part 72 identifies several other natural phenomena events that should be addressed for spent fuel storage. These include seiche, tsunami, and hurricane. The SAR may include these as design basis events, show that their effects are bounded by other events, or be silent. If they are not addressed in the SAR and they are applicable to a site, the safety analysis would be required for licensing the site-specific installation.

3. Design Criteria for Structures, Systems, and Components Important to Safety

Because Regulatory Guide 3.61 does not clearly distinguish between the principal design criteria that should be presented in Chapter 2 of the SAR and the other design criteria that should be deferred to follow-on sections, the applicant may take one of several approaches. Chapter 2 of the SAR may discuss these criteria in general terms (similar to the wording in II.3 above), with details provided in later sections. Alternatively, selected (or all) detailed criteria may be presented in Section 2. Generally, the latter approach is selected. Consequently, a complete review of the design criteria needs to address the SAR as a whole.

In following sections of SRP, design criteria applicable to each functional area (structural, thermal, etc.) are discussed in detail in that section, without regard to those that may have been presented in Section 2.

Cask system components that are to be used in facility areas subject to 10 CFR Part 50 review must satisfy both the requirements in 10 CFR Part 72 (with review as guided by this SRP) and 10 CFR Part 50 (with review guided by NUREG-0800 and applicable portions of Regulatory Guide 3.53[3]). Acceptance of the cask system in areas covered by the 10 CFR Part 50 license requirements is not covered by this SRP for Part 72 approval. If the reviewer knows that the cask system will be used at a reactor site, contact should be made with the Nuclear Reactor Regulation (NRR) Project Manager, to inform him/her of a potential heavy loads issue.

Regardless of where the descriptions and associated criteria are located in the SAR, include a summary description and evaluation of the safety protection systems in the Design Criteria section of the SER. The system descriptions should describe the functions of the various system components in providing confinement, cooling, subcriticality, radiation protection of the public and workers, and spent fuel retrieval. Summary criteria for the performance of the system as a whole in providing for these capabilities or functions should be described and evaluated.

Criteria relating to redundancy, and allowable levels of response under normal, off-normal, and design basis conditions and events should be described and evaluated. In general, no degradation in physical condition or functional performance should result from normal or off-normal-conditions. The design criteria on limits of permissible system response and degradation that may result from DBE should be evaluated against the capabilities to perform the principal safety functions. Considerations of permissible responses should include detectability and corrective actions that may be included proposed as conditions of system use.

Table 2-1 summarizes design criteria (and design bases) that should be generally be identified during the initial stages of the review. Obviously, this listing may vary depending on the details of the cask design.

VI. EVALUATION FINDINGS

Provide a summary statement similar to the following:

The staff concludes that the principal design criteria for the [cask designation] are acceptable in meeting the regulatory requirements of 10 CFR Part 72. This finding is based on a review which considered the regulation itself, appropriate Regulatory Guides, applicable codes and standards, and accepted engineering practices. A more detailed evaluation of design criteria and the assessment of compliance with these criteria are presented in Sections 3 through 14 of the SER.

Table 2-1 Summary of Design Criteria/Bases

Design Criteria (Specify Normal/Off-Normal/Accident, If Applicable)

Design Life **(License restricted to 20 years)**

Structural

Design Code Containment (e.g., ASME[a], AISC[b], etc.)	Non-containment
Basket	
Trunnions	
Storage radiation and protective shielding and enclosure	
Transfer radiation and protective shielding and enclosure	
Cooling structure or system	
Design Weight	
Design Cavity Pressure	Normal/Off-Normal/Accident
Response and Degradation Limits	Normal/Off-Normal/Accident

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[a] American Society of Mechanical Engineers

[b] American Institute of Steel Construction

Thermal

Maximum Design Temperatures
Cladding 5-yr Cooled Fuel (As Applicable)
10-yr Cooled Fuel
Other Components

Solar Insolation Side/Top/Bottom

Fill Gas

Confinement

Method of Sealing

Maximum Leak Rates
Primary Seals
Redundant Seals
Cask Body

Monitoring System Specifications

Retrievability

Normal and Off-Normal

After DBE and Conditions

Criticality

Method of Control
(Geometry, Fixed Poison, Borated Pool Water)

Minimum Boron Concentration Fixed
Pool Water

Maximum k_{eff}

Burnup Credit (None currently permitted)

Radiation Protection/Shielding

Maximum Dose Rate

Confinement Cask Surface Normal/Off-Normal/Accident
Position

Exterior of Shielding Normal/Off-Normal/Accident
Transfer Mode Position
Storage Mode Position

Individual Workers Normal/Off-Normal/Accident
Dose Rate
Annual Dose

ISFSI Controlled Area Boundary
Normal/Off-Normal/Accident Dose Rate
Annual Dose

Design Bases

Spent Fuel Specifications
Type

Configuration/Vendor

Enrichment

Type of Cladding

Assemblies/Cask

Decay Heat/Assembly
5-yr Cooled Fuel
10-yr Cooled Fuel, etc.

Gas Volume (@ Temperature)

Fuel Condition/Damage Allowed

Gamma/Neutron Source Terms
Fuel
Assembly Hardware

Control Components

Normal Design Event Conditions

Ambient Temperature
Maximum
Minimum

Loading
(Wet/Dry)

Storage Handling Orientation
(Vertical/Horizontal)

Other Conditions Considered in III.2.b.(1)

Off-Normal Design Event Conditions

Summarize Events Considered in III.2.b.(2)

Design Basis Accident Design Events and Conditions

End Drop Lift Height (or Maximum Acceleration)

Side Drop Lift Height (or Maximum Acceleration)

Tip-Over (Acceleration, if applicable)

Fire Duration
Temperature

Other Events Considered in III.2.b.(3)
(As Applicable)

Design Basis Natural Phenomena Design Events and Conditions

Flood

Earthquake

Tornado

Other Events Considered in III.2.b.(4)

(As Applicable)

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask," February, 1989.

2. U.S. Code of Federal Regulations, Title 10, Part 71, "Packaging and Transportation of Radioactive Material."

3. U.S. Nuclear Regulatory Commission Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants,", April 1974.

4. U.S. Nuclear Regulatory Commission, NUREG-0800 (Section 3.5.1.4), "Standard Review Plan, Missiles Generated by Natural Phenomena," July 1981.

3.0 STRUCTURAL EVALUATION

I. REVIEW OBJECTIVE

This section evaluates the design and analysis of the structural performance of the storage cask system for normal, off-normal, and accident conditions. The objective of the structural review is to ensure that confinement, subcriticality, radiation shielding, and retrievability of the fuel are appropriately maintained with high assurance under all credible loads for normal, off-normal, and design basis accident and natural phenomena ("accident level") events and conditions.

II. AREAS OF REVIEW

All storage cask systems include a confinement cask. This may have internal components and integral external components. Some cask systems have multiple additional components that are subject to NRC evaluation and approval. In recognition of the diversity of the various cask system components, the review and the acceptance criteria are divided into four categories as appropriate:

- Confinement Cask
- Reinforced Concrete Components
- Other System Components Important-To-Safety
- Other Components Subject to Approval

The following areas of review are included within the above categories as appropriate:

a. "Scope"

b. "Structural Design Criteria and Design Features"

(1)" Design Criteria"

(a) "General Structural Requirements"
(b) "Applicable Codes and Standards"

(2) "Structural Design Features"

c. "Structural Materials"

d. "Structural Analysis"

(1)" Load Conditions"

(a) "Normal-conditions"
(b) "Off-Normal-conditions"
(c) "Accident -Level Events and Conditions"

(2) "Structural Analysis Methods"

(a) "Finite-Element Analysis"
(b) "Closed-Form Calculations"
(c) "Prototype or Scale Model Testing"
(d) "Structural Analysis of Specific Components"

(3) "Structural Evaluation"

(a) "Summary Structural Capability"
(b) "Fabrication and Construction"
(c) "Structural Compatibility with Functional Performance Requirements"

III. REGULATORY REQUIREMENTS (10 CFR Part 72)

1. Structures, systems, and components important to safety must meet the regulatory requirements stated in 10 CFR Part 72.24(c)(3), (c)(4) and 72.122(a),(b), and (c).

2. Radiation shielding, confinement, and sub-criticality must meet the regulatory requirements stated in 10 CFR 72.24(d), 72.124(a), 72.236(c), (d), and (l).

3. Storage system must be designed to allow ready retrieval of spent fuel without posing operational safety problems as stated in 10 CFR Part 72.122(h)(1) and 122(f).

4. As stated in 10 CFR Part 72.102(f), the design-basis earthquake must be equivalent to or exceed the SSE of a nuclear plant at sites previously evaluated under 10 CFR Part 100.

5. As stated in 10 CFR Part 72.24(c) and 72.236(g), the analysis and evaluation of the structural design and performance must demonstrate that the cask system will enable storage of spent fuel for a minimum of 20 years with adequate margin of safety.

6. Reinforced concrete structures may have roles in shielding, forming ventilation passages and weather enclosures, and providing protection against natural phenomena and accidents. The pertinent regulations include 10 CFR Part 72.24(c) and 72.182(b) and (c).

IV. ACCEPTANCE CRITERIA

As listed in Subsection II, this acceptance criteria section is also subdivided into four (4) categories: 1. Confinement Cask, 2. Reinforced Concrete Components, 3. Other Safety-Related System Components, 4. Other Components Subject to Approval.

Acceptance criteria for the design and analysis of storage cask systems and components are based on meeting the following relevant requirements.

1. Confinement Cask

The "confinement cask" structural review includes drawings, plans, sections, and technical specifications for those structural components comprising confinement barriers. The review also includes those structural and sealing interfaces and connections necessary to complete the confinement system (as defined in 10 CFR Part 72).

Confinement cask components that serve no structural function are evaluated to confirm that they do not impair the functioning of the confinement cask.

a. Scope

Structures, systems, and components important to safety must be described in sufficient detail to enable an evaluation of their structural effectiveness. Applicable codes and standards for structures, systems, and components important to safety must be identified.

Structures, systems, and components important to safety must be designed and fabricated to quality standards commensurate with the importance to safety of the function to be performed.

Structures, systems and components important to safety must be designed to accommodate the combined loads during normal, off-normal, accident, and natural phenomena events with an adequate margin of safety.

b. Structural Design Criteria and Design Features

(1) Design Criteria

NRC generally considers the design criteria listed below to be acceptable to meet the structural requirements of 10 CFR Part 72.

(a) General Structural Requirements

Confinement of radioactive material must be maintained under normal, off-normal, and accident conditions, and natural phenomenon events.

The cask and any racks within the cask must not deform under credible loading conditions such that the subcritical condition of the fuel or the retrievability of the fuel would be jeopardized.

Spent fuel cladding must be protected against gross rupture caused by degradation resulting from design or accident conditions. In addition to fuel damage caused by basket deformation, the fuel must not experience accelerations that would damage its structural integrity, so that subcriticality conditions or the retrievability of the spent fuel are not jeopardized.

The cask must be analyzed to show that it will not tip over or drop in its storage condition as a result of a credible natural phenomenon event. (This criterion is to preclude damage to an entire array. A tip-over or drop is always assessed as a bounding condition during handling operations.)

Radiation shielding, in the cask system, required for protection of the public or workers must not degrade under normal or off-normal-conditions or events. The shielding function may be acceptably degraded by DBE (e.g., loss of liquid neutron shielding resulting from a drop accident). However, the loss of function must be readily visible, apparent, or detectable. Procedures for testing shielding effectiveness should be included as part of procedures to be used after such DBE.

(b) Applicable Codes and Standards

The structural design, fabrication, and testing of the confinement system and its redundant sealing system should comply with an acceptable code or standard. Use of codes and standards that have been accepted by NRC for these expedites the evaluation process. The alternative use of other codes and standards, use of criteria composed of extracts from separate codes and standards, with overlapping scopes, or substitution of other criteria, in whole or in part, in place of acceptable published codes or standards requires extensive NRC review and may delay the evaluation process.

An accepted code for design, fabrication, and test of steel confinement casks is Section III of the ASME Boiler and Pressure Vessel Code.[1] NRC has accepted use of either Subsection NB or NC. Other design codes or standards may be acceptable depending on their application. NRC has accepted use of the applicable subsections of Reference 1, Division 1 for cask system components used within the confinement cask but not integrated with it. This includes the "basket" which is a structure used in casks to restrain and position multiple fuel elements.

NRC has accepted use of applicable subsections of Reference 1, Division 1, for structural external integral elements of the confinement (e.g. Subsection NF for integral supports).

NRC accepts use of Regulatory Guides 7.11 and 7.12[2] as bases for determining the potential for brittle fracture. These Regulatory Guides also incorporate a portion of NUREG/CR

1815[3] by reference.

The fatigue limits of the cask structural materials may be based on the provisions of the Reference 1 or the guidance provided in Regulatory Guide 7.6[2] Since casks are typically subjected to non-cyclic loads, fatigue may not be a significant concern.

(2) Structural Design Features

Descriptive information on the cask is presented in Chapter 1 and any additional information provided in Chapter 3 of the SAR. The drawings, figures, tables, and specifications included in the SAR documentation should fully define the structural features of the cask. The structural components of a storage cask may include: cask body, including an inner shell, an outer shell, and a lead gamma shield; inner lid and bolts; port covers and bolts; outer lid and bolts; inner and outer lids and vent port covers to be welded in place; neutron shields and shell; trunnions; fuel basket; and impact limiters, if used.

Coordinate with the confinement review to verify that the confinement boundaries are clearly identified. The confinement boundary includes the primary confinement vessel, its penetrations, seals, welds, and closure devices, and the redundant sealing system. Ensure that proper specifications for all welds have been provided and, if applicable, the bolt torques for closure devices have been adequately designed and specified.

Review the list of weights and calculation of the cask center of gravity and verify that the appropriate limiting cases are used in the structural evaluations.

Review the structural materials of the cask that are in direct contact with each other to verify that they will not produce a significant chemical or galvanic reaction.

Review confinement boundary weld designs for compliance with the design code used for the confinement boundary). Acceptable requirements are found in ASME Code Section III,[1] NB-3352 and NC-3352 "Permissible Types of Welded Joints", and NB-4240 and NC-4240 "Requirements for Weld Joints in Components".

NRC has previously accepted alternative confinement boundary weld designs that achieve equivalent structural integrity but do not meet all the provisions of NB-3352 or NC-3352 for full penetration welds, or do not meet the requirements for full volumetric non-destructive examination (NDE) volumetric examination (NB-5200 or NC-5200, typically for Category C welded joints). NRC has also accepted alternative designs for the welds of the head or flat end plate to the cylindrical portion of the confinement vessel. NRC has required the alternative designs to include redundant seals.

Welds must be well-characterized on drawings using standard welding symbols and/or notations as discussed in American Welding Standard (AWS) A2.4[4].

c. Structural Materials

The information provided on materials must be consistent with the application of the accepted design criteria, codes, standards, and specifications selected for the storage cask system. For example, if Section III of the ASME code[1] is used for the design criteria, the materials selected for the cask must be consistent with those allowed by the ASME code subsection used for design. Acceptable requirements are ASME-adopted specifications given in Section II, Part A "Ferrous Metals", Part B "Nonferrous Metals", Part C "Welding Rods, Electrodes, and Filler Metals", and Part D "Properties.

The review of materials includes: sources of the information; properties that are used in the structural evaluation (including those that affect performance under both static and dynamic loadings for normal, off-normal, accident conditions, and natural phenomena events); and suitability for the proposed life of the ISFSI. Preferred sources are industry and government codes, standards, and specifications. Other sources should be reviewed for applicability and acceptability, such as manufacturer's test data and handbooks. Published articles, research reports, and texts are typically not primary sources and have generally not been accepted by NRC as sources for material properties.

The review is to determine the acceptability of all materials that have a structural role for confinement system structures and other structures important to safety (e.g. basket, impact limiters, and shielding). The review includes evaluating the suitability of the materials for the structural application and material properties that affect structural design and evaluation, over the approved period of use. The potential for corrosion should be analyzed. Appropriate, corrosion allowances should be established and used in the structural analyses. The review includes the static and dynamic (where appropriate) stresses and the limits used for the structural design and evaluations. The review evaluates acceptability of the sources given for the information.

For Section III,[1] Subsection NB or NC applications, additional material requirements regarding examination prior to fabrication, testing and analyses, and traceability are applicable. Compliance with the requirements of the following Section III paragraphs, or their equivalent, must be acknowledged in the SAR: NB-2121 or NC-2121 (Permitted Material Specifications), NB-2130 or NC 2130 "Certification of Material", NB-2500 or NC-2500 "Examination and Repair of Pressure Retaining Material", and NB-2400 or NC 2400 "Welding Material".

The SAR should include tables with material properties and allowable stresses and strains (as appropriate). Verify that the properties used are appropriate for the load conditions of interest (e.g., static or dynamic, impact loading, hot or cold temperature, wet or dry conditions). Ensure that any restrictions on temperature or environmental conditions for the materials are considered. Ensure that any restrictions are addressed in Section 12 of the SAR. Determine that the sources of all material properties are clearly referenced and that the sources are acceptable.

Coordinate with the thermal review to determine the appropriate temperatures at which allowable stress limits should be defined. For most cask materials, the stress limit should be defined at the maximum temperature for that material as established by the SAR thermal analysis.

The potential for brittle fracture must be reviewed, especially for cask system components that may be subject to impacts during exterior handling and transfer operations. The potential for brittle fracture of some components important to safety has resulted in conditions of use that preclude transfer operations during extremely low temperatures. Ensure that any assumptions about internal heat generation for the brittle fracture analysis are based on the maximum storage life and the possibility of a partial load in the cask. Verify that any restrictions on cask handling at low temperatures are addressed in Chapter 12 of the SAR and included in Section 12 of the SER.

If the cask has impact limiters, their nonlinear impact characteristics should be thoroughly tested and verified. The crush characteristics and properties of the limiters in the directions that are to be used should be described and tabulated.

d. Structural Analysis

(1) Load Conditions

The following load conditions are necessary to meet the structural requirements of 10 CFR Part 72: normal operating conditions, accident conditions, and natural phenomena events (accident-level events and conditions).

NRC recognizes an additional set of conditions: "off-normal." These involve situations other than normal-conditions or events that may reasonably be expected to occur during the life and use of the system. The system should experience no permanent deformation or degradation in response to off-normal-conditions. The system may experience some deformation but no loss of safety function capability in response to accident-level events and conditions.

(a) Normal-conditions

Normal-conditions and events are those associated with the normal range of environments and with operations and storage, involving the cask system with nuclear material. The assumed limits of normal use environments are stated to support an evaluation of the cask system suitability for use at a specific site.

Loads normally applicable to confinement cask are weight, internal/external pressure, and thermal loads caused by operating temperature. The loads experienced may vary during loading, preparation for storage, transfer, storage, and retrieval operations. The weight is the maximum or design weight of the cask as it is stored and loaded with spent fuel. However, depending on the operation and procedures the weight should also include water fill. All orientations of the cask body and closure lids during normal operations and storage conditions should be evaluated.

Internal pressures result from hydrostatic pressure, cask drying and purging operations, filling with non-reactive cover gas, out-gassing of fuel, and temperature rise. Temperature variations and thermal gradients in the structural material may cause additional stresses in the cask and closure lids. Coordinate with the thermal review to determine conservative (or enveloping) values and combinations of the cask internal pressures and temperatures for the hot and cold conditions. Temperature gradients calculated in Chapter 4 of the SAR are used to determine thermal stresses. If the confinement system has several enclosed areas, all areas may not have the same internal pressures. In some casks, enclosed areas consist of the cask cavity and the region between the inner and outer lids.

The following two evaluations are required: (1) weight plus internal pressures and thermal stresses from the hot condition and (2) weight plus internal pressures and thermal stresses from the cold condition.

(b) Off-Normal-conditions

Review the off-normal-conditions and events identified in the SAR for those that affect the confinement cask structure. The confinement cask components should satisfy the same structural criteria as required for normal-conditions, as discussed in (a), above.

Off-normal-conditions should be identified. These involve conditions and events that may reasonably be expected to occur during the life of the cask system at a site. Environmental limits are stated to support comparison of the cask system design bases with specific site environmental data. Off-normal-conditions and events can involve potential mishandling, simple negligence of operators, equipment malfunction, loss of power, and severe weather (short of extreme natural phenomena). The review should identify and evaluate off-normal events and conditions described in Section 11 of this SRP.

(c) Accident-Level Events and Conditions

Guidance is included below for review of structural response to some accident-level events and conditions. These are not all of the potential events, some of those discussed below may not be applicable to all cask systems. The review to identify and evaluate accident-level events and conditions is described in Section 11 of this SRP.

The following accidents and accident-level events must be evaluated in the SAR:

(i) Cask Drop and Tip-over

The SAR should identify the operating environment experienced by the cask and the drop events (end/side/corner) that could result. Generally the design basis is established in terms of the maximum height to which the cask is lifted outside the spent fuel building or in terms of the maximum acceleration that the cask could experience in a drop.

A cask tip-over should be analyzed regardless of credibility of occurrence. The associated analysis should be based on the condition (e.g. heights and velocities) associated with the event. NRC will accept cask tip-over about a lower corner onto a hard receiving surface from a position of balance with no initial velocity. NRC has also accepted analysis of drop of a cask with longitudinal axis horizontal which, with analysis of a vertical side drop, could bound a non-mechanistic tip-over case.

(ii) Explosive Overpressure

Explosion-caused overpressure and reflected pressure may result from explosion hazards associated with explosives and chemicals transported by rail or on public highways, natural gas pipelines, and vehicular fires of equipment used in the transfer of casks. Explosions may result from detonation of an air-gaseous fuel mixture. With the exception of a transfer vehicle accidents, the explosion hazards are typically similar to those for facilities subject to 10 CFR Part 50 reviews. Note, this explosive Overpressure is not meant to be that from the design basis radiological sabotage event. Physical security plans, established by licensees, working in conjunction with the cask design to protect the spent fuel against such a threat.

The review for site-specific explosion hazards would be left for the license application for the specific site if explosions are not addressed in the SAR. Alternatively, the SAR can state the level of overpressure, reflected pressure, and/or pressure differentials assumed to result from an explosion, to serve as the quantitative envelope for future comparison with hazards for specific site installations. The pressure criteria for the assumed design basis wind or tornado may also serve as an envelope for the explosive pressures, for comparison with actual site hazards.

If the SAR includes bounding explosion effects for which the cask system is to be approved, determine that there are also structural analyses for those effects for cask system structures that may be affected. As an accident-level event, the structures are not required to survive an explosion's effects without damage or permanent deformation. The maximum response should be determined and should be shown in the SAR documentation. It should be sufficiently low that the component's capability to perform its safety functions is not impaired. Post-event inspection and remedial actions may be necessary and should be outlined in the SAR.

(iii) Fire

Potential fire conditions are addressed in Section 4 of this SRP. Structural evaluation considerations for fire include increased pressures in the confinement cask, change in material properties (e.g., temporary loss of strength at elevated temperatures and permanent loss of strength because of annealing), stresses caused by different coefficients of thermal expansion and/or temperatures in interacting materials, and physical destruction (e.g., surfaces of concrete exposed to intense or prolonged high temperatures).

Review and evaluate the treatment of structural effects of the presumed fire in the SAR. Evaluate the appropriateness of analysis of structural effects for the assumed parameters of the design basis fire. Confirm that the confinement cask pressure capacity is based on the properties of the cask material at the temperature resulting from the fire.

Fire parameters included in 10 CFR Part 71 have been accepted for characterizing the heat transfer for fire during storage. Exterior spalling of concrete that may result from a fire or other high temperature condition followed by application of water or rain is considered to be acceptable and does not need to be estimated or evaluated. Such damage is readily detectable and appropriate recovery or corrective measures may be presumed. NRC accepts concrete temperatures that exceed the temperature limits of American Concrete Institute (ACI) 349[5] for accidents if the temperatures result from a fire. However, corrective actions may need to be taken for continued safe storage.

(iv) Flood

Review the evaluation of the cask system design for the structural consequences of a flood event. The SAR may stipulate an assumption that the cask system not be used at any site where there is potential for flooding. In this case, the cask would have to be place on a reactor site at a location sited above the maximum probable flood. This condition should be specified in Section 12. Alternatively, a license application for a site with flooding potential would be required to provide a full analysis.

Two possible structural consequences of floods are that (1) a vertically stored cask may tip over or translate horizontally (slide) due to the water velocity, and (2) hydrostatic pressure will exceed the capacity of the cask. The water critical velocity and hydrostatic pressure may be stated as bounds for the SAR flood analysis.

NRC accepts application of the requirements of ANSI/ANS 57.9,[6] Section 6.17.4.1 to the flood case for overturning and sliding of stored confinement casks and other cask system structures (safety factor of 1.1 for the accident-level cases). The basis for estimation of lateral pressure on a structure, because of water velocity, should be stated. NRC accepts use of Reference 8 for estimation of drag coefficients and net lateral water pressure. An approach for calculating the velocity corresponding to the cask stability limit is to assume that the cask is pinned at the outer edge of the cask bottom, that the cask rotates about that outer edge, and that the pinned edge does not permit sliding. The overturning moment from the velocity of the flood water can be compared to the stability moment of the cask (with buoyancy considered). The structural consequences of the flood event are typically bounded by analyses for the drop or tip-over accident cases.

Review the analysis of the confinement cask for flood hydrostatic pressure. The analysis should include the combined effects of weight, external hydrostatic pressure, internal pressure(s), and thermal stress. Resistance of the confinement cask to flood hydrostatic pressure should be analyzed in accordance with Reference 1, Subsection NB or NC (depending on the subsection used for design).

Additional consequences of floods are potential scouring under a foundation, damage to access routes, temporary blockage of ventilation passages with water, blockage of ventilation passages and interstitial spaces between the confinement cask and shielding structure with mud, and steep temperature gradients in the shielding structure and confinement cask. While the consequences of these conditions are analyzed in the SAR, when siting an ISFSI, the Licensee should consider these factors.

(v) Tornado Winds

Review the SAR to ensure that potential structural consequences of design basis tornado or extreme wind effects are included. Review the load combination analyses for acceptable inclusion of tornadoes and tornado missiles.

Confinement casks may be vulnerable to overturning and/or translation due to the direct force of the drag pressure while in storage or during transfer operations. ANSI/ANS 57.9[6] provides acceptable criteria for resistance to overturning or sliding.

Confinement casks are generally not vulnerable to damage from overpressure or negative pressure associated with tornadoes or extreme winds. However, they may be vulnerable to secondary effects, such as wind-borne missiles (see (vi), below) or collapse of a weather enclosure. Tornado or extreme winds have been a governing load condition in prior reviews for major structures that form part of an ISFSI system (these structures may provide for shielding, cooling paths, and/or transfer and storage operations).

Tornadoes typically produce the greatest "design level" wind effects for American sites. However, there are some potential American sites at which high winds may be more severe than the credible tornado. A SAR for a limited set of potential sites could be based on high wind effects. If the certificate is to include proven resistance to tornadoes or extreme winds, the SAR documentation must identify the wind levels (e.g., in miles or kilometers per hour), source (tornado or high wind), and specific wind-driven missiles (shape, weight, and velocity) for which the design is to be evaluated.

Tornado parameters are provided by Regulatory Guide 1.76 NRC accepts use of ASCE 7[7] for conversion of wind speed to pressure and for shape factors for typical building shapes. Conversion of tornado or other wind speeds to pressure in the SAR documentation should assume that the cask system is at sea level. The source for coefficients of drag used to compute net forces on objects of other shape such as, Hoerner, S. F, "Fluid-Dynamic Drag," 1965,[8] should be cited

NRC accepts the use, for the design tornado wind pressure, of the pressure derived from conversion of wind speed, without gust or importance factors, for tornadoes. If the design basis wind is caused by extreme winds, NRC accepts the computational approach given in ASCE 7, for determining pressures. This approach adds gusts, importance, exposure, and height above ground to the analysis. The computational approach of ASCE 7, has been accepted for normal and off-normal wind loadings.

Tornadoes and high winds can produce a significant negative pressure differential between interior spaces and the outside. This is a function of wind speed and factors relating to the structure. The magnitude of negative pressure depends on other parameters of the tornado or wind, and on wall pressure coefficients (as expressed in ASCE 7). There is no need for negative pressure to be separately stated to establish an envelope for approval. Negative pressure is insignificant with regard to confinement cask pressure analysis.

In the past, NRC has not considered that there will be sufficient warning of tornadoes that operations such as transfer between the fuel pool facility and storage site may never be exposed to tornado effects. Overturning during on-site transfer is typically a potential or mandated design basis event. The tornado analysis should determine if tornado-caused overturning is bounded by drop and tip-over cases. The SAR documentation should show that the cask system design will continue to perform its intended safety functions (criticality, radioactive material release, radiation exposure, and ready retrievability).

(vi) Tornado Missiles

Review the evaluation of the cask system design for the structural consequences of wind driven missile impact. Tornado missiles are described in Regulatory Guide 1.76[2] and NUREG-0800[9]. The SAR should define the missile parameters for which the cask system is to be evaluated. Missile effects that should be addressed are those that may cause a tip-over, which is the bounding case, and those that may cause physical damage by impact. The damage should not result in unacceptable radiation dose or ready retrievability of the fuel.

NRC has accepted use of the analytical approaches given in ORNL-NSIC-5, Volume 1, Chapter 6,[10] for estimating the potential effects of missile impact on steel sheets, plates, and other structures. Further guidance on analytical approaches acceptable for use with ISFSI design is in NUREG-0800,[9] Section 3.5.3, "Barrier Design Procedures." NRC has accepted use of Kennedy, R.P., "A Review of Procedures for the Analysis and Design of Concrete Structures to resist Missile Impact Effects,"[11] for analysis and design of reinforced concrete structures to resist missiles.

Cask systems are not required to survive missile impacts without permanent deformation. However, the maximum extent of damage from a design basis event must be predicted and should be sufficiently limited. The capability of the structures, systems and components important to perform their safety functions should not be impaired.

(vii) Earthquake

Review the evaluation of the cask design for the structural consequences of the earthquake event. As stated explicitly in Part 72, the DBE must be no less than the SSE for the reactor. Cask designs must satisfy the load combinations that include earthquake, including those for sliding and overturning in ANSI/ANS 57.9,[6] Section 6.17.4.1). The review should ensure no tip-over, drop, or impacts between storage casks results from an earthquake. A tip-over accident (an unlikely but credible event) is considered a design basis event and the fact that the effects of a tip-over are analyzed provides defense-in-depth.

The SAR documentation should include analysis of the potential for impacts between components of the cask system. These could include contact between the confinement shell and its inner components or its outer shield, and the rocking and fall back of a vertically or horizontally oriented confinement cask on its supports.

Cask systems are not required to survive DBE without permanent deformation. However, the maximum extent of damage from a design basis event must be predicted. The capability to provide principal safety functions should not degrade.

(2) Structural Analysis Methods

Structural analysis, stresses, and stress combinations resulting from different loads are reviewed. The reviewer should be satisfied that analytical approaches and tools are acceptable, and that the analytical tools are used appropriately. Computations should have been performed and internally reviewed by the applicant under an acceptable quality assurance procedure. The scope of the review does not include performing detailed parallel computations (such as finite element analyses) to validate submitted computations or their results. The reviewer may perform separate, less extensive calculations when these could most readily evaluate suspected problems.

Analysis of stresses and stress combinations resulting from different structural loads should be in accordance with the subsection of Section III of the ASME Code[1] used for the design of the component.

Subsections NB (Class 1 components) and NC (Class 2 components) of Reference 1 provide requirements for categorizing stresses and for determining allowable stress limits for confinement casks. Definitions, for stress categories and stress intensity limits for normal and off-normal operating conditions, are provided in Subsections NB and NC. For level D or accident conditions, Appendix F provides definitions of the stress intensity limits. In accordance with these definitions, stress intensity is based on the maximum shear stress theory for ductile materials. Since the maximum shear stress is not identical to the maximum octahedral shear stress, octahedral shear stresses should not be compared with the stress intensity limits. Values for the stress intensity limits are defined in Appendices I and III of the ASME code. Stresses resulting from inertial and pressure loads should be considered as primary stresses since they are not self-limiting. Thermal stresses resulting from temperature gradients may be considered as secondary stresses if they are self-limiting and do not cause structural failure.

(a) Finite-Element Analyses

Because of the complexity of many of the structural load conditions, structural design computations are often performed with finite-element analysis.

The finite element analyses should be performed with a general purpose program that is well benchmarked and widely used for many types of structural analyses. Examples of this type of program include ANSYS[12] and NIKE3D.[13] Codes, such as SCANS [14]and CASKS [15]are confirmatory tools and are not applicable for primary analyses in the SAR because they have simplifying assumptions regarding cask geometry, materials, and structural behavior.

When possible, solutions from finite-element analyses should be compared with closed-form calculations. While it is unlikely to exactly duplicate the complex load condition being analyzed with the finite element program, portions of the evaluation can be verified. For example, the stress state caused by internal pressure in the cask can be checked with the formulas for the stress in a cylinder with end-caps.

Linear material properties should be used in the analyses to be consistent with the provisions of Section III of the ASME Code[1]. Inelastic material properties can be used for cask components that are not stress-limited and that respond inelastically to the load conditions for storage casks. Sources for the inelastic material properties should be provided in the SAR.

Lead shielding, which is typically not stress-limited, can be modeled either with elastic or inelastic properties. Since the yield stress of lead is relatively low, a low elastic modulus is used if the lead is modeled with elastic properties. An appropriate plasticity model of lead can be used to account for its inelastic behavior.

Nonstructural components of the confinement cask are generally not included in finite element models. Any influence they may have on the structural performance of the cask should be included. Possible influences by the nonstructural components include their inertial weight, restraint to motion of the structural components, and localized influence on load applications because of geometrical effects.

Bolted connections can be modeled either discretely or with contact conditions. To discretely model the bolted connections, appropriate element types and material properties should be used. With contact conditions, the interfaces joined by the bolts can be modeled as tied.

The number of discrete finite-elements used in the model should reflect the type of analysis being performed. Regions in the model of high stress or displacement should have a higher number of elements than regions that have a nearly equilibrated state of stress or which are in uniform stress field. Sensitivity studies should be conducted to determine the appropriate number of nodes or elements for a particular model.

(b) Closed-Form Calculations

Closed-form calculations should be performed for relatively simple structural load conditions or conditions for which a formula has been developed. These types of calculations can be used for analyses involving principles of conservation of energy and comparisons of overturning moments. A source of closed-form equations accepted by NRC is "Roark's" Formulas for Stress and Strain.[16] Closed-form calculations are typically used to check the results of finite element analyses.

Use of a particular equation or formulation for the load conditions should be justified as appropriate. The most important aspect of the calculations to evaluate is the basis for the assumptions used in the calculations. In many cases, either portions of the cask are not included in the calculation or the load conditions are idealized.

Linear material properties should be used in the analyses to be consistent with the provisions of Section III of the ASME Code.[1] Linear elastic analysis should be the basis for all closed form calculations.

(c) Prototype or Scale Model Testing

Prototype or scale model testing may be performed in lieu of impact analysis for cask drop conditions or to support the analytical results of an analysis. However, use of scale model testing to directly demonstrate that the cask design meets the regulatory safety-performance requirements may be difficult, because leakage rates and other radiological limits may not correspond to the same scaling factor used for the scale model. Consequently, impact tests aimed at demonstrating regulatory compliance by test alone usually are prototypical tests. A sufficient number of tests should be performed to cover all design impact conditions and other uncertainties such as the receiving target surface hardness.

Drop tests can be performed to obtain an equivalent static-load to be used for a quasi-static analysis of the cask. Drop tests can also be used to obtain key data, such as the spring stiffness of the target surface, which is then used to perform a dynamic analysis of the cask.

A scale model must properly simulate the distribution of the loadings (weights), the geometry (dimensions), and the material properties of the cask. Any structural parts of the cask that are omitted in the scale model should be justified and the effects on the results of the test adequately discussed.

The applicant should develop a test plan to identify the test conditions, parameters to be measured during and after the test, and the test acceptance criteria.

(d) Structural Analysis for Specific Cask Components

A few specific examples of structural analysis for some of the confinement cask components are listed below:

(i) Trunnions

The trunnion design, its connection with the cask body, and the cask body in the local area around the trunnions are reviewed. The trunnions can be either a non-redundant or redundant design. In either case, the design should meet the requirements of ANSI N14.6[17] for critical loads or the requirements of NUREG-0612.[18] Non-redundant lifting systems should be designed for not less than 6 times and 10 times the design lift weight of the load cask without producing stresses in excess of the material yield strength and ultimate strength, respectively. Redundant lifting systems should be designed for not less than 3 times and 5 times the design loaded lift weight of the cask, without producing stresses in excess of the material yield strength, and ultimate strength, respectively. Acceptance testing requirements for trunnions are discussed in Section 9 of this SRP.

For a typical trunnion design, the maximum stress occurs at the base of the trunnion and is a combination of bending and shear stresses. A conservative technique for computing the bending stress is to assume that the lifting force is applied at the cantilevered end of the trunnion and that the stress is fully developed at the base of the trunnion. If other assumptions are used, they should be justified. The stresses and forces in the trunnion connections with the cask body and in the cask body near the trunnions should be evaluated.

(ii) Fuel Basket

Review the fuel basket design for analysis of the combined effects of weight, thermal stresses, and cask-drop impact forces. The weight supported by the basket should be the maximum or design weight of spent fuel. All credible potential orientations of the cask and basket during cask drop should be evaluated. End or side drops typically produce the greatest structural demand on various basket components. In the end drop, the basket is supported by the bottom of the confinement cask cavity upon impact. In the side drop the basket structure and points of contact with the confinement cask must support the mass of the basket and loaded fuel.

Two approaches that have been accepted for analyzing the structural capability of the basket to acceptably survive cask drop are described below:

The first approach uses dynamic analyses in a two-step process. First, a dynamic analysis is performed of the cask body impacting a target surface. The response of the cask body is reviewed for the maximum response from the cask drop impact. This maximum response can then be translated into a forcing function which then can be applied to the supporting contact points of an appropriate model of the fuel basket.

The second approach uses a quasi-static analysis of the basket subjected to the peak inertial load from the cask-drop impact. The peak load should be applied, using an appropriate model of the basket with the location(s) most vulnerable to the impact. Support provided by the inside surface of the cask cavity should be represented by the appropriate boundary conditions on the outside edge of the basket. The peak inertial load should be conservatively selected such that it bounds the possible inertial loads resulting from a cask-drop accident onto the bounding target surfaces. If applicable, the inertial load should also account for dynamic amplification effects by using a dynamic amplification factor.

The buckling capacity of the basket should be evaluated. Acceptable guidance for evaluating the buckling capacity of cask basket materials is given in the ASME B&PV Code[1], Section III, and in NUREG/CR-6322[19]. Selection of the appropriate end conditions in the buckling capacity equations should be based on sensitivity studies. These studies can bound the range of conditions, which are typically either fixed for a welded connection or free if there is no rigid connection.

(iii) Closure Lid Bolts

Review the design analysis for the closure-lid bolts to ensure that the combined effects of weight, internal pressure(s), thermal stress, O-ring compression force, cask impact forces, and bolt pre-load are properly included. Typically, pre-load and bolt torque are specified for the closure bolts. These are based on bolt diameter and the coefficient of friction between the bolt and the lid. Externally applied loads such as the internal pressure and impact force produce not only direct tensile bolt force but also an additional prying force caused by lid rotation at the bolted joint. The tensile bolt force obtained by adding together the pressure loads, impact forces, thermal load, and O-ring compression force is compared with the tensile bolt force computed from only the pre-load and operating temperature load. The larger one of the two tensile forces calculated above controls the design. The maximum design bolt force is then obtained by combining the larger direct tensile bolt force with the additional prying force. The weight is from the maximum or design weight of the closure lids and any cask components supported by the lids. Acceptable methods for analysis of closure bolts are given in NUREG/CR-6007.[20]

The bolt engagement lengths should be reviewed. If the lids are fabricated from relatively non-hardened materials, threaded inserts may be used in the closure lids to accommodate the hardened material of the bolts.

(3) Structural Evaluation

(a) Summary Structural Capability

Review the structural analyses for summary tables or statements about margins of safety or compliance with ASME Code stress limits, overturning, and other criteria. The comparisons of capability versus demand for the various applicable loading conditions should be in the terms used by the design code (e.g., type of stress). Margins of safety should be included based on the comparisons of capacity versus demand for each of the structural components analyzed. The minimum margin of safety for any of the structural sections of a component should be included for the different load conditions.

(b) Fabrication and Construction

NRC has accepted fabrication of confinement casks in accordance with Reference 1 . Any deviations from use of the subsection of Reference 1 used for design as the code for construction, fabrication, or assembly of the confinement cask must be explicitly justified in the SAR. This must be acceptable to NRC. The review should especially address any specifications for preparation for welding, materials to be used in welds, performance of welding, and inspection of welds that do not fully comply with Reference 1.

Welding procedure qualifications and welding performance qualifications should be in accordance with the requirements of ASME B&PV Code,[21] Section IX. For confinement welds, the SAR documentation should include the bases for detailed welding procedure specifications (WPSs) that identify acceptable ranges of essential welding variables (listed in the ASME Code, Section IX, for all approved welding processes). The welding variables should be recorded as Quality Assurance records during production runs. All welds should be performed by pre-qualified personnel in accordance with written procedures.

Testing of weld integrity may be by a combination of ASME-approved weld test techniques, which do not necessarily result in full radiographic examination but some volumetric inspection (e.g. Ultrasonic (UT)) may be necessary.

(c) Structural Compatibility with Functional Performance Requirements

The SAR documentation should be reviewed to confirm that the structural design of the structure provides for satisfactory functional performance. This includes operating suitability within specified limiting conditions and satisfying the basic safety criteria under all of the credible events and environmental conditions.

The confinement system and other structures important to safety are to have sufficient structural capability, for every applicable section to withstand the worst-case loads, under accident-level events and conditions, to successfully preclude the following:

- Unacceptable risk of criticality
- Unacceptable release of radioactive materials to the environment
- Unacceptable radiation dose to the public or workers
- Significant impairment of ready retrievability of stored nuclear materials

This does not require that all confinement system and other structures important to safety survive all design basis accidents and extreme nature phenomena without any permanent deformation or other damage. Some load combination expressions for the DBE and conditions for structures important to safety permit stress levels that exceed yield. The SAR should include computations of the maximum extent of potentially significant transient deformations and any permanent deformations, degradation, or other damage that may occur. These should be shown to be acceptable by computations, analyses, and/or tests acceptable to NRC.

Structures important to safety are not required to survive accident-level events and conditions to the extent that they remain suited for use for the life of the cask system without inspection, repair, or replacement. If the life of structures important to safety may be degraded by DBE or conditions, there must be requirements and procedures for determination and correction of the degradation, or other acceptable remedial action.

Review the operational controls and limits to ensure that adequate restrictions on cask handling and operations are included to preclude the possibility of damage to the structure or the confined nuclear material. Operating controls and limits (reviewed under Section 12 of this SRP) should be included in the SAR and SER, that describe actions to be taken and inspections that are to be conducted if events that may cause such damage occur.

2. Reinforced Concrete Components

The AREAS OF REVIEW and structural evaluation guidance applicable to reinforced concrete (RC) components of the cask system is presented in this subsection and Subsection IV.1.d(1), (2), above.

RC structures subject to NRC evaluation include structures, systems, and components that are to be included in the approved cask system. These may be of concern because of their importance to safety (per 10 CFR Part 72.24 (c)) or safety function.

a. Scope

RC structures may have roles in providing radiological shielding, forming ventilation passages, weather enclosures, structural supports, access denial, foundations, earth retention, anchorages, floors, walls, movable shields, bulk fill, and protection against natural phenomena and accidents. Bulk fill may be within a weld or other enclosing structure to provide shielding or strength.

RC structures may be cast in place, cast at the site, or cast elsewhere. RC structures may also be combinations of cast in place and precast sections that are integrated by bolting, welding, fitting, grouting, or placing additional concrete at the site. They may also include concrete that may be cast as part of a composite confinement cask with metallic liner. This subsection does not address a metallic liner of a composite confinement cask, its closures, or its internal components.

Embedments and attachments to RC structures are analyzed as parts of the RC structure unless they are specifically addressed elsewhere in Section 3 of the SRP. Embedments and attachments are considered to include components that are cast in or grouted into the RC structure; inserts; embedded pipes and conduits; and lightning protection and grounding systems.

b. Structural Design Criteria and Design Features

(1) Design Criteria

(a) General Structural Requirements

All concrete used in storage cask system ISFSIs, and subject to NRC review, should be reinforced, regardless of the functional role or need for structural strength or integrity. The concrete specifications should state the design code or standard for RC applicable to its intended use and which is acceptable to NRC.

The structural design of the RC structures shall withstand the effects of credible accident conditions and natural phenomena events without impairing their capability to perform safety functions. The principal safety functions include maintaining subcriticality, containing radioactive material, providing radiation shielding for the public and workers, and maintaining retrievability of the stored fuel.

NRC has not required that exterior RC pavements used for vehicular traffic, parking, or equipment access to the ISFSI storage area be designed as important to safety. The SRP does not address design or evaluation of pavements that are not considered structurally integral with the foundation of an RC cask system structure that is subject to review. RC aprons that extend from a structure and are structurally integral with the structure are also elements of the foundation. As such they should be reviewed for compliance with the same code applicable for the attached RC structure. If a pavement incorporates points for fastening supports that are important to safety (as may be used for transfer operations) that section of the pavement necessary for the function should be designed as a foundation in accordance with ACI 349[5]

RC pads that support confinement casks in storage are not "pavements." They should be designed and constructed as foundations under the applicable code (ACI 318[22] or ACI 349,[5] see (b), below). The pads typically are not classified as important to safety. However, in some cases they may be determined to be important to safety. If this is the case, the potential for liquefaction or other soil instabilities due to vibrating ground motion should be considered and the pad designed and constructed in accordance with an applicable code or standard such as ACI 349.[5]

(b) Applicable Codes and Standards

Review the codes and standards identified in the SAR and their proposed application. Codes and standards that have been accepted for ISFSI RC structures are identified below, by application-- concrete containments, RC structures important to safety but not within the scope of ACI 359,[23] other RC structures subject to NRC approval, and steel attachments to RC structures.

ANSI/ANS 57.9[6] is generally applicable to ISFSI design and construction (with exceptions for confinement casks). Table 3-1 includes extracts of ANSI/ANS 57.9[6] that are especially applicable to RC structure design and construction. The table also includes corresponding evaluation guidance for review of the SAR documentation.

NRC has not accepted use of a set of criteria that has been derived by selection of criteria from more than one code, except when the greatest minimum requirements of both codes are met. However, in recognizing a graded approach to quality assurance, NRC has approved the use of ACI 349[5] for design and material selection for RC structures important to safety (not confinement), but has allowed the optional use (instead of ACI 349[5]) of ACI 318[22] for construction, as described in this subsection.

Review of the SAR is based on nuclear safety and the other components and features of the cask system. There are codes other than those discussed herein that may be applicable to the design and construction of the cask system. It is acceptable that such codes (e.g., the National Fire Protection Association (NFPA) Electric, Life Safety, and Lightning Protection Codes[24, 25, 26] be included in the design by reference in the SAR documentation. Where designs of structures subject to approval are also covered by such other codes, the review should include evaluation of compliance with those codes.

(i) Concrete Containments

ACI 359[23], also designated as Section III, Division 2 of the ASME Boiler and Pressure Vessel Code[1], Subsection CC, is acceptable for prestressed and reinforced concrete that is an integral component of a radioactive material containment vessel that must, in operation or in testing, withstand internal pressure. Application of ACI 359[23] is based on the containment function, regardless of whether the concrete structure is fixed or portable and regardless of where the concrete structure is fabricated. ACI 359[23] also applies to structural concrete supports that are constructed as an integral part of the containment.

If ACI 359[23] is applicable to an ISFSI structure, it is applicable for the full design, material selection, fabrication, and construction of that structure. NRC has not accepted the substitution of elements of ACI 349[5] or ACI 318[22] for any portion of ACI 359[6] for an ISFSI structure. ISFSI structures for which ACI 359[23] is applicable shall also meet the minimum functional requirements of ANSI/ANS 57.9,[6] where specific requirements in the subject area are not included in ACI 359.[23]

(ii) RC Structures Important to Safety But Not within the Scope of ACI 359[23]

NRC accepts use of ACI 349[5] for design, material selection and specification, and construction of all RC structures that are not within the scope of ACI 359;[23] except that additional or more stringent requirements given in ANSI/ANS 57.9,[6] as incorporated by reference in NRC Reg Guide 3.60[2], must also be met.

The following identifies the portions of ACI 349[5] and ASTM standards that are applicable to design (including material selection) that must be met by those applicants that choose to use ACI 318[22] for construction. The paragraph references are as in ACI 349-90.[5] This includes metal embedments. Unlisted and excepted sections cover construction requirements, for which NRC accepts substitution of ACI 318.[22]

Chapter 1,	"General Requirements", Section 1.1 and 1.5 (less references to construction), Section 1.2, Section 1.4
Chapter 2,	"Definitions", All
Chapter 3,	"Materials, All, except Section 3.1, 3.2.3, 3.3.4, 3.5.3.2, 3.6.7, 3.7
Chapter 4,	"Concrete Quality", Section 4.1.4
Chapter 6,	"Form work, Embedded Pipes, and Construction Joints", Section 6.3.6(k), 6.3.8
Chapter 7,	"Details of Reinforcement", All
Chapter 8,	"Analysis and Design" - General Considerations, All
Chapter 9,	All (but see 2.2.d(1), below)"
Chapters 10-19,	All
Appendix A,	All
Appendix B,	"Steel Embedments," All, but note that the load combinations and load variation requirements of ANSI/ANS 57.9[6] must be met in addition to those of ACI 349[5] Section 9.2 cited at Section B.3.2
Appendix C,	"Special Provisions for Impulsive and Impactive Effects", All, except that the load combinations and load variation requirements of ANSI/ANS 57.9[6] must be met in addition to those of ACI 349[5] Section 9.2

ASTM standard specifications applicable to design and material specification (as referenced in ACI 349[5]-90) and acceptable to NRC for RC structures design and construction are:

A 36, A 53, A 82, A 184, A 185, A 242, A 416, A 421, A 496, A 497, A 500, A 501, A 572, A 588, A 615, A 706, A 722, C 33, C 144, C 150, C 595, and C 637 [Note that these do not include the following ASTM standard specifications that are listed in ACI 318:

A 616, A 617, A 767, A 775, and C 989. These standard specifications are applicable if ACI 349[5] is used for construction.]

(c) Other RC Structures Subject to Approval

NRC accepts use of either ACI 318[22] or ACI 349[5] for RC structures that are subject to the approval action but are not important to safety. If ACI 349[5] is used for design, NRC accepts use of ACI 318[22] for construction.

(d) Steel Attachments to RC Structures

Codes and standards applicable for steel attachments to RC structures are described in Subsection IV.3 for structures important to safety and in Subsection IV.4 for other structures subject to approval.

(2) Structural Design Features

Review the adequacy of the information provided in the SAR documentation on the physical design of RC structures. This should include the following as a minimum:

Dimensioning of all surfaces

Locations, sizes, configuration, spacing, welding, enclosure (e.g., spirals, stirrups), and depth of cover of reinforcement

Locations and specifications for control, contraction, and construction joints

Materials, with defining standards or specifications

Review information on the physical design of embedments and attachments. This should include the following as a minimum:

Locations, configuration, depth of embedment, interfaces; material; connections and connectors; and, protective or functional coatings

Dimensions, materials, and specifications for welds

c. Structural Materials

(1) RC Components

Review the identification and stated properties of the RC component materials for completeness, accuracy, and acceptability.

Materials and material properties used for design and construction of RC structures within the scope of ACI 359 must comply with the descriptions and requirements of ACI 359.

Materials and material properties used for the design and construction of RC structures important to safety but not within the scope of ACI 359 should comply with the requirements of ACI 349.

Materials and material properties used for the design and construction of RC structures that are not important to safety but are to be included in the approval should comply with the requirements of ACI 318 (or ACI 349 if that code is used for design of the structures).

(2) Embedments and Attachments

Review the identification and stated properties of the material to be used for embedments, inserts, conduits, pipes, or other items that are to be embedded in the concrete, for completeness and acceptability. Embedments must satisfy the requirements of the code used for design of the RC structure in which they are embedded (e.g., ACI 359, ACI 349, or ACI 318). Aluminum should not be used for any embedded objects that will be in contact with the wet concrete (because of the potential for concrete degradation from an adverse chemical reaction).

Review the identification and stated properties of the material to be attached to the RC structures for completeness and acceptability. The material must satisfy requirements appropriate to its importance to safety. Unless otherwise specified in this SRP, steel structural attachments must comply with the appropriate requirements of Reference 28.

d. Structural Analysis

(1) Load Conditions

Guidance for review of load conditions applicable to ISFSI structures in general is provided in Subsection IV.1.d, above. This subsection focuses on load conditions of special concern, and load combinations for, RC structures. Review the application of these to the RC structures for appropriateness, completeness, and correctness.

Load definitions and load combinations shown in Table 3-1 have been accepted by NRC for analysis of steel and reinforced concrete ISFSI structures important to safety. The load combinations are as included or derived from References 6 and 5.

Structures important to safety are to have sufficient capability for every section to withstand the worst case normal and off-normal-conditions without permanent deformation and with no degradation of capability to withstand any future loadings.

(a) Normal-conditions

SAR documentation is reviewed for adequate inclusion of the following conditions that may be of particular concern for RC structures:

Live and dynamic loads associated with transfer of the confinement cask to and from its storage position.

Live and dynamic loads associated with installing closures.

Load or support conditions associated with potential differential settlement of foundations over the life of the cask system.

Thermal gradients associated with normal range operations and ranges of ambient temperature.

Thermal gradients that may result from impingement of rain on highly heated concrete.

(b) Off-Normal-conditions

The SAR should be reviewed for adequate inclusion of the following off-normal operations and events that may be of particular concern for RC structures:

Live and dynamic loads associated with equipment or instrument malfunctions, negligence, or accidental misuse during transfer of the confinement cask to and from its storage position.

Situations in which a confinement cask is jammed or is moved at an excessive speed into contact with a RC structure.

The impact of RC structures by a suspended transfer, confinement, or storage casks.

Off-normal ambient temperature conditions, although they may be less severe than the accident-level conditions. These may be of concern because of different sets of factors in the off-normal and accident-level load combinations; and, because concrete temperature limits for off-normal-conditions are the same as for normal-conditions. Greatly elevated concrete temperatures are allowed for accident-level conditions (ACI 349,[5] Section A.4).

(c) Accident-Level Events and Conditions

The SAR should be reviewed for adequate inclusion of the following conditions associated with accident-level events and conditions that may be of special concern for RC structures:

Loads associated with accidental drops or other impacts resulting from gross negligence during transfer of the confinement cask to and from its storage position.

Events that produce extreme thermal gradients in the concrete.

Contact caused by earthquake between the confinement cask and the RC structures

Drop of a separate closure into position or onto the structure.

The ACI codes [5,22,23] are intended to ensure ductile response beyond initial yield of structural components. ACI 349 imposes additional conditions on design (over those of ACI 318[22]) that effectively increase the ductility. Design areas that should be reviewed to ensure that the RC structural design provides code levels of ductility are principally as shown below. The steel reinforcement schedules and drawings should be checked to ensure that any reinforcing steel quantities, sizes, and locations are as determined by the design analysis. Use of more shear and enclosing reinforcement (e.g., stirrups, ties, and spirals) than required does not reduce ductility for the member. Constraints on use of excess steel to ensure that ductility is not reduced do not apply to the shear and enclosure reinforcement. Things to consider include:

Upper limit (60,000 psi, 4219 kgf/cm^2) on the specified yield strength of reinforcement and lower limit (3000 psi, 211 kgf/cm^2) on concrete specified compressive strength (f'c).

Limit on the amount (cross-section area) of compressive reinforcement in flexural members.

Requirements on continuation and development lengths of tensile reinforcement.

Specifications for confinement and lateral reinforcement in compression members, other compressive steel, and at connections of framing members.

The design ensures that flexure controls (and limits) the response.

Requirements for shear reinforcement.

Limitations on the amount of tensile steel in the flexural members relative to that which would produce a balanced strain condition.

Projected maximum responses to design basis loads within the permissible ductility ratios for the controlling structural action.

Embedments designed to fail in the steel before pullout from the concrete.

The construction specifications or descriptions (to the extent included in the SAR documentation) should be reviewed to ensure that substitution of materials, use of larger sizes, or placement of larger quantities of steel will be precluded; and that provisions for splicing or development of reinforcing steel do not reduce ductility of the members.

(2) Structural Analysis Methods

Review the analytical documentation for structural analysis methods used for design and verification of the RC structures.

The structural analysis of structures within the scope of ACI 359[23] shall be in accordance with ACI 359.[23]

NRC accepts strength design as presented in the current ACI 349[5] for RC structures important to safety (that are not within the scope of ACI 359[23]). If another design approach is used, the review should include in-depth comparison of that approach with the provisions of ACI 349.[5] This effort would be part of the SAR evaluation.

NRC accepts use of procedures and approaches described in the regulations referenced in Regulatory Guide 3.53[2] and that are applicable to ISFSI. NRC accepts use of guidance for analysis of natural phenomena in NUREG 0800,[9] however the load combinations shown in Table 3-1 and the design and construction requirements of the codes cited above govern. NRC has accepted ASCE 7[7] for estimation of wind, snow, and rain loads, and for conversion of tornado wind speed to pressure. NRC has also accepted seismic analysis in accordance with ASCE 4[27] and ASCE 7[7]. NRC has accepted tornado missile impact analysis in accordance with Reference 11.

(a) Strength Design

Strength (or "Ultimate Strength") design is the usual approach used in American RC design. Strength design is the only design approach that has been accepted for ISFSI RC structures not within the scope of ACI 359[23]. Strength design is the approach used in the current ACI 318[22] and ACI 349[5] codes. These current design codes are based on extensive empirical experience with concrete construction. The current strength design approach as presented in these codes includes empirically derived requirements and constraints. Determination that a RC structure designed by another approach satisfies ACI 349[5] typically requires clause-by-clause review of the code for compliance.

(b) Allowable Stress Design

Allowable stress design was formerly used as the basis for ACI codes for RC design. Those codes do not reflect additional experience gained through observations of structural performance and experimental testing that is included in the current approach to strength design. A clause-by-clause comparison of the structural design for compliance with ACI 349[5] should be performed for RC structures not designed using ACI 349.[5]

(c) Analytical Codes and Models

NRC has accepted the use of different analytical codes and models for structural analysis of RC structures. Uses have included development of stresses resulting from seismic events and thermal gradients. NRC does not require use of computer models and codes for analysis of the responses or stresses of simple ISFSI concrete structures. NRC has not required the codes that are used to have been developed under rigid nuclear safety quality controls (e.g., ASME NQA-2). However, the codes must be appropriately validated for their intended use.

Analytical codes capable of dynamic analysis have been used and their results accepted by NRC for multi-story and complex RC structures. The reviewer must determine if the use or absence of use of codes for specific analyses is acceptable. Bases for acceptance may be the simplicity of the structure, extent or details provided in other calculations, or demonstration that the structural demands, in the area of analysis, are sufficiently low relative to the estimated capacity of the structure. Acceptance, based on one or more of these conditions, should be such that further refinement of the computations would have negligible change in the conclusion.

(3) Structural Evaluation

(a) Summary Structural Capability

Review the selection or identification of the critical sections of the RC structures used for demonstrating that the structures have acceptable Regulatory Guides of safety under the different load combinations. "Critical sections" are those sections that have the lowest margins of safety under the various loading conditions and types of stress. These sections may be selected based on inspection, sensitivity analysis, and/or finite element analysis. The following provides guidance for evaluating the identification of critical sections:

Loads and stress demands for structures should be within the scope of ACI 359[23] shall be as defined and described in ACI 359.[23]

Unless the lowest margins of safety are determined by finite element analysis using the applicable load combinations, critical sections should be identified for each structurally distinct element of the RC structure. An integrally cast structure may have multiple structurally distinct elements. (e.g., the different sides, base, and roof of a vault; and the base, corners, side walls, lips, and any structural discontinuities (e.g., at a trunnion) of an RC cylinder.

The level of refinement needed in identifying critical sections depends primarily on the margins of safety and secondarily on the importance to safety.

Many RC structures are designed primarily to provide radiation shielding. Such structures may have significantly excess capacity for structural loadings because of the use of section thicknesses selected for shielding and satisfying code requirements for minimum reinforcing steel. Structures important to safety may have such high margins of safety that only elementary structural computations are necessary to acceptably demonstrate compliance with all of the applicable load combinations. For simple elementary analysis, the margin of safety, for a particular section, should consider the highest axial, bending, and shear stresses occurring concurrently.

Intensive analysis to prove that the truly critical sections are used is expected when margins of safety are close to the minimum acceptable values.

The critical sections for bending, shear, axial stress, and combined stresses are typically different for a single structural element. They may also differ for different load combinations.

The lowest margins of safety for structural elements may be for different types of stresses under different load combinations.

Design and evaluation for accident-level loads involve structural loadings and responses that are not typically addressed in non-nuclear construction. The structural shapes are not typical. Selection of representative sections for analysis by observation and experience may not be adequate without further computations to demonstrate that no other sections would have lower margins of safety. This could involve, for example, analyses of immediately adjacent sections to prove that margins of safety for the stress type rise in both directions from the section.

Table 3-1 identifies and describes loads used in load combinations for RC structures not within the scope of ACI 359.[23] The symbols and terminology used in the SAR should correlate with these loads. However, if the symbols and terminology used in the SAR are different but acceptable, they may be used in the SER in place of those in Table 3-1, for consistency between the SER and SAR.

NRC does not require analysis of load combinations for situations in which nuclear material is not present. However, RC structures should not be exposed to credible damage that may not be evident or discovered before completion of construction or use. This could reduce the structural capacity or functional capability of the structure below that needed to meet demand when nuclear material is present. (e.g., hidden damage could occur in handling and shipping precast RC structures).

RC structures subject to review but not "important to safety" should satisfy the load combinations of ACI 318,[22] as a minimum.

(b) Fabrication and Construction

(i) Code Construction Criteria

Structures that are within the scope of ACI 359[23] must be fabricated and constructed in compliance with ACI 359.[23]

NRC accepts construction of RC structures that are not within the scope of ACI 359[23] in accordance with ACI 349[5] or ACI 318.[22] Selection and validation of concrete mix to meet design requirements is considered to be a construction function. Specification of cement type, aggregates, and special requirements for durability and elevated temperatures is considered to be a design or material selection function and therefore to be governed by ACI 349 (ACI 359 if applicable).

The following identifies sections of ACI 318,[22] Building Code Requirements for Reinforced Concrete (chapters, appendix, and paragraphing per ACI 318-89) that have been accepted by NRC for construction of ISFSI RC structures that are not within the scope of ACI 359.[23]

Chapter 1, "General Requirements", Section 1.1.1, 1.1.2, 1.1.3, and 1.1.5 (less references to design and material properties); Section 1.3
Chapter 2, "Definitions", use ACI 349 Chapter 2
Chapter 3, "Materials", Section 3.1, Section 3.8 (except delete A 616 and A 617)
Chapter 4, "Durability Requirements", All
Chapter 5, "Concrete Quality, Mixing, and Placing", All.
Chapter 6, "Form work, Embedded Pipes, and Construction Joints", All (less references to design and material properties, these are governed by ACI 349)[a]

ASTM Standard Specifications Acceptable for Construction and Associated Testing

C 31, C 39, C 42, C 94, C 109, C 172, C 192, C 260, C 494, C 496, C 685, and C 1017[b]

The following standards relating to construction are identified in ACI 349 and may be used: C 88, C 131, C 289, and C 441.

NRC accepts construction of RC structures not important to safety in accordance within ACI 318.

(ii) Evaluation of Construction Commitments

Review the SAR documentation for inclusion of acceptable specification for the planned construction and fabrication. These should be evaluated against the requirements related to construction in ACI 349[5] or ACI 318[22] for RC structures not within the scope of ACI 359.[23] For structures that are within the scope of ACI 359,[23] there must be a commitment

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[a] Use ACI 349 for the remainder

[b] All are in both ACI 318 and ACI 349 except that C 1017 is only cited in ACI 318

to fabricate and construct in accordance with ACI 359.[23]

Construction requirements should prohibit use of aluminum in any forms, chutes, ties, or other objects used in construction that will be in contact with the wet concrete. This is because of the potential for concrete degradation from an adverse chemical reaction with aluminum, including the fine particles that may be collected by wet concrete in pipes and chutes.

Construction specifications or drawing notes should preclude use of more, larger cross-section, or higher-yield reinforcing steel than that derived by the design analysis. There is no constraint on use of higher strength concrete than that assumed by the design analysis if other properties required of the concrete mix are provided.

(iii) Structural Compatibility with Functional Performance Requirements

Review the SAR documentation for an analysis of structural responses to DBE that demonstrates that the structures, systems, or components important to safety can continue to perform their intended safety functions. The design codes and load combinations used for RC structures important to safety can permit permanent deformation or other damage under design basis event loading. If structural damage could occur, it is essential that continuing capability for needed functional performance be demonstrated. For RC structures this typically involves shielding the confinement cask from external events, maintaining cooling ventilation, and allowing for ready retrieval of the confinement cask. Demonstrating continuing capability involves recognizing the nature and extent of credible damage that may occur and understanding potential interactions between the damaged RC structures and other cask system structures important to safety.

Under conditions acceptable to NRC, RC structures are not required to survive an accident-level event with the same capability for a full design life and the same ability to withstand further accident-level events. RC structure degradation should be readily apparent in the course of routine inspections and surveillances, or would be discovered by the inspections and/or tests that may be proposed as responses to accident-level events. For example, tornado missile impact may degrade the radiation shielding by cratering the exterior of the RC structure. This could be simply and adequately repaired. The impact could also cause spalling at an inner, hidden surface, which could affect shielding and cooling air flow. The design should preclude the interior spalling unless practical means of detecting the situation and remedying it are proposed.

NRC has accepted returning the stored fuel to the fuel pool and not making further use of the damaged component, as remedial action for an accident-level event. NRC may also accept inspection and repair of damage to the structure, depending on the design and proposed actions.

3. Other Important to Safety System Components

a. Scope

The review should include the structural evaluation of all structures that are important to safety (as defined in 10 CFR Part 72) that are not addressed as components of the confinement cask (Subsection IV.1, above) and are not reinforced concrete (see Subsection IV.2, above). The review includes drawings, plans, sections, and technical specifications for these structures, systems, and components.

General guidance for structural review of cask system components is provided in Subsection IV.1.d (1), and (2), above. The guidance in this subsection supplements that guidance.

b. Structural Design Criteria and Design Features

(1) Design Criteria

(a) General Structural Requirements

Structural requirements are driven by the functional roles of the system components and maintaining safety. Safety requirements are expressed in referenced rules, standards, and codes and as criteria specific for the component. The basic safety requirements are that the structural and functional design preclude the following:

Unacceptable risk of criticality.
Unacceptable release of radioactive materials to the environment.
Unacceptable radiation dose to the public or workers.
Significant impairment of ready retrievability of stored nuclear materials.

(b) Applicable Codes and Standards

NRC accepts use of ANSI/ANS 57.9[6] and the codes and standards cited therein as the basic references for ISFSI structures important to safety that are not designed in accordance with the ASME B&PV Code Section III[6]. The principal included references applicable to steel structures and components are the following:

- AISC, "Specification for Structural Steel Buildings - Allowable Stress Design and Plastic Design"[28]
- AISC, "Code of Standard Practice for Steel Buildings and Bridges"[29]
- AWS D1.1, "Structural Welding Code-Steel"[30]
- ASCE 7, "Minimum Design Loads for Buildings and Other Structures",[7] however note that the load combinations of ANSI/ANS 57.9[6] are to be used

These documents cite further sources of criteria, which are considered to have the effect of being cited or quoted in the basic structural criteria. In addition, NRC accepts design of lifting equipment and components of vessels and other equipment provided for lifting, important to safety in compliance with ANSI N14.6.[17]

NRC has not required that steel ISFSI structures important to safety be designed or built to comply with ANSI/ANS N690, "Nuclear Facilities - Steel Safety-Related Structures for Design Fabrication and Erection".[31]

(2) Structural Design Features

Review the design description in the SAR documentation for inclusion of the functional performance required of the structures. The design description should provide for the corresponding capability.

Auxiliary cask system equipment important to safety has often been of special design. The structural design features that provide for safety should be supported by design or operational analysis that establishes that the basic safety criteria will be met, regardless of problems that may occur in mechanical, electrical, human operator, or other operations.

NRC has accepted and approved cask system designs that depended on the operation of new mechanical systems for system use. NRC approval does not certify that the mechanical systems will operate as projected, but rather, that the proper functioning is necessary to successfully complete a specified operation. NRC approval is based on an evaluation that regardless of the systems success, or lack of success in mechanical operation, the basic safety criteria will be met. The basic safety criteria are as stated in (a), above.

Review the proposed system design against planned operations, and potential off-normal and accident-level events and conditions. Determine if the structural design of the equipment provides for continuing satisfaction of the basic safety criteria. The review should consider that the equipment could fail to operate at any time i.e., during operations at the physical limits of speed or range, or during an event of credible, off-normal, and accident-level.

c. Structural Materials

Structural materials for components important to safety not addressed in Subsections IV.1 and IV.2, above, should be fully defined in the SAR documentation. Properties that relate to structural performance, and resistance or response to thermal, radiation, or other applicable environments should be included in the SAR. Confirm that these are based on acceptable sources. Resistance to corrosion in the prospective environments should be included, with need for protective coatings and/or corrosion allowances. Controls on material quality should be cited or should be included in codes that are incorporated in the design, fabrication, and construction criteria by reference.

d. Structural Analysis

Guidance on structural analysis is at Subsection IV.1.d(1) and (2) for cask system structures in general. This subsection provides supplemental guidance relating primarily to steel structures other than the confinement cask and its contents and integral components.

(1) Load Conditions

Load definitions and load combinations shown in Table 3-1 have been accepted by NRC for analysis of steel and reinforced concrete ISFSI structures important to safety. The load combinations are as included or derived from References 6 and 5.

Structures important to safety are to have sufficient capability for every section to withstand the worst-case loads under normal and off-normal-conditions such that no permanent deformation and no degradation of capability to withstand any future loadings occur.

The load combinations of ANSI/ANS 57.9[6] are to be used (tornado loads should be treated by substitution for Accident Loads (A) in the load combination expressions). Design by the allowable stress or plastic design approach is acceptable.

(2) Structural Analysis Methods

Analytical methods should be appropriate for the type of materials and type of construction. Analytical methods, codes, and models may have to be adapted for highly specialized cask system equipment designs. These require special review attention. The review should ensure that the approach is fully documented, supported, and acceptable. The potential for risk to safety, from design error, for special cask system equipment, should be considered in determining the extent of proof of suitability of the approach acceptable to NRC.

(3) Structural Evaluation

The basic guidance for evaluation of the variety of cask system equipment and structures that may be important to safety is to ensure compliance with the basic safety criteria Subsection IV.3.b(1)(a), above.

4. Other Components Subject to Approval

a. Scope

Structures that are not important to safety, included in the description of the cask system, are reviewed for proper functioning to the extent that they are required elements of the total cask system. The review includes evaluating all structures that are proposed for approval as a design acceptable to NRC. The review should ensure that the information provided is sufficient to confirm the proper functioning of the structure and the overall system. The review should also address the potential response of those system elements, that are not important to safety, to accident-level conditions and natural phenomena events, to ensure that they do not jeopardize the safety provided by other elements of the system. NRC approval does not include review of the structural design of such structures.

b. Structural Design Criteria and Design Features

(1) Design Criteria

(a) General Structural Requirements

NRC review of structures subject to approval but not important to safety is based on determining if the structures can perform their function properly, and to ensure that the response of the structures to credible off-normal and accident-level events and conditions does not create secondary hazards for cask system components or the stored nuclear materials.

(b) Applicable Codes and Standards

The principal requirement documents for structures, which are not important to nuclear safety, but which are to be included in the review of principal requirements documents include:

- ASCE 7[7]
- Uniform Building Code[32]
- AISC, "Specification for Structural Steel Buildings, Allowable Stress Design and Plastic Design"[28]
- AISC "Code of Standard Practice"[29]
- ASME B&PV Code, Section VIII [33]
- [DOE facilities only] DOE Order 6430.1A or its successor document(s)

(2) Structural Design Features

Review the adequacy of descriptions of cask system components not important to safety but subject to NRC approval. The descriptions should adequately describe the intended functions. Greater structural information and assurance of acceptable fabrication and construction is needed if there is a credible possibility that structural response or failure of the components may cause a secondary risk to other components that are important to safety or to the subject nuclear material. For example, the components may impact components important to safety in response to tornado or seismic events.

c. Structural Materials

Review the identification of structural materials to the extent appropriate to determine if they are adequate for their intended function. The level of review and extent of information provided should be based on the possibility and consequences of secondary effects on components that are important to safety (see b(2), above). Materials should be as permitted or specified in the applicable code(s) (see b(1), above).

d. Structural Analysis

(1) Load Conditions

Load definitions and load combinations shown in Table 3-1 have been accepted by NRC for analysis of steel and reinforced concrete ISFSI structures important to safety. These may also be used for structures not important to safety.

NRC has accepted use of load combinations given in the Uniform Building Code[32] for ISFSI structures not important to safety.

NRC has accepted use of load combinations given in References 5 , 6 and 7 for ISFSI structures not important to safety.

NRC has accepted load descriptions, combinations, and analytical approach as given in the ASME B&PV Code, Section VIII, for pressure systems, vessels, and casks that do not form elements of the confinement cask.

(2) Structural Analysis Methods

Review structural analysis methods, codes, and models used, and their use. These should be in accordance with the design code applicable to the component (as discussed at b(1)(b), above).

(3) Structural Evaluation

The reviewer may determine that NRC structural evaluation is not appropriate. This may be because of insufficient information on which to base an evaluation. This may also result from determination that NRC approval should not encompass specific components (that are not important to safety), even though inclusion in the approval is sought in the application. The SER should include the rationale for exclusion.

V. REVIEW PROCEDURES

Structural evaluation guidance is provided in Subsection IV of this SRP Section. The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. Description of Structures, Systems, and Components Important to Safety

Verify that the functional characteristics of the structures, systems, and components important to safety are clearly identified. The information furnished in the SAR is reviewed for sufficiency in accordance with the requirements of 10 CFR Part 72.24 & 72.26, Contents of Application:

Technical Information, as related to information pertinent to structural evaluation. Any additional required information is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications is checked against those listed in Subsection 2 of this SRP section. The reviewer assures himself that the codes, standards, and specification are acceptable, and the applicable editions are being used and referenced.

3. Loads and Loading Combinations

Verify that the loads and load combinations are as those specified in Subsection 2 of this SRP Section. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted as such to the applicant for further justification.

4. Design and Analysis Procedures

Determine that the design and analysis procedures and assumptions are based on accepted engineering practice and are conservative. The behavior of the structure, under various loads, and the manner in which these loads are treated, in conjunction with other coexistent loads, are reviewed to establish compliance with guidance described in Subsection 2 of this SRP Section.

5. Structural Acceptance Criteria

The limitations on allowable stresses and strains, in the confinement cask, reinforced concrete components, system components important to safety, and other components subject to review, are compared with those specified in applicable codes and standards. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, evaluate the justification provided to show that the functional integrity of the structure will not be affected. If the justification is not acceptable, a request for the required additional justification and bases are made.

6. Materials, Quality Control, and Special Fabrication Techniques

The information provided on materials, quality control programs, and special fabrication techniques, if any, is reviewed and compared with that specified in Subsection 2 of this SRP Section. If a new material not used in prior approvals is used, the applicant is requested to provide sufficient test and user data to establish the material acceptability. Similarly, any new quality control programs or construction techniques are reviewed and evaluated to ensure that there will be no degradation of structural quality that might affect the structural integrity and function.

7. Testing, and In-service Surveillance Requirements

Pressure test procedures of the confinement cask are reviewed and compared with procedures described in ASME Code, Section III, Subsection NB-6000.[1] Acceptance test and maintenance requirements for trunnions are reviewed and compared to those described in the ASME Code and ANSI N14.6,[17] as applicable. Any other proposed testing and in-service surveillance programs are reviewed on a case-by-case basis. Review SAR Chapter 9 to verify that all appropriate acceptance tests have been included and that evaluations are addressed in Section 9 of the SER.

VI. EVALUATION FINDINGS

The structural evaluation must provide reasonable assurance that the cask system will enable safe storage of spent fuel. This finding is based on a review that considered the regulation, appropriate Regulatory Guides, applicable codes and standards, and accepted engineering practices. Acceptance of the structural design of a storage cask system is based on meeting the relevant requirements of the following regulations:

Structures, systems and components important to safety are described with drawings and text in sufficient detail to enable an evaluation of their structural effectiveness.

The applicant has met the requirements of 10 CFR Part 72.24, "Contents of Application": Technical Information, as relates to information pertinent to Structural Evaluation.

The applicant has also met the requirements of 10 CFR Part 72.26, "Contents of Application" and 10 CFR Part 72.44(c), "License conditions", as they relate to the inclusion of technical specifications pertaining to the structures of the proposed cask system.

The applicant has met the requirements of 10 CFR Part 72.122(b) and (c) and 10 CFR Part 72.24(c)(3). The structures, systems, and components important to safety are designed to accommodate the combined loads of normal, accident, and natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask for various design loads are determined by analysis. Total stresses for the combined loads of normal, accident, and natural phenomena events are acceptable and if they are found to be within limits of applicable codes, standards, and specifications.

The applicant has met the requirements of 10 CFR Part 72.124(a), "Criteria for Nuclear Criticality Safety", and 10 CFR Part 72.236 (b), "Specific requirements for spent fuel storage cask approval". The design and fabrication of handling, packaging, transfer, and storage systems include margins of safety for the nuclear criticality parameters. The applicant has demonstrated adequate safety for the handling, packaging, transfer, and storage under normal, off-normal, and credible accident conditions.

The applicant has met the requirements of 10 CFR Part 72.236(l), "Specific requirements for spent fuel storage cask approval". The design analysis and submitted bases for evaluation acceptably demonstrate that the cask and other systems important to safety will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

The applicant has met the requirements of 10 CFR Part 72.120, "General considerations", and 10 CFR Part 72.122, "Overall requirements", as they relate to the structural design, the application including:

- provisions for design, fabrication, erection, and testing to acceptable quality standards;

- provisions of adequate structural protection against environmental conditions and natural phenomena, fires, and explosions;

- provisions for appropriate inspection, maintenance, and testing;

- provisions for adequate accessibility in emergencies;

- provisions for a confinement barrier that acceptably protects the spent fuel cladding during storage;

- provisions for structures that are compatible with appropriate monitoring systems; and,

- provisions for structural designs that are compatible with ready the retrievability of spent fuel.

The applicant has met the requirements of 10 CFR Part 72.236(e), (f), (g), (h), (i), (j),(k), and (m), specific requirements, as they apply to the structural design, for spent fuel storage cask approval. The cask system structural design acceptably provides for:

- redundant sealing of confinement systems;
- adequate heat removal without active cooling systems;
- storage of the spent fuel for a minimum of 20 years;
- compatibility with wet or dry spent fuel loading and unloading facilities;
- acceptable ease of decontamination;
- conspicuous and durable marking;
- compatibility with removal of the stored fuel from the site, transportation, and ultimate disposition by the Department of Energy; and
- inspections for defects that might reduce confinement effectiveness.

TABLE 3-1 LOADS AND LOAD COMBINATIONS

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description

Designations and Descriptions of Loads

Notes: This table is not applicable to the analysis of confinement casks and other components designed in accordance with the ASME B&PV Code, Section III [Ref.1]

Capacity and loads may be terms of loads, or forces, moments, or stresses caused by such loads; and the usage must be consistent among the terms used in the load combination. Force, versus mass, units are to be used for loads

Symbol	Capacity or Load Term	Source (reference)	Capacity or Load (or Demand) Description
U	Strength	ANSI 57.9 [Ref, 6], ACI 349 [Ref. 5]	Minimum available strength (capacity) to meet the load combination, calculated in accordance with the requirements and assumptions of ACI 349, and after application of the strength reduction factor, Ø, as defined and prescribed at Section 9.2, "Design Strength," of ACI 349. If strength may be reduced during the design life by differential settlement, creep, or shrinkage, those effects shall be incorporated in the dead load instead of by reduction of available strength.
U_f	Strength of foundation	ANSI 57.9	Minimum available strength of foundations to meet the applicable load combinations.
U_s	Soil bearing capacity	ANSI 57.9	Allowable soil bearing (or pile) capacity, to be determined by foundation analysis or to be expressed as a minimum in a SAR
O/S	Overturning/ Sliding Resistance	ANSI 57.9	Required minimum available resistance capacity against each of overturning or sliding of the structure.
D	Dead load	ANSI 57.9	Dead load of the structure and attachments including permanently installed equipment and piping. The weight and static pressure of stored fluids may be included as dead loads when these are accurately known or enveloped. Loads caused by differential settlement, creep, and/or shrinkage, if they produce the most adverse loading conditions, are included in dead load. If differential settlement, creep, or shrinkage would reduce the combined loads it shall be neglected.

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description
L	Live loads	ANSI 57.9 ACI 349	Live loads, including equipment (such as a loaded storage cask) and piping not permanently installed, and all loads other than dead loads that might be experienced that are not separately identified and used in the load combination, and that are applicable to the situation addressed by the load combination. Typically includes the gravity and operational loads caused by handling equipment, and routine snow, rain, ice, and wind loads and normal and off-normal impacts of equipment. Loads caused by piping and equipment reactions are included. Depending on the case being analyzed, includes normal or off-normal events not separately identified, as may be caused by equipment or instrument malfunction, negligence, and other man-made or natural causes.
DB	"Design Basis" (Accident-level) loads	10 CFR Part 72 [Ref. 34]	Design basis loads are controlling parameters for external events that include: (1) Estimates of extreme credible natural events to be used for deriving design bases based on consideration of historical data or the rated parameters, physical data, or analysis of upper limits of the physical processes involved and (2) estimates of extreme credible external man-induced events to be used for deriving design bases that will be based on analysis of human activity in the region taking into account the site characteristics and the risks associated with the event. Design basis loads include credible accidents and extreme natural phenomena. Presumption of concurrent independent accidents or severe natural phenomena producing compounding design basis loads is not required.
T	Thermal loads	ANSI 57.9 NUREG 0800, Section 3.8.3) [Ref. 9]	Thermal loads, including loads caused by "normal" condition temperatures, temperature distributions, and thermal gradients within the structure, expansions and contractions of components, and restraints to expansions and contractions, but excepting thermal loads that are separately identified and used in the load combination. Thermal loads shall presume that all loaded fuel has the maximum thermal output allowed at time of initial loading in the ISFSI. Thermal loads shall be determined for the most severe of both steady state and transient conditions. For multiple cask storage facilities, thermal loads shall be determined for the worst case loadings on potentially critical sections (e.g., all in place, only one cask in place, alternate casks in place).

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description
T_a	Accident-level thermal loads	ACI 349	Thermal loads produced directly or as a result of off-normal or design basis accidents, fires, or natural phenomena. [Note: Although off-normal and design basis thermal loads are treated the same in the load combinations, there is a distinction between off-normal and design basis temperature limits for concrete. Off-normal temperature limits are as for "normal" conditions.] For multiple cask storage facilities, thermal loads shall be determined for the worst case loadings on potentially critical sections.
A	Accident loads	ANSI 57.9	Loads caused by the direct and secondary effects of an off-normal or design basis accident, as could result from an explosion, crash, drop, impact, collapse, gross negligence, or other man-caused occurrences.
H	Lateral soil pressure	ANSI 57.9	Loads caused by lateral soil pressure as would exist in normal, off-normal, or design basis conditions, as corresponds to the load combination in which used. Derived using unfactored loads.
W	Wind loads	ACI 349	Wind loads produced by normal and off-normal maximum winds. Pressure caused by wind and with consideration of wind velocity, structure configuration, location, height above ground, gusting, importance to safety, and elevation may be calculated as provided by ASCE 7.
W_t	Tornado loads	ACI 349	Loads caused by wind pressure and wind generated missiles caused by the design basis tornado or design basis wind (for sites where design basis wind rather than tornado produces the most severe pressure and missile loads). Pressure caused by wind velocity and elevation may be calculated as provided for these factors in ASCE 7. Tornado wind velocity or pressure does not have to be increased for structure importance, gusting, location, height above ground, or importance to safety (these do apply for design basis wind).
E	Earthquake loads	ANSI 57.9 ACI 349	Loads caused by the direct and secondary effects of the design basis earthquake (DBE). The DBE is comparable to the "Safe Shutdown Earthquake" used for analysis of nuclear facilities under 10 CFR Part 50. Secondary loads caused by earthquake-caused secondary events (e.g., fire, explosion, collapse of adjacent structure) are used in the load combinations as E but are not required to be considered as simultaneous with the earthquake loads (but the structure should be presumed to be in the condition that may follow a design basis earthquake).

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description
F	Flood load	ANSI 57.9	Loads caused by the direct and secondary effects of the design basis or off-normal flood, including flooding caused by severe and extreme natural phenomena (e.g., seiches, tsunamis, storm surges), dam failure, fire suppression, and other accidents.

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description

Load Combinations for Reinforced Concrete Structures

These load combinations apply to RC structures important to safety that are not within the scope of ACI 359 (ASME B&PV Code, Section III, Division 2)

Load Combination	Derivation (Reference)	Acceptance Criteria
D (in the load combinations)	ANSI 57.9 [Ref. 6] ACI 349 [Ref. 5]	Dead load, D, as used in the load combinations is to be varied by +5% if that produces the most adverse loading condition. If the dead load or its component reduces the effects of the combination of the other loads, and the dead load or its component would always be present, 90% of the dead load shall be used to simulate the most adverse loading condition.
L (in the load combinations)	ANSI 57.9	Live load, L, as used in the load combinations is to be varied from 0 to 100% to simulate the most adverse loading conditions. Live load caused by casks with stored fuel need only be varied by credible increments of loading of an individual cask. Live loads caused by multiple casks should be varied for the presence and positioning of one or more cask, as necessary to simulate the most adverse loading condition.
L (for precast structures prior to final integration in-place)	ACI 349	Live loads for precast structures shall consider all loading and restraint conditions from initial fabrication to completion of the structure, including form removal, storage, transportation, and erection. NRC is only concerned with analysis of loading of RC structures prior to use for ISFSI functions to the extent that the structures should not risk damage that may not be evident, thereby jeopardizing the capacity of the structures when in use. If the damage would be visibly evident prior to installation, analysis of the pre-completion event is not required.
All loads other than D (in the load combinations)	ANSI 57.9 ACI 349	If any load reduces the effects of the combination of the other loads and that load would always be present in the condition of the specific load combination, the net coefficient for that load shall be taken as 0.90. If the load may not always be present the coefficient for that load shall be taken as zero.
$U > 1.4\ D + 1.7\ L$	ANSI 57.9	Capacity/demand >1.00 for all sections
$U > 1.4\ D + 1.7\ (L + H)$	ANSI 57.9	Capacity/demand >1.00 for all sections

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description
$U > 1.05\ D + 1.275\ (L + H + T)$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > 1.05\ D + 1.275\ (L + H + T + W)$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > D + L + H + T + E$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > D + L + H + T + A$		ANSI 57.9	Capacity/demand >1.00 for all sections. An overturning accident for a cask in transfer or in separate storage on a pad is to be assumed unless more severe overturning also occurs as a result of a natural phenomenon.
$U > D + L + H + T_a$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > D + L + H + T + W_t$		ANSI 57.9 ACI 349	The load combination (capacity/demand >1.00 for all sections) shall be satisfied without missile loadings. Missile loadings are additive (concurrent) to the loads caused by the wind pressure and other loads, however local damage may be permitted at the area of impact if there will be no loss of intended function of any structure important to safety.
$U > D + L + H + T + F$		ACI 349	Capacity/demand >1.00 for all sections
$U_s > 1.4\ D + 1.7\ L$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + W$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + E$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + A$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + W_t$		ANSI 57.9 ACI 349	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions. Computation for tornado missile loadings not required.
$U_s > D + L + F$		ACI 349	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_f > D + L$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description
	$U_f > D + L + 1.7\,H$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$U_f > D + L + 1.275\,H + T$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$U_f > D + L + 1.275\,H + T + W$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$U_f > D + L + H + T + E$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$U_f > D + L + H + T + A$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$U_f > D + L + H + T_a$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$U_f > D + L + H + T + W_t$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$U_f > D + L + H + T + F$	ANSI 57.9	Capacity/demand >1.00 for foundation sections.
	$O/S \geq 1.5\,(D + H)$	ANSI 57.9	Capacity/demand ≥1.00 for structure, to be satisfied for both overturning and sliding
	$O/S \geq 1.1\,(D + H + E)$	ANSI 57.9	Capacity/demand ≥1.00 for structure, to be satisfied for both overturning and sliding
	$O/S \geq 1.1\,(D + H + W_t)$	ANSI 57.9	Capacity/demand ≥1.00 for structure, to be satisfied for both overturning and sliding

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description

Load Combinations for Steel Structures

These load combinations apply to steel structures important to safety that are not within the scope of the ASME B&PV Code, Section III [Ref. 1]

[To be inserted----Work in progress]

S (in the load combinations)	ANSI 57.9 [Ref. 6]	Strength of a section, member, or connection computed in accordance with the "allowable stress method" of the AISC Specification for Structural Steel Buildings [Ref. 29]
D (in the load combinations)	ANSI 57.9	Dead load, D, as used in the load combinations is to be varied by +5% if that produces the most adverse loading condition.
L (in the load combinations)	ANSI 57.9	Live load, L, as used in the load combinations is to be varied from 0 to 100% to simulate the most adverse loading conditions. Live load caused by casks with stored fuel need only be varied by credible increments of loading of an individual cask. Live loads caused by multiple casks should be varied for the presence and positioning of one or more cask, as necessary to simulate the most adverse loading condition.
L (for precast structures prior to final integration in-place)	ACI 349	Live loads for precast structures shall consider all loading and restraint conditions from initial fabrication to completion of the structure, including form removal, storage, transportation, and erection. NRC is only concerned with analysis of loading of RC structures prior to use for ISFSI functions to the extent that the structures should not risk damage that may not be evident, thereby jeopardizing the capacity of the structures when in use. If the damage would be visibly evident prior to installation, analysis of the pre-completion event is not required.
All loads other than D (in the load combinations)	ANSI 57.9 ACI 349	If any load reduces the effects of the combination of the other loads and that load would always be present in the condition of the specific load combination, the net coefficient for that load shall be taken as 0.90. If the load may not always be present the coefficient for that load shall be taken as zero.
U > 1.4 D + 1.7 L	ANSI 57.9	Capacity/demand >1.00 for all sections
U > 1.4 D + 1.7 (L + H)	ANSI 57.9	Capacity/demand >1.00 for all sections

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description
$U > 1.05\ D + 1.275\ (L + H + T)$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > 1.05\ D + 1.275\ (L + H + T + W)$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > D + L + H + T + E$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > D + L + H + T + A$		ANSI 57.9	Capacity/demand >1.00 for all sections. An overturning accident for a cask in transfer or in separate storage on a pad is to be assumed unless more severe overturning also occurs as a result of a natural phenomenon.
$U > D + L + H + T_a$		ANSI 57.9	Capacity/demand >1.00 for all sections
$U > D + L + H + T + W_t$		ANSI 57.9 ACI 349	The load combination (capacity/demand >1.00 for all sections) shall be satisfied without missile loadings. Missile loadings are additive (concurrent) to the loads caused by the wind pressure and other loads, however local damage may be permitted at the area of impact if there will be no loss of intended function of any structure important to safety.
$U > D + L + H + T + F$		ACI 349	Capacity/demand >1.00 for all sections
$U_s > 1.4\ D + 1.7\ L$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + W$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + E$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + A$		ANSI 57.9	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.
$U_s > D + L + W_t$		ANSI 57.9 ACI 349	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions. Computation for tornado missile loadings not required.
$U_s > D + L + F$		ACI 349	Vertical and lateral soil bearing capacities/demand > 1.00 for all foundation reactions.

TABLE 3-1 LOADS AND LOAD COMBINATIONS (cont.)

Symbol	Capacity or Load Term	Source (Reference)	Capacity or Load (or Demand) Description
$U_f > D + L$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + 1.7\ H$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + 1.275\ H + T$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + 1.275\ H + T + W$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + H + T + E$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + H + T + A$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + H + T_a$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + H + T + W_t$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$U_f > D + L + H + T + F$		ANSI 57.9	Capacity/demand >1.00 for foundation sections.
$O/S \geq 1.5\ (D + H)$		ANSI 57.9	Capacity/demand ≥ 1.00 for structure, to be satisfied for both overturning and sliding
$O/S \geq 1.1\ (D + H + E)$		ANSI 57.9	Capacity/demand ≥ 1.00 for structure, to be satisfied for both overturning and sliding
$O/S \geq 1.1\ (D + H + W_t)$		ANSI 57.9	Capacity/demand ≥ 1.00 for structure, to be satisfied for both overturning and sliding

VII. REFERENCES

Except for Federal Regulations, the documents listed below are suitable as references in SAR relevant to structural design and evaluation only to the extent described in this section and its appendices. References are to the latest version of the document, except where the specific edition is indicated. Additional requirements and criteria that are included in cited documents by their reference to further documents are considered as equivalent to their inclusion in the cite. References to Parts of the Code of Federal Regulations shall be presumed as being to the current Code. The "current Code" is considered to include all changes effective as of the date of submission of the application for approval. [Note: The date of publication in the Federal Register of changes and additions to the Code is earlier than the effective date.]

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code , Section III, "Rules for Construction of Nuclear Power Plant Components"

2. U.S. Nuclear Regulatory Commission Regulatory Guides:
   - 1.76 "Design Basis Tornado for Nuclear Power Plants," April 1974.
   - 3.53 "Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation," July 1982.
   - 3.60 "Design of an Independent Spent Fuel Storage Installation(Dry Storage)."
   - 7.6 "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," U.S. Nuclear Regulatory Commission, March 1978, Rev.1
   - 7.11 "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 inches (0.1m) " June 1991
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4.0 THERMAL EVALUATION

I. REVIEW OBJECTIVE

The purpose of the thermal review is to ensure that the cask and fuel material temperatures remain within the allowable values or criteria for normal, off-normal, and accident conditions. This includes confirmation that the fuel cladding temperatures (fission product barrier) are maintained throughout the storage period to protect the clad against degradation leading to gross rupture. The review also confirms that the thermal design of the cask has been analyzed with acceptable analytical and/or test methods.

II. AREAS OF REVIEW

This section evaluates the design and analysis of cask thermal performance for normal, off-normal, and accident conditions. This section includes guidance on review of thermal design criteria, design features, model specifications, and material properties. Guidance on review of thermal analyses, including computer programs, temperature and pressure calculations, and conduction of NRC independent evaluations is also provided.

The following areas are considered:

1. "Cask System Thermal Design"
   a. "Design Criteria"
   b. "Design Features"
2. "Thermal Load Specification/Ambient Temperature"
3. "Model Specification"
   a. "Configuration"
   b. "Material Properties"
4. "Thermal Analyses"

a. "Computer Programs"
b. "Temperatures and Pressures"
c. "Confirmatory Calculations"

5. "Supplemental Information"

III. REGULATORY REQUIREMENTS

10 CFR Part 72 requires that an analysis and evaluation of cask system thermal design and performance demonstrate that the cask will enable safe storage of the spent fuel for a minimum of 20 years with an adequate margin of safety. The spent fuel cladding must be protected against degradation which may lead to gross ruptures. Thermal structures, systems, and components important to safety must be described in sufficient detail to enable evaluation of their effectiveness. Applicable Part 72 thermal requirements, in part, are identified in 10 CFR Part 72.24(c)(3), 72.24(d), 72.122(h)(1), 72.122(l), 72.128(a)(4), 72.236(f), 72.236(g), and 72.236(h).

IV. ACCEPTANCE CRITERIA

1. Fuel-cladding (zircalloy) temperature at the beginning of dry cask storage is generally below 380°C for 5-year cooled fuel and 340°C for 10-year cooled fuel with an adequate margin of safety for normal-conditions and a minimum of 20 years cask storage. (see references 12 and 13)

2. Fuel-cladding (zircalloy) temperature is maintained below 570 °C (1058°F) with an adequate margin of safety for short term accident conditions. (PNL-4835[1])

3. The maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions assuming 1 percent, 10 percent, and 100 percent ruptured fuel rods respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.

4. Cask and fuel materials are maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions so as to support the performance of a component's intended safety function.

5. The cask system is to be passively cooled [10 CFR 72.236(f)].

6. The thermal performance of the cask is within the allowable design criteria specified in SAR Section 2 (e.g. materials, decay heat specifications) and SAR Section 3 (e.g. thermal stress analysis) for normal, off-normal, and accident conditions.

V. REVIEW PROCEDURES

Design features and design criteria, initially presented in Sections 1 and 2 of the SAR, should be reviewed for additional detail. The reviewer should examine heat loads from both contents and external sources, and assess models used by the applicant for thermal analyses. Temperatures and pressures calculated in the SAR are confirmed to evaluate compliance with design and criteria and regulatory requirements.

One of the most important results of the thermal evaluation is confirmation that the fuel cladding temperature is sufficiently low to prevent unacceptable degradation during 20-year storage. Temperature distributions and temperature criteria that are used in determination of thermal stresses for all cask system components exposed to heat generated by the fuel are evaluated. These cask system components include the cask, transfer equipment, and any shielding components.

Temperature, temperature distributions, and pressures developed within the confinement cask are evaluated. They are further used and evaluated in Section 3, "Structural Evaluation", and Section 7, "Confinement Evaluation". Evaluation of the computation of stresses or loads caused by temperature gradients and interacting materials at different temperatures or with different coefficients of thermal expansion is part of the review under Section 3, "Structural Evaluation".

Thermal performance of the cask under accident conditions, as evaluated in this section, is also addressed as appropriate in the overall accident analyses presented in Section 11 of the SER.

1. Cask System Thermal Design

   a. Design Criteria

   Review the principal design criteria and structure, system and component specifications presented in Section 2 of the SAR and any additional detail provided in Section 4.

   b. Design Features

   Review the description of the significant thermal design features and operating characteristics of pertinent subsystems of the cask system. Design features typically include, in part, the cask body, thermal fins, shielding materials, fuel baskets, containment seals, drain and vent ports, and pressure relief devices. Verify that the thermal design features will adequately perform their intended safety function during normal, off-normal, and accident conditions.

   All thermal design features must be passive. Review the specifications for cask system components for inclusion of material composition and thermal properties, operating pressures and criteria, and operating pressures and criteria for any relief devices or rupture disks. Review the general description of the cask presented in Section 1 of the SAR and any additional information provided in Section 4. In addition to material compositions, dimensions of the cask components, and spacing of fuel assemblies in the basket, the thermal design may include external air passages. Rupture disks may be used on cask components, such as shielding, to ensure that elevated temperatures do not cause thermal expansion or phase changes resulting in structural damage to the cask shell. All drawings, figures, and tables must be sufficiently detailed to support an in-depth staff evaluation.

Any instrumentation used to monitor cask thermal performance must be described in enough detail to support an in depth staff evaluation . The monitoring instrumentation components must have a safety classification commensurate with their function. The safely classification, presented in Section 2 of the SAR, must be justified. Applicable operating controls and criteria, such as temperature criteria and surveillance requirements, should be clearly indicated in Section 12 of the SAR, discussed in the SER, and included in the license or certificate of compliance as appropriate..

2. Thermal Load Specification/Ambient Temperature

Examine the specification for the design-basis fuel decay heat presented in Section 2 of the SAR. Ensure that this decay heat is consistent with the specified burnup and cooling times, if included. Generally, decay heat is calculated using the same computer codes as those used to determine radiation source terms. Coordinate the review of fuel source terms for consistency with the shielding review, as appropriate. Alternatively, the decay heat from the design basis fuel may also be derived from Regulatory Guide 3.54.[2] Except for neutrino energy, all decay heat should be considered to be deposited in the fuel.

If control components or other assembly hardware (e.g., shrouds) are included with the fuel assemblies, their heat loads should be specified and justified.

In general, Staff accepts solar insolance presented in 10 CFR Part 71[3] for 10 CFR Part 72 applications. Because of the Large thermal inertia of a storage cask, the values listed in 10 CFR Part 71.71 may be treated as the average insolance over a 24-hour day (12 hours of daylight) in a steady-state calculation. If a less conservative approach is presented, the SAR must thoroughly describe and justify its use.

Review the ambient temperatures used for calculating temperature distributions and maxima for normal, off-normal, and as "design basis" natural phenomena. The assumed temperatures and temperature variations with time should be clearly stated in the SAR. These assumptions form criteria for comparison with recorded data and/or projections for potential installations of the cask system at specific sites.

When calculating maximum thermal gradients and temperature differences within individual components or between locations, changes in temperature over time may need to be determined. These changes over time should consider the thermal mass of specific components . Statements about assumed bounding temperatures ranges, ambient temperature conditions, and variations of external heat sources over time should be defined so that these may be easily compared with available site or regional data. The general licensee must also determine if its site conditions are bounded by the SAR computations.

For those cask system components whose material properties and performance vary with temperature, review the assumptions used in determining temperature maxima, minima, gradients, and differences for the different cask system. Also review the assumptions used to determine fuel clad temperatures. The changes in temperatures over time that were assumed should result in the worst conditions for the structural analysis. The calculated temperatures, in the various cask system components, should be compared to the limiting temperature criteria for the appropriate material. Ferritic materials are subject to failure by brittle fracture at low temperatures. Review the assumed low temperatures for cask system handling operations for consistency with material properties.. Ambient temperature restrictions may be appropriate for cask handling operations. Any limiting conditions on ambient temperatures should be addressed in Chapter 12 of the SAR, discussed in Section 12 of the SER, and included in the license or certificate of compliance as appropriate.

For unloading operations, evaluate temperature and pressure calculations supporting procedural steps for cool down of the cask and reflood of the cask internals presented in SAR Chapter 8. To ensure that the cask does not over pressurize or there are not excess thermal stresses on the fuel assemblies, the applicant's analysis should specify and justify the appropriate temperature and flow rate of the quench fluid, assuming maximum fuel cladding temperatures. This review should be coordinated with the Section 3 structural review and Section 8 procedures review.

Analysis for accident-level ("design basis") temperatures should not be considered to envelope the analysis of normal or off-normal temperatures. Normal and off-normal temperature demands for structural capacity have different acceptance criteria. Therefore, all three conditions should be analyzed. In addition, the duration over which accident temperature conditions exist should be evaluated. Because material properties may be functions of temperatures and time, long term elevated temperatures can cause gradual degradation of material properties (such as in concrete).

3. Model Specification

a. Configuration

Verify that the model used in the thermal evaluation is clearly described. Separate models may be used for the evaluation of normal and accident conditions. Coordinate with the structural review to evaluate any damage that may result from accidents or natural phenomena events. All models should be shown to be conservative.

Examine the sketches or figures of the model used for thermal calculations. Verify that the dimensions and materials of the model are consistent with those in the drawings of the actual cask, as presented in Section 1 of the SAR. If possible, examine computer inputs to verify consistency with model sketches and engineering drawings. Differences between the actual cask configuration and the models should be identified, and the models shown to be conservative.

Pay particular attention to gaps between cask components. Tolerances should be considered so that the thermal resistance of each gap is treated conservatively. Gases (e.g., air, helium) assumed to be present in the gap must be described and justified. If a specific gas other than air, in the cask cavity or gaps between cask components, is relied upon for heat removal, then the applicant must show that the gas is retained during the entire storage period. For cask components that are important to heat removal, manufacturing techniques for joining components, surface roughness, contact pressures, and gap conductance values must be adequately described and justified.

Review the decay heat load axial distribution. Ensure that decay heat generated in the spent fuel is limited to the active fuel region of the assemblies. The model should specifically account for the peaking in the central region. Heat from control components, if applicable, should also be distributed appropriately. The position of heat sources relative to other cask components must be identified.

Examine the heat-transfer processes used in the analyses. Typically conduction and radiation are defined as the primary heat transfer mechanisms within the cask itself. Convection by natural circulation should be limited to that between the external surface of the cask and the ambient environment. The staff has not previously approved thermal models for natural circulation internal to the cask because of the difficulty in modeling and lack of test data.. Use of effective thermal conductivity coefficient for regions within the confinement cask, other than the fuel (e.g., gaps), may overestimate heat transfer. If such use is made of effective thermal conductivity, verify that the same values have been determined from test data that are representative of similar geometry, materials, temperatures, and heat fluxes used in current application. Pay particular attention to the effective thermal conductivity of neutron shield regions such as those embedded with thermal fins. Voids or gaps typically exist, either caused by tolerances or by shrinkage, and must be considered in calculating effective conductivity. Also, pay particular attention to the values assumed for surface emissivities and view factors and the manner of accounting for radiation heat transfer in determining the effective thermal conductivities.

Coordinate the thermal review with the structural review to ensure that, for components external to the confinement cask, situations that may produce the worst cask loads are analyzed. In illustration, for cask systems that may have multiple shielded casks and/or ones that provide cooling air passages by a single, integral structure, the greatest gradients and loadings caused by thermal expansion may occur with casks in alternate storage or in temporary handling positions, whereas the highest material temperatures probably occur with casks in every position.

Review how the SAR treats heat transfer through the fuel assemblies, and, if applicable, the manner in which an effective conductivity of each fuel assembly is determined. Guidance on effective thermal conductivity of the fuel is presented below in the discussion on material properties.

Verify that the SAR addresses the thermal interaction among casks in an array by using a view factor less than unity. Generally, this will result in an operating control and limit in Section 12 that imposes a minimum spacing between storage casks.

b. Material Properties

Verify that the material compositions and thermal properties are provided for all components used in the calculational model. Verify that the thermal properties used in the safety analysis are appropriate and that potential degradation of materials over their service life has been evaluated and taken into account. The source of the thermal property data must be traced to an authoritative reference (generally not a text book). NRC has accepted the ASME B&PV Code, Division 1, Section II Material Specifications and the Section III appendices as a primary source for material properties. Pay particular attention to non-standard materials (e.g., neutron shielding and seals). Temperature and anisotropic dependencies of thermal properties must be considered. If regional thermal properties are determined from a combination of individual materials, the manner in which these effective properties are calculated must be described.

If the transverse effective thermal conductivity of the fuel is greater than 0.5 BTU/hr-ft-°F (~0.86 W/m-°K) (under the conditions described in Ref. [4] [5] the method in which it was determined must be thoroughly described and supported. If the thermal model is axisymmetric or three-dimensional, the longitudinal thermal conductivity should generally be limited to the conductivity of the cladding (weighted by its fractional area) within the fuel assembly. Gaps between fuel pellets and cracks in the pellets themselves can result in a considerable uncertainty on the contribution of the fuel to longitudinal heat transfer.

Both maximum and minimum temperature criteria on cask material and components must be indicated. Justification and references for these criteria must be adequately described. Criteria on concrete temperatures are established by ACI 359[6] for structures, systems and components within the scope of that code; and, by Appendix A to ACI 349[7]

4. Thermal Analysis

a. Computer Programs

Determine which computer codes were used in the thermal evaluation. The applicant's computer code used to perform the thermal evaluation should be well verified and validated. The two most frequently encountered codes in SARs are ANSYS[8] and HEATING[9]. Both are capable of general 3-D steady state and transient calculations, as deemed necessary. Assess that the number of dimensions and temporal treatment are appropriate for the calculations being performed.

At least two codes, SCANS[10] and CASKS [11], have been developed to perform simple, approximate confirmatory analyses. These codes are not acceptable for use in a SAR for thermal design and analysis. In addition, since these codes address temperatures of only the cask body, they cannot be used as the sole confirmatory tool for the thermal review.

The SAR documentation should include input and output file listings for the thermal evaluations. The reviewer should be familiar with the codes used in the SAR documentation. If use of codes not previously accepted by NRC is proposed, development of reviewer familiarity with those codes is considered to be part of the necessary review process. The applicant should also describe, in the SAR, the code and justification for use in the thermal evaluation. Verify that the information from the thermal model is properly input into the code. Verify that the output has been properly interpreted and applied in the thermal and structural analyses. The scope of confirmatory calculations is partly dependent on the quality of the output data and its use.

b. Temperatures and Pressures

The SAR should include a table that lists the maximum and minimum temperatures of all components important to safety under normal, off-normal and accident conditions. The table should specify each component's operating temperature range. Verify that temperatures for key components have been calculated and that they do not exceed the allowable range. Justification must be provided in the SAR for any material important to safety that exceeds acceptable temperature ranges. If compliance with minimum temperature criteria relies on a specific minimum heat load from the fuel, such heat load must be quantified and included as an operating control and criterion in Section 12 of the SAR.

Pay particular attention to the cladding maximum temperature. Staff accepts temperature criteria based on the diffusion controlled cavity growth (DCCG)[12]. Comparable criteria are also defined in PNL-6189[13], even though the maximum temperatures are based on a different failure mechanism (creep). Experience with previous SARs has shown that most of the review effort is generally devoted to confirming that these temperature criteria are satisfied.

Based on the calculated temperature distributions, verify the pressure of the gas in the cask cavity. This pressure will be dependent on the initial volume of the cover gas, the volume of fill gas in the fuel rods, the number of rods assumed to have ruptured, and the fraction of the fission product gas trapped in the fuel matrix. NRC accepts, as minimums, that normal-conditions occur with at least 1% of the fuel rods ruptured, that off-normal-conditions occur with up to 10% of the fuel rods ruptured, and that 100% of the fuel rods will have ruptured following a design basis accident event. NRC also accepts that a maximum of 30% of the significant radioactive gases within a ruptured fuel rod is available for release into the cask cavity (see Section 7, paragraph V.3). Verify that design criteria pressures stated in Section 2 are consistent with the calculated maximum pressures. Verify that the pressure testing specified in Section 9 is consistent with the calculated pressures.

Some storage systems rely upon natural circulation flow of air through internal passages to remove heat from the stored confinement cask. For storage systems with internal air flow passages, blockage of inlet and/or outlet flow is an accident situation that must be evaluated. Total blockage of all inlets and outlets will result in heatup of the fuel which approaches adiabatic conditions. Part Ia. blockages can result in operating temperatures higher than those for normal operation. To assure that blockages do not go undetected for significant periods, NRC Committee to Review Generic Requirements has required objective evidence that inlet and outlets flows are not obstructed. Consequently, for this type of storage systems, NRC has accepted periodic visual inspections coupled with temperature measurements to verify proper thermal performance to detect flow blockages.. The inspection interval should be more frequent than the time required for heatup of the fuel to the accident temperature criteria with total blockage of all inlets and outlets.

Review of the heatup calculations should especially address any assumptions regarding limiting components and quasi-steady state responses. The initial ambient temperature for the heatup calculation should bound the maximum "normal condition" temperature. The resulting heatup time history should be included in the SAR documentation. This supports the proposed inspection and monitoring frequency. The information is also useful, to users, in developing contingency operation procedures. It provides information about the available time in which to take corrective actions before the fuel accident temperature criteria may be exceeded.

The most extreme thermal conditions may result from credible ambient temperatures, temperature-time histories, an adjacent fire, or any off-normal or DBE that result in ventilation passage blockage. The worst case structural loads may occur at temperatures lower than those of design basis accidents or natural phenomena, since load combination expressions effectively require greater safety factors for normal and off-normal analyses than for design basis (accident) events and conditions. For storage systems without internal air flow passages, the worst case accident thermal conditions typically have been fire.

Some structures, systems, and components may experience the most severe conditions if exposure to high temperatures is followed by dousing (as by rain or fire water). Burning of fuel and other combustibles (e.g., tires) associated with vehicles involved in transfer operations should be presumed as a design basis event with the cask in the most exposed situation during transfer or loading into storage. If the SAR does not address fire or if the site-specific fire parameters exceed those of the SAR, the site-specific application will need to include analysis of the worst case credible fire.

Fire parameters included in Part 71 [10 CFR Part 71.73] have been accepted for characterizing the heat transfer during the in-storage fire. A small amount of exterior concrete spalling that may result from a fire or other high temperature condition and/or application of fire water or rain on heated surfaces. The small amount is not expected to exceed the regulatory requirements in 10 CFR Part 72.106 and therefore does not need to be estimated or evaluated in the SAR. Any significant spalling damage is readily detectable and appropriate recovery or corrective measures may be presumed. NRC accepts that concrete temperatures may exceed the temperature criteria of ACI 349 for accidents if the temperatures result from a fire. In that case, corrective action may be required for continued safe storage.

The methods that are acceptable for analyzing and reviewing the consequences of a fire are dependent upon the duration of the fire and the margin between the predicted temperatures and the actual component thermal limits. For a very short duration fire, calculation of the fuel temperature increase by assuming that the cask wall is adiabatic is acceptable. For a somewhat longer fire, an evaluation of the cask body with no credit for the thermal capacity of the fuel assemblies and basket is acceptable. A 30-minute fire, or one in which material temperatures are close to the criteria of their acceptable operational range, will require a detailed model of the cask and its contents. Cask system components (e.g., neutron shield) may be assumed to be intact at the start of the fire, unless the fire is a secondary effect resulting from another credible event which may have physically affected the cask system (e.g., an aircraft impact).

Some storage systems may use a transfer cask to move the loaded confinement cask to the ISFSI storage site. When the confinement cask is within the transfer cask, cooling is typically less than for normal operation. Fuel cladding temperatures would therefore be expected to be higher than for normal storage conditions. Fuel temperatures may rise toward defined criteria.

Examine the temperature distribution calculations for the fuel container inside the transfer cask. Verify that heat transfer through gap regions has been treated in a conservative manner, and that material properties and dimensions of the transfer cask are consistent with the design data defined in the SAR documentation. The initial ambient temperature should be the maximum "normal condition" temperature. Cask preparation for storage or for unloading operations may include situations when the confinement cask is evacuated while it is in the transfer cask. For such conditions, determine that the maximum fuel cladding temperature has been calculated in an acceptable manner. The short term accident temperature of 570 °C (1058°F) for zircalloy fuel cladding has been accepted as a suitable criterion for fuel transfer operations. If the calculation is based on heatup over a limited time period, verify that limiting conditions for the operations have been imposed which assure that the temperature will remain acceptable during the process, and that normal cooling will begin before the temperature criterion is exceeded.

Using the accident condition temperatures, verify that the SAR has correctly determined the post accident pressure of the gas in the cask cavity. The pressure should be determined based on the assumption that 100% of the fuel rods have failed. The resulting load on the cask confinement boundary will be used in the structural analysis with the appropriate load combinations for accident conditions.

c. Confirmatory Calculations

The reviewer should perform confirmatory evaluations of the thermal performance of these cask structures, systems and components identified as important to safety. These should specifically include steady-state temperature distributions, local heat balances, temperatures reached and temperature distributions within any reinforced concrete structures, systems and components, and cask cavity pressures for the bounding ambient temperatures. To the degree possible, confirmatory calculations should use a different thermal method to provide the most reliable confirmation. Similar confirmation of transient temperatures (e.g., during a fire) should be performed, as applicable to the SAR analysis.

The minimum confirmatory review should include verification that key design parameters have been appropriately determined and correctly input into the computer program(s) used for the thermal analysis. Key parameters include proper dimensions, material properties (including surface emissivities and view factors for radiation), and definition of heat sources. A heat balance at the outer surface of the cask should be performed to verify that the heat from the spent fuel and solar insolance equal that removed by convection and radiation. Correlations for the heat transfer coefficient should then be assessed to confirm that they are appropriate for the existing storage conditions. The temperature of the cask inner surface should be estimated by calculating the temperature distribution across the cask body with simple heat balance approximations. Finally, the difference between the cask inner surface temperature and the maximum cladding temperature should be compared with that of similar casks/baskets reviewed in previous SARs.

If a more detailed confirmatory review is required, a portion of the cask or basket may be modeled and evaluated to assure that the SAR results are realistic and conservative. An extensive confirmatory evaluation is necessary if major errors are suspected, if the applicant's margin in a complex analysis is small, or if little conservatism exists in the SAR modeling approach. As an alternative, the applicant may be required to perform design-verification testing of an as-built cask system to confirm the thermal analyses presented in the SAR. Testing may include tests to verify gap conductance values assumed in modeling thermal resistance. The test conditions, configuration, and type and location of instrumentation used, if any, should be sufficiently described in Section 9 of the SAR.

NRC accepts simplifying assumptions for the effects of reinforcing steel in determining the thermal performance and temperature distributions of reinforced concrete. Use of a homogeneous material, instead of modeling the concrete and reinforcing steel, as separate elements, is acceptable if the substitute hypothetical material has appropriately adjusted the thermal properties, and the reinforcing steel is cover with concrete in accordance with the applicable structural code. Thermal performance and/or temperature distributions for reinforced concrete designs which have features that provide for significant thermal transfer below the concrete surface (as by internal studs welded to an exposed steel plate) may require more specific analysis.

5. Supplemental Information

Supplemental information can include copies of applicable references (if not generally available to the reviewer), computer code descriptions, input and output files, and any other information that the applicant has deemed necessary. Likewise, the reviewer should request any additional information needed to complete the review process.

VI. EVALUATION FINDINGS

Review the Part 72 acceptance criteria and provide a summary statement for each similar to the following:

Thermal structures, systems, and components (SSCs) important to safety are described in sufficient detail in Sections _____ of the SAR to enable an evaluation of their effectiveness. Cask structures, systems and components important to safety remain within their operating temperature ranges.

The [cask designation] is designed with a heat-removal capability having testability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.

The spent fuel cladding is protected against degradation that leads to gross ruptures by maintaining the cladding temperature for _____ -year cooled fuel below ______ °C in an [applicable gas] environment. Protection of the cladding against degradation will allow ready retrieval of spent fuel for further processing or disposal.

The staff concludes that the thermal design of the [cask designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the [cask designation] will enable safe storage of spent fuel. This finding is based on a review which considered the regulation itself, appropriate Regulatory Guides, applicable codes and standards, and accepted engineering practices.

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN FOR STORAGE CASKS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

5.0 SHIELDING EVALUATION

I. REVIEW OBJECTIVE

The purpose of the shielding review is to ensure that the shielding features of the proposed cask system as designed for an ISFSI provide adequate protection against direct radiation from the cask contents and limit the dose to the operating staff and members of the public to be within regulatory requirements during normal operating and design basis accident conditions.

II. AREAS OF REVIEW

1. "Shielding Design Description"
   a. "Design Criteria"
   b. "Design Features"
2. "Radiation Source Definition"
   a. "Gamma Source"
   b. "Neutron Source"
3. "Shielding Model Specification"
   a. "Configuration of Shielding and Source"
   b. "Material Properties"
4. "Shielding Analyses"
   a. "Computer Programs"
   b. "Flux-to-Dose-Rate Conversion"
   c. "Dose Rates"
   d. "Independent Calculations"
5. "Supplementary Information"

III. REGULATORY REQUIREMENTS (Part 72)

Part 72 requires that spent fuel radioactive waste storage and handling systems be designed with suitable shielding to provide adequate radioactive protection under both normal and accident conditions. The shielding structures, systems, and components (SSCs) important to safety must be described in sufficient detail to enable evaluation of their effectiveness. Such SSCs should be analyzed with the objective of assessing the impact on public health and safety. SSCs important to safety must be designed to withstand the effects of both credible accidents and severe natural phenomena without impairing their capability to perform their safety functions. The applicable 10 CFR Part 72 shielding requirements, in part, are identified in 72.24(c)(3); 72.24(d); 72.104(a); 72.106(b); 72.122(b); 72.122(c); 72.128(a)(2); and 72.236(d).

IV. ACCEPTANCE CRITERIA

The task of identifying dose rate limits for direct radiation from storage casks is complicated by three considerations. First, regulatory dose limits are stated in terms of total absorbed doses rather than dose rates. Second, dose analyses must include other potential sources of radiation besides direct radiation from spent fuel in the cask. Third, the regulatory requirements (listed below) for acceptable cask use at an ISFSI are site-specific and must be separately evaluated on a case-by-case basis. These evaluations are performed as required in a site-specific license application or as required by 10 CFR Part 72.212 for a utility using a cask under the general license.

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters.

2. During normal operations and anticipated occurrences:

   a. Combine the doses from direct radiation, planned discharges, and uranium fuel cycle operations within the region.

   b. The annual dose equivalent to an individual outside the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ.

3. Dose rates from the cask must be consistent with a well-established as low as reasonably achievable (ALARA) program for activities in and around the storage site.

4. After a design basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ.

5. The occupational dose limits and radiation dose limits for individual members of the public in 10 CFR Part 20 [Subparts C and D] must be met.

V. REVIEW PROCEDURES

1. Shielding Design Description

a. Design Criteria

Review the design criteria presented in Section 2 of the SAR and any additional shielding-related criteria. Confirm that the criteria are compatible with shielding assumptions made for dose rate calculations for both occupational workers and the public. Dose rates at the cask surface and in the vicinity of a loaded cask may vary during storage, transfer, and in-storage activities. Acceptable dose rates are dependent on the total expected exposure to workers during ISFSI-related operations, or to the public, who are assumed to be at the closest boundary of the controlled area, 100 meters from the storage cask. NRC has accepted a range of surface dose rates, but the doses calculated for the workers and public must comply with the criteria in 10 CFR Parts 20 and 72.

Part 72 does not establish specific cask dose rate limits. Previously, cask dose rates from 20 to 400 mrem/hour have been accepted in Part 72 evaluations. Acceptable dose rates depend on a number of factors such as the geometry of the storage array, the proximity to the site boundary or other public access areas, the routine time workers will spend in the storage array for activities like monitoring or maintenance, and the proximity to other areas frequently occupied by workers. For example, the review should consider the proximity of the storage array to equipment that must be monitored or serviced frequently. Also, high dose rates at the cask top or bottom may be acceptable if these areas are not routinely occupied during storage and the expected exposure, during cask transfer are controlled; however, the dose rates at the side of the same cask may need to be low if personnel would be in that area on a periodic schedule.

b. Design Features

Review the general description of the cask presented in Section 1 of the SAR and any additional information provided in Section 5. All drawings, figures, and tables describing shielding features must be sufficiently detailed to support an in-depth staff evaluation.

2. Radiation Source Definition

Examine the design basis fuel described in Section 2 of the SAR to verify that the source term calculations are based on the fuel that will provide the bounding source. The SAR should examine all fuel designs and conditions of burnup for which the cask system is to be certified to ensure that the bounding fuel type and values are used. Particular attention should be given to the enrichment, burnup, and cooling time. Generally the specifications in Section 2 will indicate a maximum fuel enrichment that is used in the criticality analysis. For shielding evaluations, however, the neutron source term increases considerably with decreasing initial enrichment and constant burnup. Consequently, the SAR either may specify a minimum initial enrichment or establish the specific source terms as operating controls and limits for cask use.

Generally the applicant will use ORIGEN-S[1] (e.g., as a SAS2 sequence of SCALE), ORIGEN2[2], or the U.S. Department of Energy Characteristics Data Base[3] to determine the source terms. Although the latter two are easy to use, both have energy group structure limitations that are discussed below. If the applicant has chosen ORIGEN2, verify that the chosen cross section library is appropriate for the fuel being considered. Many libraries are not appropriate for a burnup that exceeds 33,000 MWd/MTU.

a. Gamma Source

Verify that the gamma source terms are specified as a function of energy for both the spent fuel and activated hardware. If the energy group structure from the source term calculation differs from that of the cross section set of the shielding calculation, the applicant may need to regroup the photons. Regrouping can be accomplished by using the nuclide activities from the source term calculation as input to a simple decay code with a variable group structure. Some applicants will merely interpolate from one structure to the other. In general, only gammas with energies from approximately 0.8 to 2.5 Mev will contribute significantly to the dose rate through typical types of shielding, so regrouping outside this range is of little consequence. Pay attention to whether the source terms are specified per assembly, per total assemblies, or per metric ton, and ensure that the total source is correctly used in the shielding evaluation.

Determining source terms for fuel assembly hardware is generally not as straightforward as for the spent fuel, especially if one of the ORIGEN codes is used. The effort devoted to reviewing this calculation should be appropriate to the contribution of these terms to the dose rates presented in the shielding evaluation. Also, note whether the cask is intended to contain special hardware, such as control assemblies or shrouds, and ensure that source terms from these components are included if applicable.

Depending on the cask design, neutron interactions may result in the production of energetic gammas being produced near the cask surface. If this source term is not treated by the shielding analysis code, verify that it is determined by other appropriate means.

As part of the source term determination, the applicant must calculate the quantities of certain nuclides (e.g., ^{85}Kr, ^{3}H, and ^{129}I) for use in analyzing doses from the release of radioactive material in later sections of the SAR. If these are presented in the section on shielding evaluation, they should be reviewed at this time. Often the applicant will tabulate all nuclides that are important to the direct radiation dose rate. This information can be used to resolve differences that may exist between the source terms of the applicant and those of the reviewer.

b. Neutron Source

Verify that the neutron source term is expressed as a function of energy. The neutron source will generally result from both spontaneous fission and alpha-n reactions in the fuel. Depending on the method used to determine these source terms, the applicant may need to determine the energy group structure independently. This is often accomplished by selecting the nuclide with the largest contribution to spontaneous fission (e.g., ^{244}Cm) and using that spectrum for all neutrons, since the contribution from alpha-n reactions is generally small.

3. Shielding Model Specification

Verify that the models used in the shielding evaluation are adequately described for storage under both normal and accident conditions. For example, if the cask has an external neutron shield, determine whether it would be damaged by a tip-over accident or degraded in a fire. Coordinate this analysis with the structural and thermal reviews, as appropriate.

a. Configuration of the Shielding and Source

Examine the sketches or figures that indicate how the shielding is modeled. Verify that the model dimensions and materials are consistent with those specified in the cask drawings presented in Section 1 of the SAR. Ensure that voids, streaming paths, and irregular geometries are taken into account or otherwise treated in a conservative manner. Differences, if any, between normal-conditions of storage and accident conditions must be clearly stated.

Verify that source term locations for both spent fuel and structural support regions are modeled properly. Within the spent fuel region, the fuel and basket materials may be homogenized to facilitate shielding calculations. Because of a cosine burnup profile, a uniform source distribution is generally conservative for the top and bottom, but not for the axial center. If axial peaking appears to be significant, verify that it has been treated appropriately. The assembly structural support regions (e.g., top and bottom end pieces and plenum) should be correctly positioned relative to the spent fuel. These support regions may be individually homogenized with the basket materials. Generally, at least three source regions are necessary (i.e., fuel and top/bottom assembly hardware).

Verify that the locations for the various dose calculations are shown or adequately described. Ensure that these dose points are representative of all locations relevant to radiation protection issues. Pay particular attention to dose rates from streaming paths to which occupational workers would be exposed (e.g., at vent/drain port covers, lid bolts, etc.). Also, note the end points of shielding such as lead in the cask wall in relation to the assembly hardware. Additional information regarding the selection of locations for dose calculations is provided in Section 5.4.c, below.

b. Material Properties

Verify that material compositions and densities are provided for all materials used in the calculational model. For nonstandard materials (e.g., neutron shields) the source of the data must be referenced, and validation criteria must be indicated in Section 9. Many shielding codes will allow the densities to be input directly in g/cm^3. If input is required in atoms/barn-cm, pay particular attention to the conversion.

Confirm that temperature-sensitive shielding materials will not be subject to temperatures at or above their design limitations during both normal and accident conditions. Determine if the potential for shielding material to experience changes in material densities at temperature extremes has been properly examined. For example, elevated temperatures may reduce hydrogen content through loss of bound or free water in concrete or other hydrogenous shielding materials.

4. Shielding Analyses

a. Computer Programs

Examine the computer program(s) used for the shielding analysis. These codes may use Monte Carlo, deterministic transport, or point-kernel techniques for problem solution. (The point-kernel technique is generally appropriate only for gammas, since casks typically do not contain sufficient hydrogenous material to apply removal cross sections for neutrons.) Some shielding codes whose use has been accepted by NRC include, TORT/DORT [4], ONEDANT, ONEDANT[5,6], and MCNP[7].

Other codes and models whose use has been previously accepted by NRC for ISFSI source and shielding analyses include:

ANISN (one-dimensional neutron attenuation code);

MICROSKYSHINE (air-scattering code);

MORSE (Monte Carlo multigroup three-dimensional neutron and gamma transport computer code);

MCBEND (Monte Carlo multigroup three-dimensional neutron and gamma transport computer code similar to MORSE developed by the United Kingdom (UK) National Radiation Protection Board (NRPB));

QAD-CGGP (three-dimensional point kernel gamma transport shielding computer code);

RANKERN (three-dimensional point kernel gamma transport shielding computer code similar to QAD-CGGP);

MARC-1 (a suite of linked computer codes used for calculating the radiological effects of releases of radionuclides to the environment developed by the UK NRPB);

LINGAP and HMARC (modules of MARC-1 used to calculate the effects of an atmospheric release);

STREAMING (code for calculation of attenuation of a gamma flux incident on a variety of shielding penetrations, such as ducts and voids);

Determine that the number of dimensions of the code is appropriate for the dose rates being calculated. Generally at least a two-dimensional calculation is necessary. One-dimensional codes provide little information about off-axis locations and streaming paths that may be significant to determining occupational exposure. Even for dose rates at the end of the cask, one-dimensional codes require a buckling correction that is not particularly straightforward, since merely using the cask cavity diameter may underestimate the dose rate.

The SAR should include a representative computer code input file. As discussed in Section 5.4.d below, if the reviewer is familiar with the code used in the SAR analysis, examining the input file can significantly expedite the review. Verify that the information from the shielding model is properly input into the code.

Verify that the cross-section library used by the code is appropriate for shielding casks. If the applicant has not independently determined a source term for neutron-induced gamma radiation or subcritical multiplication of neutrons, ensure that a coupled cross-section set is used and that the code has been executed in a manner that accounts for these secondary source terms.

b. Flux-to-Dose-Rate Conversion

The shielding analysis code will sometimes perform this conversion directly with its own data library. This conversion will most often use ANSI/ANS 6.1.1-1977[8]. Note that the 10 CFR Part 20 radiation protection requirements are based on fluence-to-dose conversions that are essentially the same as those in the 1977 version of the ANSI/ANS Standard. In practice, the photon dose rates are not very dependent on the choice of conversion factors, but the neutron dose rates as determined from the 1991 version of this Standard may be significantly less than those determined from 10 CFR Part 20 or the 1977 version of the ANSI/ANS Standard.

c. Dose Rates

Verify that the dose rates appear reasonable and that their variation with location is consistent with the geometry and shielding characteristics of the cask system. The following guidance pertains to the selection of points at which the dose rates should be calculated.

For normal-conditions, the SAR should indicate the dose rate at all locations accessible to occupational personnel during cask loading, transport to the ISFSI, and maintenance and surveillance operations. Generally these locations include points at or near various cask components and in the immediate vicinity of the cask. These dose rates will be used in conjunction with the operating procedures from Section 8 and the estimated occupational doses discussed in Section 10, to determine whether the cask system will meet Part 20 requirements.

NRC has previously accepted a calculated dose rate of 0.25 mSv/yr at the ISFSI controlled area boundary as sufficient evidence that the limit on exposure of the public will not be exceeded under normal-conditions. This may be addressed by the SAR or it may be left to the site-specific license application. If the SAR addresses this dose rate, a discussion should be provided that would help determine whether a potential ISFSI would be within the dose rate envelope. These could involve identifying the minimum controlled area dimensions to ensure that the 0.25 mSv/yr dose is not exceeded. Alternatively, the presentation could provide the maximum number of casks that could be stored in an ISFSI with the minimum distance of 100 meters between the stored fuel and the controlled area boundary (10 CFR Part 72.106(b)) to meet the 0.25 mSv/yr criterion.

To show applicant compliance with these requirements in a SAR, Staff has accepted a calculation of a dose rate less than 0.25 mSv/yr (25 mrem/yr) from one cask (or a representative array of casks) at an assumed distance to the controlled area boundary. Such calculations, in practice, can give only a general assessment of the proposed cask system. In addition to unknown information about the ISFSI itself, the implied assumption that an individual would be at the controlled area boundary for 8760 hours (the entire year) is very conservative. If the above dose rate criteria are satisfied, NRC accepts that the direct-dose regulatory requirements can also be satisfied, although the exact details needed to comply with these limitations will vary from site to site. Therefore, the SAR needs to address such requirements only in fairly general terms. Detailed calculations do not need to be presented if Section 12. "Operating Controls and Limits," of the SAR assigns ultimate compliance responsibilities to the site licensee.

The dose rate at one meter from the cask surface should be calculated for accident conditions. The model used for these calculations must be consistent with the expected condition of the cask after an accident or natural phenomena event.

d. Independent Calculations

The reviewer should perform an independent evaluation of dose rates in the vicinity of the cask for normal and accident conditions.

In determining the level of effort appropriate for these calculations, the reviewer should consider: (1) the degree of sophistication and margin in the SAR analysis; (2) a comparison of SAR dose rates with those of similar casks that have been previously reviewed, if applicable; (3) the typical variation in dose rates expected between different codes and cross-section sets; (4) the fact that actual dose rates will be monitored and limited by Part 20 requirements; (5) the restrictions that can be placed on ISFSI operations that affect measured dose rates which are documented in Section 12 of the SER, a site-specific license, or the Certificate of Compliance; (6) experience of the applicant in use of the methods and codes in prior ISFSI submittals; (7) use of new, or previously reviewed, methods or codes; and (8) inclusion in the design of any significant departures from prior cask system designs (e.g., an unusual shield geometry, use of new types of materials, or different source terms).

The minimum review should include an examination of the applicant's input to the computer program used for the shielding analysis. Verify use of proper dimensions, material properties, and an appropriate cross-section set. Gamma and neutron source terms should be independently evaluated.

If a more detailed review is required, the gamma dose rates can be evaluated with a code such as MicroShield[9] or QAD-CGGP[10] to ensure that the SAR results are reasonable and conservative. As noted earlier, the use of a simple code for neutron calculations is often not appropriate. An extensive evaluation is necessary if major errors are suspected. To the degree possible, the use of a different shielding code and cross-section set from that of the SAR analysis will provide a more independent evaluation.

5. Supplemental Information

Supplemental information can include copies of applicable references (especially if a reference is not generally available to the reviewer), computer code descriptions, input and output files, and any other information that the applicant has deemed necessary. Likewise, the reviewer should request any additional information needed to complete the review process.

VI. EVALUATION FINDINGS

Review the 10 CFR Part 72 acceptance criteria and provide a summary statement for each similar to the following:

Shielding structures, systems, and components important to safety are described in sufficient detail in Sections _____ of the SAR to enable an evaluation of their effectiveness.

Shielding structures, systems, and components important to safety are evaluated in Section _____ of the SAR with the objective of assessing the impact on health and safety resulting from the operation of the ISFSI.

The staff concludes that the design of the shielding system of the [cask designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system design provides reasonable assurance that the [cask designation] will enable safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

VII. REFERENCES

1. Petrie, L. M., et al., "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Vol. 1–4, Rev. 4, Oak Ridge National Laboratory, April 1995.

2. Oak Ridge National Laboratory, "ORIGEN2.1: Isotope Generation and Depletion Code—Matrix Exponential Method," 1991.

3. TRW Environmental Safety Systems, Inc., "DOE Characteristics Data Base, User Manual for the CDB_R," November 16, 1992.

4. Rhoads, W. A., "The TORT Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code," ORNL-6268, Oak Ridge National Laboratory, November 1987.

5. O'Dell, R. D., et al., "Revised User's Manual for ONEDANT: A Code for One-Dimensional, Diffusion Accelerated, Neutral Particle Transport," LA-9184-M, Rev., Los Alamos National Laboratory, December 1989.

6. Alcouffe, R. E., et al., "User's Guide for TWODANT: A Code Package for Two-Dimensional, Diffusion Accelerated, Neutral Particle Transport," LA-10049-M Rev., Los Alamos National Laboratory, April 1992.

7. Los Alamos National Laboratory, "MCNP 4A, Monte Carlo N-Particle Transport Code System," December 1993.

8. Institute for Nuclear Materials Management/ American Nuclear Society, ANSI/ANS 6.1.1, "American National Standard for Neutron and Gamma-Ray Fluence to Dose Factors," 1977.

9. Worku, G., et al., "MicroShield Version 4.2 User's Manual," Grove Engineering, Inc., Rockville, Maryland, 1995.

10. Radiation Shielding Information Center, "QAD-CGGP, A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor," CCC-493, Oak Ridge, Tennessee, 1994.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN FOR STORAGE CASKS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

6.0 CRITICALITY EVALUATION

I. REVIEW OBJECTIVE

The purpose of the criticality review is to ensure that the spent fuel handling, packaging, transfer, and storage system (cask system) remains subcritical under normal, off-normal, and accident conditions.

II. AREAS OF REVIEW

1. "Criticality Design Criteria and Design Features"

2. "Fuel Specification"

3. "Model Specification"

   a. "Configuration"
   b. "Material Properties"

4. "Criticality Analyses"

   a. "Computer Programs"
   b. "Multiplication Factor"
   c. "Benchmark Comparisons"

5. "Supplemental Information"

III. REGULATORY REQUIREMENTS

The cask system must be designed to be subcritical under all credible conditions. Regulations specific to nuclear criticality safety of the cask system are in 10 CFR Part 72.124 and 72.236(c). Other pertinent regulations include 10 CFR 72.24(c)(3), 72.24(d), and 72.236(g).

IV. ACCEPTANCE CRITERIA

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95 percent confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety must occur before an accidental criticality is possible.

3. When practicable, criticality safety of the design is based on favorable geometry, permanent fixed neutron absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design provides for a positive means to verify their continued efficacy.

4. Criticality safety of the cask system does not rely on:

   a. use of burnup credit,
   b. use of burnable neutron absorbers, and
   c. use of more than 75 percent credit for fixed neutron absorbers, unless comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are implemented.

V. REVIEW PROCEDURES

Review the design features and design criteria in Sections 1 and 2 of the SAR. Review Section 6 for any additional details on design features and design criteria. Assess the bounding specifications for the spent fuel. Examine the models used by the applicant in its criticality analyses. Verify that criticality safety considerations under normal, off-normal, and accident conditions have been addressed. Verify that the cask system design complies with 10 CFR Part 72. Any k_{eff} calculations performed by the reviewer to evaluate that the applicant's design should, to the extent practicable, be performed by independent methods.

1. Criticality Design Criteria and Design Features

Review the principal design criteria presented in Section 2 of the SAR and any additional detail provided in Section 6. Review the general description of the cask presented in Section 1 of the SAR and any additional information provided in Section 6. Verify that the information in Section 6 is consistent with the information in Sections 1 and 2. Also, verify that all drawings, figures, and tables are sufficiently detailed to support an in-depth staff evaluation.

In addition to the general dimensions of the cask components and spacing of fuel assemblies in the basket, the criticality design generally relies on neutron poisons. These poisons may be in the form of fixed poisons in the basket structure and/or soluble poisons in the water of the spent fuel pool. Staff accepts the use of borated water as a means of criticality control if the applicant specifies a minimum boron content, which in turn becomes an operating control and limit in Section 12 of the SAR. These operating controls should be discussed in the staff' SER. If borated water is used for criticality control, administrative controls and/or design features must be implemented to ensure that accidental flooding with unborated water cannot occur, or the criticality evaluation must consider accidental flooding with unborated water. If the cask is also intended for transport, then borated water cannot be relied upon for criticality control.

2. Fuel Specification

Review the specifications for the ranges or types of spent fuel that will be stored in the cask as presented in Sections 1 and 2 of the SAR and any additional information provided in Section 6.

Verify that the spent fuel specifications given in Section 6 are consistent with, or bounded by, the specifications given in Sections 1 and 2. Of primary interest is the maximum enrichment and type of fuel assemblies. For BWR fuel, the maximum fuel pin enrichment should be specified. The criticality calculations for BWR assemblies should use the maximum pin enrichment, or the applicant should show that using an average assembly enrichment is conservative. Although the burnup of the fuel affects its reactivity, Staff does not currently allow credit for burnup, either in depleting the quantity of fissile nuclides or in producing fission product poisons. Specifications for the fuel that will be stored in the cask should be included in Section 12 of the SAR (and SER) and will also be explicitly listed in the Certificate of Compliance.

The fresh fuel assumption must be used in the criticality analyses; therefore, inadvertent loading of the cask with unirradiated fuel is not a major concern. Nevertheless, detailed loading procedures may need to include steps to prevent misloading when fuel exceeding the cask design basis is present in the spent fuel pool. Such details can be addressed only on a site-specific basis and should be before deployment of an actual ISFSI.

Because the cask is typically designed to store many types and configurations of fuel assemblies, the applicant must demonstrate that criticality requirements are satisfied for the most reactive case. A determination of which fuel is bounding in a criticality analysis depends on many factors and usually requires examining several types of fuel assemblies and compositions. The design basis fuel has often been the Westinghouse 17x17 OFA (optimized fuel assembly); however this will not be the case for all cask designs because of cask-specific effects on reactivity. Therefore, the applicant must demonstrate and the reviewer should verify, that the fuel assembly used as the design basis is the most reactive for the specific cask design. Section 12 of the SAR and SER should either clearly indicate the design-basis assemblies or reference the SAR section in which they are identified.

Determine if the applicant has included any specifications regarding the fuel condition. Storage casks to date have typically not been intended for significantly damaged fuel, that is, fuel rod cladding with defects greater than pinhole leaks and hairline cracks. The criticality analyses have generally specified that any damaged fuel rods must be replaced with dummy rods that can displace an equal amount of water as the original rods . These requirements, if invoked by the applicant, must be included as operating controls and limits and discussed in Chapter 12.

3. Model Specification

a. Configuration

Verify that the model used in the criticality evaluation is adequately described for normal, off-normal, and accident conditions. Coordinate with the structural reviewer to understand any damage that could result from accident or natural phenomena events. In general, these events should not significantly impact the criticality model.

Examine the sketches or figures of the model used for criticality calculations. Verify that the dimensions and materials of the model are consistent with those in the drawings of the actual cask. Differences between the actual cask configuration and the models should be identified, and the models should be shown to be conservative. Substitution of ordinary water for end sections and support structures of the fuel is a common and conservative practice in criticality analysis; however, substitution of borated water is non-conservative. Tolerances for poison material dimensions and/or concentrations must be defined, and the most reactive conditions used in the criticality analysis. The analysis must identify all design conditions that are important and then address these conditions for normal, off-normal, and accident conditions.

Verify that the applicant has considered deviations from nominal design configurations. The evaluation of k_{eff} should not be limited to a model in which all the fuel bundles are neatly centered in each basket compartment with the center line of the basket coincident with the center line of the cask. For example, a cask with steel confinement and lead shielding may have a higher k_{eff} when the basket and fuel assemblies are positioned as close as possible to the lead.

In addition to a fully flooded cask, the SAR should also address configurations in which the cask is partially filled with water (borated, if applicable) and the remainder of the cask is filled with steam, consisting of ordinary water at partial density. These configurations are considered to be representative of loading and unloading operations in the spent fuel pool. The SAR should also consider the possibility of preferential or uneven flooding within the cask. In particular, watch for situations where there is water in the fuel regions but not in the flux traps. The analysis must demonstrate that the cask remains subcritical for all credible conditions of moderation.

Examine whether the applicant has prepared a heterogeneous model of each fuel rod or has homogenized the entire fuel assembly. With current computational capability, homogenization is now an uncommon practice and should generally be avoided. If such homogenization is used, however, the applicant must clearly demonstrate that it has been treated conservatively. As a minimum, the applicant should calculate the k_{eff} of one assembly and of several critical benchmark experiments (see Section 5(d)(iii)) using both homogeneous and heterogeneous models. The applicant should then compare the results of these calculations.

b. Material Properties

Verify that the material compositions and densities are provided for all materials used in the calculational model. For nonstandard materials (e.g., fuel and poisons) the source of the data must be referenced. Ensure that validation of the poison concentration is addressed in acceptance testing in Section 9 of the SAR. Many criticality codes will allow the densities to be input directly in g/cm^3. If input is in units of atoms/barn-cm, pay particular attention to the conversion.

Part 72 requires that when solid neutron absorbing materials are used, a positive means to verify their continued efficacy must be provided. Continued efficacy can be demonstrated by requiring acceptance testing of the poisons during fabrication (specified in Section 9 of the SAR), by showing that the small neutron flux from spontaneous fission and subcritical multiplication results in a negligible depletion of poison material over the storage period, and by assessing the structural integrity and potential for material degradation during storage. If continued efficacy can be demonstrated by design and material properties, then a surveillance or monitoring program to "verify" continued efficacy of solid neutron absorbers may not be necessary.

Examine the applicant's choice of cross-sections and determine that an appropriate set has been used. Cross-sections may be distributed with the criticality computer codes or developed independently from another source. For multigroup calculations, the spectrum of the neutron flux used to construct the group cross-sections must be similar to that of the cask. Additional concerns with multigroup calculations are presented in the next section.

4. Criticality Analysis

a. Computer Programs

Both Monte Carlo and deterministic computer codes may be used for criticality calculations. Monte Carlo codes are generally more suited to three-dimensional geometry and, therefore, more widely used to evaluate spent fuel cask designs. The two most frequently used Monte Carlo codes are SCALE/KENO[1] and MCNP[2]. KENO is a multi group code that is part of the SCALE sequence, whereas MCNP permits the use of continuous cross-sections.

If a multigroup treatment is used, ensure that the neutron spectrum of the cask has been appropriately considered. In addition to selecting a cross-section set collapsed with an appropriate flux spectrum, a more detailed processing of the energy-group cross-sections is also required to properly account for resonance absorption and self-shielding. The use of KENO as part of the SCALE sequence will enable such processing directly. Some cross-section sets include data for fissile and fertile nuclides (based on a potential scattering cross-section, s_p) that can be input by the user. If the applicant has used a stand-alone version of KENO, ensure that potential scattering has been properly considered. Furthermore, the "working-format" library, commonly distributed with SCALE/KENO to enable calculations of the code-manual's sample problems, is not intended for criticality calculations of actual systems. In 1991 the staff provided information on cross-section problems to all ISFSI licensees, applicants, and dry storage vendors[3].

For analyses of a cask model with separate regions of water and steam, the use of a multigroup cross-section set raises additional concerns. Verify that the applicant has addressed the differences of the flux spectrum in the two regions. If the results of these calculations indicate that k_{eff} is close to 0.95, additional independent calculations using a different code and/or cross-section library may be helpful. The reviewer should also closely examine the applicant's benchmark analysis, to verify the applicability of critical experiments considered.

b. Multiplication Factor

Examine the results and discussion of the calculations for k_{eff} for the storage cask. Determine if variations in the results caused by different models and sensitivity analyses can be explained and appear reasonable.

For Monte Carlo calculations, assess if the number of neutron histories and convergence criteria are appropriate. As the number of neutron histories increases, the mean value for k_{eff} should approach some fixed value and the standard deviation associated with each mean value should decrease. Depending on the code used by the applicant, a number of diagnostic calculations are generally available to demonstrate adequate convergence and adequate statistical variation. For deterministic codes a convergence limit is often prescribed in the input. The selection of a proper convergence limit and the achievement of this limit must be described and demonstrated.

Because of the importance and complexity of the criticality evaluation, independent calculations should be performed to ensure that the most reactive conditions have been addressed and that the reported k_{eff} is conservative. In deciding how to perform independent calculations, consider the code used by the applicant, the degree of conservatism in the analysis, and the margin of safety in the results. As with any design and review, a small margin of safety and a small degree of conservatism necessitate a more extensive analyses.

As the reviewer, develop a model that is independent of the applicant's. If the reported k_{eff} for the worst case is substantially lower than the acceptance criterion of 0.95, a simple model known to produce very conservative results may be all that is necessary for the independent calculations.

If possible and appropriate, perform the independent calculations with a computer code different from that used by the applicant. Likewise, the use of a different cross-section set can provide a more independent confirmation.

c. Benchmark Comparisons

Computer codes for criticality calculations must be benchmarked against critical experiments. A thorough comparison provides justification for the validity of the computer code, its use on a specific hardware configuration, the neutron cross-sections used in the analysis, and consistency in modeling by the analyst (using the benchmark results for calculations performed by another analyst does not address this last issue). The calculated k_{eff} of the cask must then be adjusted to include the appropriate bias and uncertainties from the benchmark calculations.

Examine the general description of the benchmark comparisons. Verify that the analysis of the experiments used the same computer code, hardware, and cross-section data as those used to calculate the k_{eff} values for the cask.

The reviewer should closely examine the applicant's benchmark analysis to determine that the benchmark experiments are relevant to the actual cask design. No critical benchmark experiment will precisely match the fissile material, moderation, neutron poisoning, and configuration in the actual cask. However, the applicant can perform a proper benchmark analysis by selecting experiments that adequately represent cask and fuel features and parameters that are important to reactivity. Key features and parameters that should be considered in selecting appropriate critical experiments include type of fuel, enrichment, H/U ratio (rod diameter and pitch), reflector, neutron energy spectrum, and poisoning. The applicant must justify, and the reviewer verify, the suitability of the critical experiments chosen to benchmark the criticality code and calculations. UCID- 21830[4] provides information on benchmark experiments that may be applicable to the cask being analyzed.

Detailed guidance on determining a code bias from benchmark experiments has not been formalized. The reviewer needs to assess if a sufficient number of appropriate benchmark experiments have been analyzed and how the results of these benchmark calculations have been converted to a bias for the cask calculations. Simply performing an average of the biases from a number of benchmark calculations is typically not sufficient, particularly if one benchmark yields results that are significantly different from the others. In addition, benchmark comparisons must be checked for bias trends with respect to parameter variations (such as pitch-to-rod-diameter ratio, assembly separation, reflector material, neutron absorber material, etc.). Ref. 4 provides some guidance, but other methods have also been considered appropriate.

For Monte Carlo codes, the statistical uncertainties of both benchmark and cask calculations also need to be addressed. The uncertainties should be applied to at least the 95 percent confidence level. As a general rule, if the acceptability of the result depends on these rather small differences, the reviewer should question the overall degree of conservatism of the calculations. Considering the current availability of computer resources, a sufficient number of neutron histories can readily be used so that the treatment of these uncertainties should not significantly affect the results.

Verify that only biases that increase k_{eff} have been applied. For example, if the benchmark calculation for a critical experiment results in a neutron multiplication that is greater than unity, it should not be used in a manner that would reduce the k_{eff} calculated for the cask. Only corrections that increase k_{eff} should be applied to preserve conservatism.

The reviewer may have already performed a number of benchmark calculations applicable to storage casks and will have a reasonable estimation of the bias to be applied to his independent calculation of the cask. If such is not the case, or if the acceptability depends on small bias differences, the reviewer again needs to determine if sufficient conservatism has been applied to the calculations.

5. Supplemental Information

Ensure that all supportive information or documentation is provided. This would include, but not be limited to, justification of assumptions or analytical procedures, test results, photographs, computer program descriptions, input/output, and applicable pages from referenced documents. The reviewer should request any additional information needed to complete the review.

VI. FINDING EVALUATIONS

Review the 10 CFR Part 72 acceptance criteria and provide a summary statement for each similar to the following:

Criticality structures, systems, and components important to safety are described in sufficient detail in Sections _____ of the SAR to enable an evaluation of their effectiveness.

The _____ cask and its spent fuel transfer systems are designed to be subcritical under all credible conditions.

The criticality design is based on favorable geometry, fixed neutron poisons, and soluble poisons of the spent fuel pool [as applicable]. Fixed neutron poisons have been shown to remain effective for the 20-year storage period and an exemption to 10 CFR Part 72.124(b) for the requirement to provide a positive means to verify their continued efficacy is recommended [if applicable].

The analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for a minimum of 20 years with an adequate margin of safety.

The staff concludes that the criticality design features for the [cask designation] are in compliance with 10 CFR Part 72, as exempted [if applicable], and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the [cask designation] will enable safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission NUREG/CR-0200, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," Vol. 1–4, Rev. 4, April 1995.

2. Los Alamos National Laboratory, "MCNP 4A, Monte Carlo N-Particle Transport Code System," December 1993.

3. U.S. Nuclear Regulatory Commission Information Notice No. 91-26, "Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set Provided with the Keno and Scale Codes," April 15, 1991.

4. Lloyd, W. R., "Determination and Application of Bias Values in the Criticality Evaluation of Storage Cask Designs," UCID-21830, Lawrence Livermore National Laboratory, January 1990. (This report contains a substantial bibliography of numerous benchmark experiments and validation testing.)

7.0 CONFINEMENT

I. REVIEW OBJECTIVE

To ensure that radiological releases to the environment are within the limits established by the regulations and to ensure that the spent fuel cladding and fuel assemblies are sufficiently protected during storage against degradation that leads to gross ruptures.

II. AREAS OF REVIEW

This section evaluates the design and analysis of the confinement system of the cask for normal, off-normal, and accident conditions. Design features and design criteria for the confinement, initially presented in SAR Chapters 1 and 2, are evaluated in more detail. The confinement monitoring capability, if applicable, is assessed. Release of radionuclides, associated with spent fuel, are evaluated by estimating their leakage to the environment and subsequent impact on a real individual beyond the controlled area (i.e., site boundary).

Because the regulatory requirements in 10 CFR Part 72 for doses at and beyond the controlled area boundary include both the direct dose and that from an estimated release of radionuclides to the atmosphere (based on the tested leaktightness of the confinement) , an overall assessment of the compliance with these regulatory limits is deferred until Section 10 "Radiation Protection" of the SER.

The performance of the cask confinement system under accident conditions, as evaluated in this section, is also addressed, if appropriate in the overall accident analyses presented in Section 11.

Review Procedures are presented for the following areas:

1. "Confinement Design Characteristics"

   a. "Design Criteria"
   b. "Design Features"

2. "Confinement Monitoring Capability"

3. "Nuclides with Potential for Release"

4. "Confinement Analyses"

   a. "Normal-conditions"
   b. "Leakage of One Seal"
   c. "Accident Conditions/Natural Phenomena Events"

5. "Supplemental Information"

III REGULATORY REQUIREMENTS

1. Description of Structures, Systems, and Components Important to Safety

Confinement structures, systems, and components important to safety must be described in sufficient detail to enable an evaluation of their effectiveness. [10 CFR Part 72.24(c)(3); 10 CFR Part 72.24(l)]

2. Protection of Spent Fuel Cladding

The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. [10 CFR Part 72.122(h)(1)]

3. Redundant Sealing

The cask must be designed to provide redundant sealing of confinement systems. [10 CFR Part 72.236(e)]

4. Monitoring of Confinement System

Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [10 CFR Part 72.122(h)(4); 10 CFR Part 72.128(a)(1)]

5. Instrumentation

Instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal and off-normal operation. Those control systems that must remain operational under accident conditions must be identified. [10 CFR Part 72.122(i)]

6. Release of Nuclides to the Environment

The quantity of radionuclides expected to be released annually to the environment must be estimated. [10 CFR Part 72.24(l)(1)]

7. Evaluation of Confinement System

The cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR Part 72.236(l); 10 CFR Part 72.24(d)]

Structures, systems, and components important to safety must be designed to withstand the effects of credible accidents and severe natural phenomena without impairing their capability to perform safety functions. [10 CFR Part 72.122(b)]

8. Annual Dose Limit in Effluents and Direct Radiation from An ISFSI

During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ. [10 CFR 72.104(a)]

IV. ACCEPTANCE CRITERIA

Staff generally considers the following criteria to be the minimum necessary to meet the confinement requirements of 10 CFR Part 72:

1. The cask must provide redundant sealing of the confinement system.

2. The applicant's confinement design must be consistent with the "General Design Criteria" reviewed in Section 2 and the regulatory requirements. Generally, the primary confinement boundary and its redundant sealing systems have been designed to conform with Section III, Subsection NB, of the ASME Boiler and Pressure Vessel Code[1]. The staff has relied upon Section III to supply the minimum acceptable margin of safety; therefore, any deviations from Section III must be fully and completely documented and justified. However, exceptions to this requirement have been accepted. For some casks (e.g., those using nodular cast iron) Section III is not applicable. Any exceptions must be fully justified and thoroughly documented.

3. The maximum allowed leakage rates of the total primary confinement boundary and redundant seals must be specified. This will often be displayed in tabular form, including the leakage rate of each seal. Analysis of leakage should be consistent with the principles specified in

ANSI N14.5[2]. Generally the allowable leak rate is not limited by a release of radionuclides, but rather by a sufficiently low leak rate that has reasonable assurance of maintaining the necessary inert atmosphere within the cask.

4. Monitoring capability and/or surveillance plans should be provided for mechanical closure seals. If the closures are welded, the staff has previously accepted that no closure monitoring system is required. This practice is consistent with the fact that other welded joints in the confinement system are not monitored. However, the lack of a closure monitoring system has been typically coupled with a periodic surveillance program that would permit the licensee to take timely and appropriate corrective actions to maintain safe storage conditions after closure degradation.

5. The cask must provide a non-reactive environment to protect fuel assemblies against degradation of the fuel cladding, leading to gross rupture. For dry storage conditions, experimental data have not demonstrated an acceptably low oxidation rate for UO_2 spent fuel, over the 20-year licensing period, to permit safe storage in an air atmosphere. Therefore, to reduce the potential for fuel oxidations and subsequent cladding failure, an inert atmosphere (e.g., helium cover gas) has been used for storing UO_2 spent fuel in a dry environment. Other fuel types, such as graphite fuels for the High Temperature Gas-Cooled Reactors (HTGRs), may not exhibit the same oxidation reactions as UO_2 fuels and therefore may not require an inert atmosphere. Applicants proposing to use non-inert gas atmospheres should discuss how the fuel and cladding will be protected from oxidation.

V. REVIEW PROCEDURES

1. Confinement Design Characteristics

a. Design Criteria

Review the principal design criteria presented in Section 2 of the SAR and any additional detail provided in Section 7.

b. Design Features

Review the general description of the cask presented in Section 1 of the SAR and any additional information provided in Section 7. All drawings, figures, and tables describing confinement features must be sufficiently detailed to stand alone.

Verify that the confinement boundaries are clearly identified. This identification includes the primary confinement vessel, its penetrations, seals, welds, and closure devices, and corresponding information on the redundant sealing.

Verify that the initial leak testing and the confinement monitoring system or surveillance program, if applicable, is adequately described. A more detailed review of this system will be performed in Section 7.5.2 below.

Coordinate with the structural review to ensure that proper specifications for all welds have been provided and, if applicable, that the bolt torque for closure devices is adequate.

If applicable, assess the seals used to provide closure. Because of the performance requirements over the 20-year license period, evaluate the potential for deterioration. Staff has previously accepted only metallic seals for the primary confinement. Coordinate with the thermal review to ensure that the operational temperature range for the seals, specified by the manufacturer, will not be exceeded.

2. Confinement Monitoring Capability

For casks with mechanical closures, the staff has found that, to adequately demonstrate that seals can continue to function and maintain a helium atmosphere in the cask for the 20- year approval period, a seal monitoring system is needed. A seal monitoring system combined with periodic surveillance enables the licensee to determine when corrective action needs to be taken to maintain safe storage conditions. Casks that are closed entirely by welding have been found not to need seal monitoring. Although the details of the monitoring system vary, the general design approach has been to pressurize the region between the redundant seals to a pressure greater than that of the cask cavity. A decrease in pressure between these seals indicates that the gas is leaking either into the cask cavity or out to the atmosphere. In neither case is radioactive gas able to leak to the atmosphere. Hence a faulty seal can be detected with no radiological consequence. In some design that may not require an inert atmosphere in the cask, a periodic surveillance program to check seal leak tightness may be justified.

The monitoring system is generally not a system important to safety . Although its function is to monitor confinement seal integrity, failure of the monitoring system itself does not result in a release of radioactive material. However, the monitoring system should be designed so that its failure can be readily identified during routine surveillance. Staff reviews the monitoring system to assess its ability to fulfill its intended function and whether its failure would result in a degradation of the safety systems. Although the monitoring system does not need to remain functional during an accident, monitoring capability must be restored after the accident. Corrective action necessary to resume monitoring should be addressed in Section 11.

Examine the specified pressure of the gas in the monitored region to determine whether it is higher than both the cask cavity and the atmosphere. Coordinate with the structural and thermal review to verify the pressure in the cask cavity. Confirm that the set point for the monitoring system will provide sufficient time to allow implementation of corrective action before release of any gas from the cavity to the environment.

Review the applicant's analysis to verify that normal seal leakage will not cause all the gas in the monitoring system to escape over the lifetime of the cask. From a standpoint of release of fission product gases to the atmosphere, a leakage rate of the seals on the order of 10^{-5} std cc/s is often shown to be acceptable even under very conservative assumptions. This leakage rate is then specified as an acceptance test criterion in Section 9 of the SAR, even though the actual leakage rate of the seals is expected to be significantly lower. Staff has accepted reasonable assumptions on the seal leakage rates in the evaluation of the pressure monitoring system, to provide a balance of safety and design margin. If the gas used in the pressure monitoring system leaks at a rate greater than assumed in the analysis, the only consequence should be that the pressure monitoring system must be re-pressurized, since routine surveillance should identify any reduction in pressure before it drops below the cask cavity pressure.

Assess the monitoring system to ensure that its presence does not introduce additional safety concerns. If components of the pressure monitoring system also serve as part of the redundant sealing system, then the monitoring system must meet the same design requirements as the sealing itself. Coordinate as necessary with the structural review.

For redundant seal welded closures, the applicant should provide adequate justification that the seal welds are sufficiently tested and inspected to ensure that the weld will behave similarly to the adjacent parent material of the cask. Any inert gas should not leak or defuse through the weld and cask material in excess of the design leak rate.

Verify that any leak test, monitoring, or surveillance conditions are appropriately specified in Section 9, Section 11, the license, or the Certificate of Compliance.

3. Nuclides with Potential for Release

Staff has determined that, as a minimum, the nuclides shown below must be analyzed for potential release. The percentages indicated account for the fact that some of these nuclides will be trapped in the fuel matrix or exist in a chemical or physical form that is not capable of release to the controlled area boundary under credible conditions.

Nuclide	Fraction Available for Release*
^{3}H	0.30
^{85}Kr	0.30
^{129}I	0.10
^{137}Cs	$5x10^{-10}$
^{134}Cs	$5x10^{-10}$
^{90}Sr	$5x10^{-10}$
^{106}Ru	$5x10^{-10}$
^{60}Co **	$5x10^{-10}$

*From a failed fuel rod. Except for ^{60}Co, only failed fuel rods contribute significantly to the release. Total fraction of radionuclides available for release must be multiplied by the fraction of fuel rods assumed to have failed.

**Source of ^{60}Co is crud on fuel rods, estimated to be 140 $\mu Ci/cm^2$. Total ^{60}Co activity is this estimate times the total surface area of all rods in the cask[3].

The quantities of these radioactive nuclides are often presented in Section 5 of the SAR, since they are generally determined during the evaluation of the gamma and neutron source terms in the shielding analysis. Coordinate with the shielding review to verify that these nuclides have been adequately determined.

4. Confinement Analysis

Review the applicant's confinement analysis and the resulting annual dose at the controlled area boundary. In general, the staff evaluates analyses for three specific scenarios:

a. Normal-conditions

If the confinement boundary is welded or if the region between the two seals is monitored, the staff accepts that no discernible undetected leakage is credible. Hence, the dose at the controlled area boundary from atmospheric release is negligible.

b. Leakage of One Seal

This scenario is of interest because it provides a basis for corrective action that needs to be taken in case the monitoring system alarms. By examining the dose at the controlled area boundary resulting from a seal failure, the applicant can determine the time at which corrective action must be initiated. In addition, the dose can be used to evaluate the degree of inspection or calibration needed for the monitoring system.

Staff has accepted this analysis based on the assumption that 3 percent to 10 percent of the fuel rods have failed. Current practice is to assume 10 percent unless the applicant provides sufficient justification for considering a lesser figure. Coordinate with the structural and thermal review to verify that the cask cavity pressure applicable for this condition has been adequately determined.

In addition to the quantity of nuclides available for release and the pressure of the cask cavity, the dose at the controlled area boundary will depend on the seal leakage rate, the distance from the cask to the controlled area boundary, the atmospheric dispersion factor, an individual's breathing rate (except for Kr), and dose conversion factors. Maximum allowable seal leakage rates are specified as design criteria and should be discussed in Chapter 12. The minimum distance between the casks and the controlled area boundary is generally also a design criterion; however, 10 CFR Part 72 requires it to be at least 100 meters from the ISFSI. Because a release resulting from a seal failure will occur over a substantial period of time, the staff has previously accepted, as a bounding condition, the atmospheric dispersion factors of Regulatory Guide 1.145[4], based on F-stability diffusion, a wind speed of 1 m/s, and plume meandering. Also, the staff has accepted either an adult breathing rate of 2.5×10^{-4} m^3/s, as specified in Regulatory Guide 1.109[5], or a worker breathing rate of 3.3×10^{-4} m^3/s, as specified in EPA Guide No. 1[6]. Dose conversion factors for inhalation, whole body dose, and thyroid dose should be equivalent to those indicated in EPA Guide No. 11.

Review the applicant's controlled area boundary dose calculation. Verify that the applicant has determined both the whole body dose and the thyroid dose. These doses must be less than those listed in 10 CFR Part 72.104(a), which also considers other contributions to the total dose (e.g., direct dose). For conservatism, the staff often considers that an individual could be present at the controlled area boundary for the full year (8760 hours). The dose that an individual would receive in case of a seal leak is usually very small, and this conservatism has historically posed no difficulties in meeting the regulatory limits; however, if sufficient justification is provided, this criterion may be reviewed .

c. Accident Conditions/Natural Phenomena Events

Coordinate with the structural review to determine the effect of specific accident conditions and natural phenomena events on the cask confinement system. Because 10 CFR Part 72.122(b)(2) requires that structures, systems, and components important to safety withstand the effects of design bases conditions without impairing their safety function, the staff requires that the confinement system remain intact under these events. No credible scenario should have been identified that would cause failure of the confinement system. Nevertheless, to demonstrate the overall safety of dry spent fuel storage, the staff has conservatively assumed an instantaneous leak with 100 percent of the fuel rods failed for calculation of an accident dose to an individual located at the boundary of the controlled area.

The analysis for the above scenario is similar to that for failure of one seal. In this situation, the cask cavity pressure need not be considered. Because the leak is assumed to be instantaneous, the plume meandering factor of Regulatory Guide 1.145[4] is not typically applied. This is equivalent to using an atmospheric dispersion factor be based on Regulatory Guide 1.25[7]. Hence, this dispersion factor is generally found to be 4 times higher than that for the case of a single seal failure.

Review the applicant's calculation for the dose at the controlled area boundary. The regulatory limits are those listed in 10 CFR Part 72.106(b). Verify that the applicant has determined both the whole body dose and the thyroid dose. Note that for an instantaneous release (and instantaneous exposure) the time that an individual remains at the controlled area boundary is not a factor in the dose calculation.

5. Supplemental Information

Ensure that all supportive information or documentation is provided or readily available. This would include, but not be limited to, justification of assumptions or analytical procedures, test results, photographs, computer program descriptions, input/output, and applicable pages from referenced documents. The reviewer should request any additional information needed to complete the review.

VI. EVALUATION FINDINGS

Review the 10 CFR Part 72 acceptance criteria and provide a summary statement for each similar to the following:

Confinement structures, systems, and components important to safety are described in sufficient detail in Sections _____ of the SAR to enable an evaluation of their effectiveness.

The spent fuel cladding is protected against degradation that leads to gross ruptures by emplacing it in a basket structure and using ______ as a non-reactive cover gas. Temperature considerations are discussed in Section 4 of the SER.

A redundant sealing of the confinement system is provided by ______.

The confinement system is monitored with a ________ monitoring system as discussed above (if applicable).

No instrumentation is required to remain operational under accident conditions. If the monitoring system remains operational or can be returned to an operational condition after an accident, it will enable an assessment that confinement is maintained.

The quantity of radioactive nuclides expected to be released to the environment has been assessed and is discussed above. In Section 10 of the SER, the dose from these releases will be added to the direct dose to show that 10 CFR Part 72.104(a) and 10 CFR Part 72.106(b) are satisfied.

The cask confinement system has been evaluated [by appropriate tests or by other means acceptable to the Commission] to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

The staff concludes that the design of the confinement system of the [cask designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the [cask designation] will enable safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

VII. REFERENCES

1. The American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code."

2. Institute for Nuclear Materials Management, ANSI N14.5, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," 1987.

3. Sandoval, Robert P., et al, "Estimate of CRUD Contribution to Shipping Cask Containment Requirements," Sandia Report, SAND88-1358, TTC-0811, UC-71, Sandia National Laboratories, January 1991

4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145, "Atmospheric Dispersement Models for Potential Accident Consequence Assessments at Nuclear Power Plants," February 1989.

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.109, "Calculations of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," October 1977.

6. U.S. Environmental Protection Agency, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," DE89-011065, 1988.

7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 1972.

8.0 OPERATING PROCEDURES

I. OBJECTIVE

The purpose of the operating procedures review is to determine that guidance sufficient for licensees to prepare procedures to load fuel into the cask, transfer the cask to and from the ISFSI, and unload fuel from the cask, has been prepared.

II. AREAS OF REVIEW

This section evaluates the procedures required for cask operations, including:

1. "Loading the Cask"
2. "ISFSI Operations"
3. "Unloading the Cask"

Loading procedures include activities for selecting and placing the fuel into the cask, draining and drying the cask, establishing an inert environment inside the cask, and sealing the cask. ISFSI procedures include transferring the cask to the ISFSI and other operations (e.g., monitoring and surveillance) necessary for safe spent fuel storage. Unloading procedures are necessary to recover from an unforeseen problem during storage or to prepare the fuel for off-site transportation or ultimate disposition. They are generally the reverse sequence of those for loading.

III. REGULATORY REQUIREMENTS

The equipment and processes used to maintain control of radioactive effluents must be described. [10 CFR Part 72.24(l)(2)]

Operating procedures must be developed to protect health and minimize danger to life or property. [10 CFR Part 72.40(a)(5)]

Operational restrictions must be established to meet 10 CFR Part 20 limits and as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI operations. [10 CFR Part 72.104(b); 10 CFR Part 72.24(e)]

Structures, systems, and components that are important to safety must be designed to permit inspection, maintenance, and testing. [10 CFR Part 72.122(f)]

Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. [10 CFR Part 72.122(l)]

The cask must be designed to minimize the quantity of radioactive waste generated. [10 CFR Part 72.128(a)(5); 10 CFR Part 72.24(f)]

The licensee shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these drawings be followed. The instructions, procedures , and drawings must include appropriate quantitative and qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. [10 CFR Part 72.40(a)(5); 10 CFR Part 72.150; and 10 CFR Part 72.212(b)(9)]

{For vendors seeking approval of a cask design} The cask vendor shall ensure that written procedures and appropriate tests are established before the use of the casks. A copy of these procedures and tests must be provided to each cask user. [10 CFR Part 72.234(f)]

The cask must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR Part 72.236(h)]

The cask must be designed to facilitate decontamination to the extent practicable. [10 CFR Part 72.236(i)]

IV. ACCEPTANCE CRITERIA

1. The SAR should outline the major steps recommended for preparing, loading, testing, storing, and unloading the cask. The SAR should also include the major steps recommended for monitoring and maintaining safe storage conditions at an ISFSI. The recommended activities should be achievable and within the capabilities of the cask system design. The unloading procedure recommendations should ensure that stored fuel can be safely and readily retrieved. The level of detail provided in the SAR should provide a suitable basis for developing usable procedures to conduct actual operations.

2. The procedure recommendations should address the applicable operating controls and limits described in Chapter 12 of the SAR.

3. Recommendations should include provisions to ensure that: 1) occupational radiation exposures will remain as low as reasonably achievable (ALARA), 2) effective measures are taken to preclude potential unplanned and uncontrolled releases of radioactive materials, and 3) offsite dose rates will be maintained within the limits of 10 CFR Part 20 and 10 CFR 72.104 for normal operations, and 10 CFR 72.106, for accident conditions.

V. REVIEW PROCEDURES

NUREG/CR-4775[1] provides guidance on preparing operating procedures for shipping packages. Although this document addresses 10 CFR Part 71[2] specifically, most of the guidance can be adapted for storage casks under 10 CFR Part 72. The reviewer should consequently be familiar with this information before initiating the review.

In addition, SAR Chapter 12, "Operating Controls and Limits," specifies conditions that must be followed during operations. Coordinate with the review in Section 12 and verify that the proposed operating procedures are compatible with and incorporate the applicable operating limits and controls.

The SAR usually presents general operating sequences, guidance to procedure writers, and generic procedures. More detailed procedures are developed in operating manuals provided with the cask. Based on these operating manuals, detailed site-specific procedures are implemented by the ISFSI licensee.

The operating procedure sequences are described in Chapter 8 of the SAR, and the direct dose rate information evaluated in Section 5 are used to assess compliance with radiation protection requirements in Section 10.

The review process described below is specified in a format for a single review. Although one individual or group may be tasked with preparing the SER section on operating procedures, all review team members should examine the operating procedures presented in the SAR.

This review procedure is based on the assumption that the ISFSI is located at a reactor facility licensed under 10 CFR Part 50 and that loading and unloading activities are performed in the facility's spent fuel pool. Review procedures for dry fuel transfers and/or ISFSI operations at sites away from a reactor will be developed in the future.

1. Loading the Cask

The procedure should present the activities sequentially in the anticipated order of performance. Review the generic procedures discussed in Chapter 8 of the SAR to ensure that appropriate key prerequisite, preparation, and receipt inspection activities, to be accomplished before cask loading, are included. Verify that key tests, inspections, verifications and cleaning procedures to be accomplished during cask loading are specified. Where applicable, verify that the procedure guidance speaks to actions to ensure that any fluids, such as shield water and primary coolants, fill their respective cavities according to design specifications.

Review the spent fuel specifications (e.g., burnup, cooling period, source terms, heat generation, cladding damage, etc.) in Chapters 2 and 12 of the SAR, and verify that the loading procedure guidance addresses them appropriately. Depending on the types and specifications of fuel assemblies stored in the reactor spent fuel pool, detailed site-specific procedures will be necessary to ensure that all fuel loaded in the cask meets the fuel specifications for the cask design. These procedures can be evaluated only on a site-specific basis. The SAR should indicate, however, that such procedures may be necessary.

Verify that the procedure recommendations incorporate ALARA principles and practices. These may include provisions to perform radiological surveys, exposure and contamination control measures, temporary shielding, and suggested caution statements related to specific actions that could change radiological conditions. Verify that any recommended surveys incorporate the applicable operating controls and limits described in Chapter 12 of the SAR.

Where applicable, verify that methods to minimize offsite releases such as filtered ventilation, temporary containments (tents), etc. are described. The procedures should also provide for minimizing generation of radioactive waste.

A general listing of the major tools and equipment needed to support operations should be provided. Verify that specialized tools and equipment are described in sufficient detail and that their use is understood. Examples include lifting yokes, drain and vent hoses, and vacuum drying equipment. Assess if any such equipment is important to safety and compare this equipment with that identified in Section 2.

Evaluate the methods used to drain and dry the cask to determine if the acceptable level of moisture is clearly described and achievable. The staff accepts impurity levels comparable to those discussed in PNL-6365[3], which evaluates the effect of oxidizing impurities on the order of 0.25 volume percent. Of particular concern is the possible freezing of water vapor during the vacuum drying operations. Ensure that sound methods and sufficient time are provided to detect the formation of water vapor during vacuum operations. Also ensure that vacuum drying specifications incorporate the applicable operating controls and limits specified in Chapter 12 of the SAR.

Verify that the procedure recommendations address steps to fill and pressurize the cask with inert gas and that the requirements of SAR Chapter 12 are included.

Ensure that leak rate criteria (e.g., total leakage, leakage per closure, sensitivities of tests) are specified and that these criteria are consistent with those presented in Sections 2, 9, and 12. Assess if the general methods of leak testing (e.g., pressure rise, mass spectrometry) are applicable to the leak rate being tested. Pay particular attention to use of quick-disconnect fittings that may be used for draining and filling operations. Although no credit is usually taken for these devices as part of the confinement boundary, their presence can negate the results of the leak test and guidance regarding their use should be provided. Guidelines should note that leak testing should be in accordance with ANSI N14.5[4].

Verify that the recommendations for tightening the closure bolts or welding the closures are consistent with information presented in SAR Chapters 2, 3, 7, and 9, as applicable and that specific values are stated where applicable. For bolted connections, the torque value, torque sequence, minimum number of torque sequences necessary to reach the torque value, and lubricants should be recommended. For welded closures, verify that an acceptable volumetric non-destructive examination (NDE) method is specified, along with a discussion of other applicable examination methods.

2. ISFSI Operations

Examine the recommendations associated with procedures necessary to transfer the cask to the ISFSI. Particular attention should be devoted to ensuring that all accident events applicable to such transfer are bounded by the design events analyzed in SAR Chapters 2 and 11. Coordinate with the structural and thermal reviews to ensure that all conditions for lifting and handling methods are bounded by the evaluations in Sections 3 and 4.

Review the procedure recommendations to verify that they include a discussion of the inspection, surveillance, and maintenance requirements that are applicable during ISFSI storage. These should also be included in Section 12 as appropriate. Coordinate with the other reviews to ensure that requirements addressed in other sections of the SAR have been included in the operating procedures. Note that if the confinement vessel closure is bolted, the staff generally requires that the successful operation of the seals be demonstrated with an initial leak test and a monitoring system and/or a surveillance program, as discussed in Section 7.

3. Unloading the Cask

Verify that unloading procedure recommendations are adequately described in the SAR. In general, these procedures are the reverse of those for loading. The unloading procedures should contain the applicable attributes described above for review of the loading procedures.

Appropriate steps for sampling the atmosphere inside the cask before venting and opening a cask should be included. An overview of the general approach to preclude the uncontrolled release of any radioactive material during cask venting should be provided.

Special attention should be devoted to cooling, venting, and reflood requirements, since the fuel may reach a significant temperature under storage conditions. Appropriate steps may be necessary to control and limit internal cask pressure from steam flashing to ensure that the fuel and cask components are not subjected to an unacceptable thermal shock or overly stressed such that fuel retrieval is difficult. Where applicable, coordinate the review of the analysis performed to support the cooldown process with the thermal and structural reviews. Cautions and guidance regarding conduct of these activities should be provided.

If unloading requires any specialized equipment, the equipment should be clearly specified and discussed in the procedural guidance.

VI. EVALUATION FINDINGS

Review the 10 CFR Part 72 acceptance criteria and provide a summary statement for each similar to the following, as applicable:

The [cask designation] is compatible with [wet/dry] loading and unloading. General procedures for these operations are summarized in Chapters _____ of the SAR. Detailed procedures will need to be developed on a site-specific basis.

The [bolted lids/other features] of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.

The smooth surface [or other feature] of the cask is designed to facilitate decontamination. Only routine decontamination will be necessary after the cask is removed from the spent fuel pool.

No significant radioactive waste is generated during storage. Contaminated water from the spent fuel pool will be governed under the 10 CFR Part 50[5] license conditions. [If applicable]

No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading will be governed by the Part 50 license conditions. [If applicable]

General operating procedures are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed on a site-specific basis.

Operational restrictions to meet 10 CFR Part 20[6] limits are evaluated in Section 10 of the SER. Additional site-specific restrictions may also be established by the site licensee.

The staff concludes that the operating procedures and guidance for the operation of the [cask designation] are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedures provides reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

VII. REFERENCES

1. Witte, M. C., "Guide for Preparing Operating Procedures for Shipping Packages," UCID-20820, NUREG/CR-4775, Lawrence Livermore National Laboratory, July 1988.

2. U.S. Code of Federal Regulations, Title 10 Part 71, "Packaging and Transportation of Radioactive Material."

3. Knoll, R.W. et al., "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, DE88 003983, Pacific Northwest Laboratory,

November 1987.

4. Institute for Nuclear Materials Management, ANSI N14.5-1987, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," January 1987.

5. U.S. Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities."

6. U.S. Code of Federal Regulations, Title 10, Part 20, "Standards for Protection against Radiation."

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN FOR STORAGE CASKS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

I. REVIEW OBJECTIVE

The purpose of this review is to ensure that the appropriate acceptance tests and maintenance programs are included in the SAR for the spent fuel storage cask system. Also, a clear specific listing of the commitments will help avoid ambiguities about design, fabrication, and operational testing requirements when inspections are conducted.

II. AREAS OF REVIEW

The acceptance tests demonstrate that the cask has been fabricated in accordance with the design criteria and that the initial operation of the cask complies with regulatory requirements. The maintenance program describes those actions that need to be implemented during the storage period to ensure that the cask continues to perform its intended functions.

The review process described below is specified in a format for a single review. Although one individual or group may be tasked with preparing the SER section on acceptance tests and maintenance program, all review team members should examine the information presented in this section of the SAR. Special attention should be devoted to those tests (or the absence of tests) that affect their functional area of review. The following acceptance tests and maintenance programs are discussed herein:

1. "Acceptance Tests"

    a. "Visual and Non-Destructive Examination Inspections"
    b. "Structural/Pressure Tests"
    c. "Leak Tests"
    d. "Shielding Tests"
    e. "Neutron Absorber Tests"
    f. "Thermal Tests"
    g. "Cask Identification"

2. "Maintenance Program"

    a. "Inspection"
    b. "Tests"

c. "Repair/Replacement/Maintenance"

III. REGULATORY REQUIREMENTS

1. Testing and Maintenance

The program covering preoperational testing and initial operations must be described. [10 CFR Part 72.24(p)]

The cask must be designed to permit maintenance as required. [10 CFR Part 72.236(g)]

Structures, systems, and components important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR Part 72.122(a); 10 CFR Part 72.122(f); 10 CFR Part 72.128(a)(1); 10 CFR Part 72.24(c)]

A test program must be established to ensure that all required testing is performed to meet the requirements and acceptance criteria[a].

The cask and its systems important to safety must be evaluated by appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR Part 72.236(l)]

The cask must be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness. [10 CFR Part 72.236(j)]

The applicant must perform, and make provisions that permit the Commission to perform, tests that the Commission deems necessary or appropriate. [10 CFR Part 72.232(b)]

The general licensee must accurately maintain the record provided by the cask supplier for each cask showing any maintenance performed on the cask, including evidence that any maintenance and testing have been conducted under an NRC approved Quality Assurance program. [10 CFR Part 72.212(b)(8)]

2. Resolution of Adequacy or Reliability

Structures, systems, or components important to safety whose functional adequacy or reliability

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[a] A report of the pre-operational test acceptance criteria and test results must be submitted to NRC at least 30 days before the receipt of spent fuel. [10 CFR Part 72.162; 10 CFR Part 72.82(e)]

have not been demonstrated by prior use for that purpose, or which cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, must be identified. A schedule should also be included showing how safety questions will be resolved before the initial receipt of spent fuel. [10 CFR Part 72.24(i)]

3. Cask Identification

The cask must be conspicuously and durably marked with a model number, unique identification number, and an empty weight. [10 CFR Part 72.236(k)]

IV. ACCEPTANCE CRITERIA

The staff has previously accepted the following codes and standards for design, fabrication, inspection, and testing criteria for the review of dry cask storage components:

The confinement system - ASME Section III, Subsection NB. Any exception to these standards should be clearly identified.

Non-confinement system

- Internals of confinement (e.g., basket), ASME Section III, Subsection NG
- Metal cask overpack - ASME Section VIII
- Concrete cask overpack- ACI 349 and ACI 318, if appropriate
- Other remaining metal structures- ASME Section III, Subsection NF or AISC Manual of Steel Construction

Any exception to these standards should be clearly identified.

ANSI/ANS-57.9-1984 is endorsed by NRC via Regulatory Guide 3.60, "Design of an ISFSI (Dry Storage)," with exceptions stated as an acceptable method for design of a dry storage ISFSI.

Leakage tests of the confinement should generally be performed in accordance with ANSI N14.5-1987.

V. REVIEW PROCEDURES

In general, applicants commit to design, construct and test the system under review to the codes and standards committed to in Chapter 2. Specific test programs are generally not reviewed in the licensing process; however staff has recently been required to perform such reviews. The following information is provided as examples of areas that may require special attention from the reviewers. Each test, if described in the SAR, should be reviewed for the following information: (1) purpose of the test; (2) description of test method, including any applicable Standard to which the test will be performed; (3) acceptance criteria and bases for the test; and (4) actions to be taken if acceptance criteria are not satisfied.

1. Acceptance Tests

The following guidance is presented based on tests deemed acceptable by the staff in previous SAR reviews. Alternative tests and criteria may be used if appropriately explained and justified in the SAR.

a. Visual and Non-Destructive Examination Inspections

Cask components should be fabricated and examined in accordance with an accepted standard used for their design (e.g., the ASME Boiler and Pressure Vessel Code[1], Section III or Section VIII). These sections define the examination requirements in the following sections: Section II , ("Materials Specifications and Properties"), Section V ("NDE Specifications and Procedures"), and Section IX ("Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators"). The following guidance assumes that the ASME Code (the Code) is applicable to the cask being reviewed.

The NDE of weldments must be well-characterized on drawings, using standard NDE symbols and/or notations (see AWS A2.4[2]). 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 Part 72, Subpart G. The inspection plan should include visual (VT), dye penetrant (PT), magnetic particle (MT), ultrasonic (UT), and radiographic (RT) examinations, as applicable. It should identify welds that will be examined, the examination sequence, type of examination, and the appropriate acceptance criteria from Ref. 1. Inspection personnel should be pre-qualified in accordance with the current revision of SNT-TC-1A[3], as specified by the Code. All weld NDE should be performed in accordance with written and approved procedures.

Confinement boundary welds, as well as welds for components performing the redundant sealing, must meet the requirements of Section III, NB-5200 ("Required Examination of Welds"). This section 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 Section V, Articles 2, 6, and 7, respectively.

Acceptance criteria for RT should be in accordance with Section III, Subsection NB, Article NB-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 Section 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 Section V, Article 5.

Acceptance criteria should be in accordance with NB-5330, "Ultrasonic Acceptance Standards." Cracks, lack of fusion, or incomplete penetration are unacceptable, regardless of length.

NRC has accepted multiple surface examinations of welds combined with helium leak tests for inspecting the final redundant seal welded closures.

Nonconfinement welds should meet the requirements of Section III, Subsection NF or Section VIII, Division 1, as applicable. The required volumetric examination of welds is either RT or UT, as discussed in Section III, NF-5200, and in Section VIII, UW-11 of the Code. The appropriate specifications of Section V are invoked in Article 2 for RT and in Article 5 for UT. Acceptance standards are provided in Section III, Subsection NF, NF-5320 ("Radiographic Acceptance Standards") for RT, and in NF-5330 ("Ultrasonic Acceptance Standards") for UT. For Section VIII weldments, RT acceptance criteria should be in accordance with Section VIII, Division 1, UW-51, and the repair of unacceptable defects should be in accordance with UW-38. Repaired welds should be reexamined in accordance with the original acceptance criteria.

Nonconfinement welds that cannot be RT examined should undergo UT in accordance with Section V, Article 5. Acceptance criteria should be in accordance with Section 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 rationale for not requiring RT examination of these welds.

Finished surfaces of the cask should be examined visually in accordance with Section V, Article 9, of Reference. 1. The acceptance criteria for VT examined welds should be in accordance with Section VIII, Division 1, UW-35 and UW-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 Section V, Article 6. Acceptance criteria for confinement welds should be in accordance with Section III, Subsection NB, Article NB-5350; repair procedures should be in accordance with NB-4450. Acceptance criteria for nonconfinement welds should be in accordance with Section VIII, Division 1, Appendix 8, or NF-5360 ("Liquid Penetrant Acceptance Standards"); repair procedures should be in accordance with Section 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 performed to prevent post-welding operations (like grinding) from compromising the design requirements (like wall thickness).

b. Structural/Pressure Tests

Lifting trunnions should be fabricated and tested in accordance with ANSI N14.6[4]. The cask is generally considered a critical load, as defined by this Standard, during its handling in the spent fuel pool. Consequently, trunnion testing should be performed at a minimum of 150 percent of the maximum service load, if redundant lifting is employed or at 300 percent of the service load if non-redundant lifting is applicable. Site-specific details of the spent fuel pool and lifting procedures may enable the cask to be considered a non-critical load. Restrictions on cask lifting resulting from these tests should be included in Section 12 of the SAR and SER and the testing values stated explicitly in Chapter 9 of the SAR.

The confinement boundary (including that of the redundant sealing) should be hydrostatically tested to 125 percent of the design pressure, in accordance with ASME B&PV Code, Section III, Article NB-6000. Section III, Article NCA-2142.1, defines the design pressure as applicable to Level A Service Limit. In addition to the ASME Code design requirements, 10 CFR Part 72.122 requires that the cask system be designed to withstand postulated accidents. (As discussed in Section 4 of the SRP, this would be a pressure that is no less than the maximum cask cavity pressure with 100 percent failure of the fuel rods.) Note that the 10 CFR Part 71 requirements for this test are generally 150 percent of the maximum normal operating pressure, which includes the maximum cask cavity pressure with 100 percent failure of the fuel rods. If the cask is be certified under both 10 CFR Parts 71 and 72, the higher of these two values should be used. The test pressure should be maintained for a minimum of 10 minutes, after which a visual inspection should be performed to detect any leakage. All accessible welds shall be PT inspected. Any evidence of cracking or permanent deformation is cause for rejection. The hydrostatic test pressure should be clearly specified in Chapter 9 of the SAR.

Some casks contain a neutron shielding material that will off-gas at higher temperatures. Such material is usually contained inside a thin steel shell to prevent loss of mass and to provide protection from minor accidents and natural phenomena events. Rupture disks are generally provided to prevent catastrophic failure of this shell, which should be tested to at least 125 percent of relief valve pressure. The test pressure for the rupture disk should be clearly specified in the SAR.

Some cask designs use ferritic steels that are subject to brittle fracture failures at low temperatures. Section II, Part A, of ASME B&PV Code contains procedures for testing ferritic steel used in low temperature applications. Based on guidance in NUREG/CR-1815[5], Section 5.1.1, two methods for identifying suitable materials were established. 1) The nil ductility temperature (NDT) must be determined either by direct measurement (ASTM E-208) or indirect measurement (ASTM E-604); the minimum operating temperature of the steel must be specified to be 50° F higher than the NDT. 2) ASME Charpy testing procedures are accepted by the staff for verification that the material's minimum absorbed energy is 15 ft-lb at -50° F for a 1 inch thick section for an average of three specimens, or 12 ft-lb, -50° F minimum for one specimen. Coordinate with the thermal review to ensure that the correct temperatures are chosen for the tests and that the method of testing is specified in the SAR.

c. Leak Tests

Leak tests should be performed on all boundaries relevant to confinement. These include the primary confinement boundary, the boundary of the redundant sealing, and, if applicable, any additional boundaries used in the pressure monitoring system. Leakage criteria in units of std cc/s must be at least as restrictive as those specified in the principal design criteria. The general testing methods (e.g., pressure rise, mass spectrometer) and the required sensitivities should also be indicated. If cask closure depends on more than one seal (e.g., lid, vent port, drain port) the leakage criteria should ensure that the total leakage is within the design requirements. Leak testing should generally be conducted in accordance with ANSI N14.5[6].

d. Shielding Tests

Tests of the effectiveness of both the gamma and neutron shielding may be required if, for example, the cask contains a poured lead shield or a special neutron-absorbing material. Any scanning or probing with an auxiliary source for the purpose of characterizing the shielding should be described.

In addition to the above tests, dose rate measurements should be performed, after the spent fuel is loaded, to establish that design criteria have been satisfied.

e. Neutron-Absorber Tests

Fixed neutron absorbers that ensure subcriticality must be tested to verify the minimum areal density (or other applicable specification) used in the criticality analysis in Section 6 of the SAR.

f. Thermal Tests

Depending on the details of the cask design and the ability to determine the heat removal capability from a thermal analysis, testing may be required to verify the cask performance. Acceptance criteria should be established based on the conditions of the test (e.g., test heat loading, ambient conditions). Correlation of test performance to actual spent fuel loading conditions should be discussed in the SAR in order to avoid ambiguous, unreviewed analysis after the test data have been obtained.

g. Cask Identification

The cask must be marked with a model number, unique identification number, and empty weight. Generally this information will be on a data plate, which should be detailed in one of the drawings included in Section 1 of the SAR.

2. Maintenance Program

Storage casks are typically designed so that minimum maintenance is required. The following areas should be addressed, as applicable.

a. Inspection

Usually the cask has at least one monitoring system (e.g., pressure, temperature, dosimetry). The SAR should discuss how such systems will be used to provide information on a possible off-normal event and what surveillance actions are necessary to ensure that these systems are functioning properly. Detailed procedures will be implemented by the licensee at the site.

Routine periodic visual surface and weld inspections should be described. Also, inspection of lifting and rotating trunnion load-bearing surfaces should be discussed.

b. Tests

Any periodic tests of cask components or calibration of monitoring instrumentation should be described. Any routine testing of support systems (e.g., vacuum drying, helium backfill, and leak testing equipment) should be described. Periodic tests to verify shielding and thermal capabilities should be discussed.

c. Repair/Replacement/Maintenance

Repair and replacement of cask components that may be required during the lifetime of the cask should be discussed. Routine maintenance such as re-application of corrosion inhibiting materials should be described. This discussion should include methods of repair, testing, and acceptance criteria. Such information is also often included in Section 11, which describes the actions to be taken after an off-normal event or accident condition.

VI. EVALUATION FINDINGS

Review the 10 CFR Part 72 acceptance criteria and provide a summary statement for each similar to the following, as applicable:

The program covering preoperational testing and initial operations of the [cask designation] is described in Section _____ of the SAR. The maintenance program is discussed in Section _____.

Structures, systems, and components important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function to be performed. The safety importance of structures, systems, and components is identified in Section _____ of the SAR. Standards for design, fabrication, and testing are presented in Sections _____ of the SAR.

The [cask designation] is inspected and tested to ensure that there are no defects that could significantly reduce its confinement effectiveness. This insp ction and testing is described in

Section _____ of the SAR.

The cask will be marked with a data plate that indicates its model number, unique identification number, and empty weight. The data plate is described in Drawing _____ of the SAR.

The staff concludes that the acceptance tests and maintenance program for the [cask designation] are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

VII. REFERENCES

1. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code," 1992.

2. American Welding Society, AWS A2.4, "Standard Symbols for Welding, Brazing and Nondestructive Examination," 1993.

3. American Society for Nondestructive Testing, Recommended Practice No. SNT-TC-1A, December 1992.

4. Institute for Nuclear Materials Management, ANSI N14.6-1986, "American National Standard for Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kilograms) or More," September 1986.

5. Holman, W.R. and Langland, R.T., "Recommendations for Protecting against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick," NUREG\CR-1815, Lawrence Livermore Laboratory, August 1981.

6. Institute for Nuclear Materials Management, ANSI N14.5-1987, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," January 1987.

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10.0 RADIATION PROTECTION

I. REVIEW OBJECTIVE

This section evaluates the radiation protection aspects to determine whether: (a) those design features of the storage cask system meet NRC's design criteria for direct radiation; (b) the ISFSI facility radiation protection program for occupational exposure to workers is consistent with NRC's radiation protection standards and worker's exposures will be maintained as low as is reasonably achievable (ALARA); and (c) the radiation doses to the general public will meet regulatory standards during both normal operation and accident situations .

The major mode of radiation exposure from spent fuel storage cask handling in an ISFSI operation is from direct radiation. Because of the cask design requirements, it is not expected that radionuclides will be released from the cask during normal operations or design basis accidents.

II. AREAS OF REVIEW

1. "Radiation Protection Design Criteria and Design Features"

2. "Occupational Exposures"

3. "Public Exposures"

   a. "Normal-conditions"
   b. "Accident Conditions/Natural Phenomena Events"

4. "ALARA"

III. REGULATORY REQUIREMENTS

1. Design Criteria - 10 CFR Part 72.104; 10 CFR Part 72.106

2. Occupational Exposures - 10 CFR Part 20.1201; 10 CFR Part 20.1207; 10 CFR Part 20.1208

3. Public Exposures (Normal and Accident) - 10 CFR Part 72.104; 10 CFR Part 72.106

4. ALARA - 10 CFR Part 20.1101; 10 CFR Part 72.24(e); 10 CFR Part 72.104(b); 10 CFR Part 72.126(a)

IV. ACCEPTANCE CRITERIA

1. Design Criteria

Limitations on cask direct radiation dose rates are based on the shielding and confinement evaluations satisfying the regulatory requirements for public dose limits.

2. Occupational Exposure

   a. Dose Limits for Adults: Total effective dose equivalent, 5 rem/yr

   b. Dose Limits for Minors: 0.5 rem/yr

   c. Dose to an Embryo/fetus (declared pregnant woman): 0.5 rem during entire pregnancy

3. Public Exposures

   a. Normal Operation and Anticipated Occurrences:

   25 mrem/yr - whole body
   75 mrem/yr - thyroid
   25 mrem/yr - other organ

   The above doses include the cumulative effects of other affected nuclear fuel cycle facilities (i.e., the nuclear power plant) and apply to a realistic individual in the general public.

   b. Accident (Design Basis)

   Five rem to the whole body or any organ to any individual located on or beyond the nearest boundary of the controlled area.

4. ALARA

The ALARA policy should be based on the following criteria, as a minimum:

   a. No practice should be adopted unless its use produces a net benefit.

b. All exposures should be kept ALARA with technological, economical, and social factors taken into consideration.

c. The exposure to individuals should not exceed the limits recommended for the appropriate circumstances.

V. REVIEW PROCEDURES

1. Radiation Protection Design Criteria and Design Features

a. Design Criteria

Review the principal design criteria presented in Section 2 of the SAR and any additional detail regarding radiation protection provided in Section 5 ("Shielding") and Section 7 ("Confinement"). In general, most of the criteria will have been discussed in these earlier sections of the SAR and need only be briefly noted. Additional criteria that should be presented in Section 10 of the SAR (if not previously discussed) include the following:

(1) ALARA and other occupational exposure requirements of 10 CFR Part 20[1] must be satisfied.

(2) The sum of the doses from direct radiation and from release of nuclides to the atmosphere must satisfy the requirements of 10 CFR Part 72.104(a) and 72.106(b). Because of the stringent design requirements for spent fuel cask systems, the release of nuclides into the atmosphere is insignificant under normal and accident situations. Direct radiation is the major mode of exposure. Generally, a direct dose rate less than 200 mr/hr at the cask surface is a good reference level since it is consistent with the previously established dose rate level set for transportation in 10 CFR Part 71. To comply with the controlled area boundary dose requirements for an array of casks, the general cask surface area dose rate will, in practice, probably be much less than 200 mr/hr.

b. Design Features

Review the general description and operating features of the cask presented in Section 1 of the SAR and any additional information provided in Sections 5 and 7. In general, all design features relevant to radiation protection have probably been discussed earlier in the SAR and will be noted only briefly in Section 10.

2. Occupational Exposures

Review the operating procedures in Section 8 and direct radiation dose calculations in Section

5. Section 10 should use these data to estimate the doses received by occupational personnel during loading of the cask and transporting it to the ISFSI. Any significant differences in these doses for retrieving and unloading the cask should be identified. Similar dose estimates for periodic or routine maintenance, as well as surveillance activities, should also be presented. These estimates may require additional assumptions concerning adjacent casks for a typical storage configuration.

The bases for the various exposure times, number of personnel required, and appropriate dose rates should be justified. Verify that the calculated doses are consistent with these estimates. Keep in mind that the actual operations will be performed under an active dose monitoring program that further ensures compliance with 10 CFR Part 20 requirements. The worker's monitoring program should be reviewed. NRC's Regulatory Guide 8.34[2], which was developed to implement the revised 10 CFR Part 20, can be used to determine the acceptability of the occupational exposure evaluation and monitoring requirements.

3. Public Exposures

a. Normal-conditions

Review the information regarding the direct dose rate at the site boundary discussed in Section 5. The sum of doses, including some additional margin to account for dose received from other fuel cycle (reactor) operations, must satisfy the requirement of 10 CFR Part 72.104(a). As discussed in Section 5 of this SRP, the direct dose at the site boundary depends on many site-specific conditions, which the SAR may treat in a general manner. Verify that the SAR includes a requirement for site-specific dose analysis and monitoring by the ISFSI licensee or that sufficient bounding analyses are presented. The latter approach will generally require extensive calculations. If a site-specific dose analysis is included, any physical limits or operational constraints assumed in the analysis should be included in Chapter 12 of the SAR and be listed in Section 12. As discussed in Section 7 of the SRP, the normal dose from atmospheric release is often negligible, even if one redundant confinement seal is assumed to have failed.

b. Accident Conditions/Natural Phenomena Events

In a similar manner, examine the information for the direct dose rate at the site boundary discussed in Section 5 and the dose rate from release of radionuclides presented in Section 7 for accident conditions. The sum of these must satisfy the requirements of 10 CFR Part 72.106(b). For a design basis accident, it is not expected that radionuclides will be released to the environment; the major mode of radiation exposure from a design basis accident primarily stems from direct radiation. However, for environmental assessment, a hypothetical accident is postulated, with an unspecified scenario (not credible events) that assumes 100 percent rupture of the fuel rods and cask breach, thereby resulting in radionuclide release. Although not an accident event, the dose is calculated even under adverse meteorological conditions, to illustrate compliance with 10 CFR Part 72.106(b).

4. ALARA

Review the applicant's stated commitment to ALARA and determine whether this commitment influenced the cask design features and operating procedures. In a site-specific ALARA program, the following items should be reviewed and evaluated:

(a) written management policy statement,
(b) organizational structure separating the radiation protection group from the operation group,
(c) designation of a specific individual responsible for coordinating the ALARA program efforts,
(d) training of employees in ALARA principles,
(e) incorporation of ALARA principles into the design features of the facility and equipment,
(f) incorporation of ALARA principles into operational procedures,
(g) development of administrative controls on exposure below regulatory limits,
(h) use of preplanning and mock-up training,
(i) establishment of periodic reviews to determine the effectiveness of the ALARA program,
(j) trend analysis of radiological parameters, and
(k) radiation protection program audits.

To determine if the applicant's ALARA policy is acceptable, review the evidence that the design methods, approach, and interactions are in accordance with the ALARA provision in Regulatory Guides 8.8[3] and 8.10[4].

VI. EVALUATION FINDINGS

Review the 10 CFR Part 72 acceptance criteria and provide a summary for the following:

- The [cask designation] provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR Part 72.104 and 72.106. As required by Part 72, Subpart K, each licensee will perform written evaluations (before use) demonstrating that these requirements are satisfied under site-specific conditions and will establish a monitoring program to verify these requirements, as specified by 10 CFR Part 20, Subpart F.

- Occupational radiation exposures satisfy the 10 CFR Part 20 limits and meet the objective of maintaining exposures ALARA. Additional operational restrictions may be imposed by the site licensee.

- The staff concludes that the design of the radiation protection system of the [cask designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the [cask designation] will enable safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

VII. REFERENCES

1. Code of Federal Regulations, Title 10, Part 20, "Standards for Protection against Radiation."

2. U. S. Nuclear Regulatory Commission, Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," July 1992.

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable," Rev. 3, June 1978.

4. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable," Revision 1-R, May 1977.

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STANDARD REVIEW PLAN FOR STORAGE CASKS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

11.0 ACCIDENT ANALYSES

I. REVIEW OBJECTIVE

This section evaluates the identification and analysis of hazards and provides a summary analysis of system response to off-normal events and conditions as well as accident and extreme ("accident-level or design basis") events. The primary objective of the Section 11 review is to (1) ensure that all credible accidents have been identified, (2) the information presented in the SAR is complete, (3) the input on the safety performance of the cask system in each review area has been adequately analyzed, and (4) regulatory requirements have been met. Cask system performance in off-normal and accident-level events and conditions is addressed in detail in other sections of the SAR.

II. AREAS OF REVIEW

1. "Cause of the Event"
2. "Detection of the Event"
3. "Summary of Event Consequences and Regulatory Compliance"
4. "Corrective Course of Action"

III. REGULATORY REQUIREMENTS

1. Structures, systems, and components important to safety must be designed to withstand credible accidents, and natural phenomena without impairing their capability to perform safety functions. [10 CFR Part 72.24(d)(2); 10 CFR Part 72.122(b)(2), (3), (d), and (g)]
2. Dose Limits for Off-Normal Events. [10 CFR Part 72.104(a); 10 CFR Part 72.236(d); and 10 CFR Part 72.24(d)]
3. Dose Limits for Design Basis Accidents. [10 CFR Part 106(b); 10 CFR Part 72.236(d);10 CFR Part 72.24(m); and 10 CFR Part 72.24(d)(2)]
4. Criticality. [10 CFR Part 72.236(c) and 10 CFR Part 72.124(a)]
5. Confinement. [10 CFR Part 72.236(l)]
6. Retrievability. [10 CFR Part 72.122(l)]

7. Instrumentation [10 CFR Part 72.122(i)]

IV. ACCEPTANCE CRITERIA

A primary difference between off-normal-conditions and accidents/natural phenomena events is the appropriate regulatory and design limits that must be satisfied in each case. For example, the radiation dose from an off-normal event must not exceed the limits specified in 10 CFR Part 72.104(a), whereas an accident or natural phenomena event must not exceed the specifications of 10 CFR Part 72.106(b). Allowable structural criteria may also be different for accident conditions. (Because of the common regulatory and design limitations for both accidents and natural phenomena events, these scenarios are sometimes referred to in the following sections as *accident conditions*.)

1. Dose Limits for Off-Normal Events

During normal operations and anticipated occurrences, the annual dose equivalent to any individual located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to: (1) planned discharges of radioactive materials, radon and its decay products excepted, to the general environment; (2) direct radiation from ISFSI operations; and (3) any other cumulative radiation from uranium fuel cycle operations (i.e., nuclear power plant) in the affected area.

2. Dose Limit for Design Basis Accidents

Any individual located at or beyond the nearest controlled area boundary must not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident.

3. Criticality

The spent fuel must be maintained in a subcritical condition under credible conditions i.e., k_{eff} equal to or less than 0.95. At least two unlikely, independent, and concurrent or sequential changes must be postulated to occur in the conditions essential to nuclear criticality safety before a nuclear criticality accident is possible (double contingency).

4. Confinement

The cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under credible accident conditions.

5. Retrievability

Storage systems must allow ready retrieval of spent fuel for further processing or disposal.

6. Instrumentation

Instruments and control systems that must remain operational under accident conditions must be identified in the SAR.

V. REVIEW PROCEDURES

Review the off-normal-conditions, accidents, and natural phenomena events identified in Section 2 of the SAR. An evaluation of the following information should be included for each of the events, as applicable.

1. Cause of the Event

   In some cases, the event may be analyzed for regulatory purposes even though no credible cause can be identified. These events should be clearly identified as non-mechanistic.

2. Detection of the Event

   The event may be detected by visual surveillance or monitoring instrumentation and alarms. For some events, the method of detection will be intuitively obvious, whereas for others (e.g., fuel rod rupture) the event may remain undetected, at least for a significant period of time.

3. Summary of Event Consequences and Regulatory Compliance

   The event consequences should be addressed in each functional area corresponding to the earlier sections of the SAR—structural, thermal, shielding, criticality, confinement, and radiation protection. The section of the SAR in which each consequence is evaluated in detail should be referenced. The consequences should be shown to be in compliance with the applicable regulatory criteria. Because of the stringent design requirements for dry storage casks, it is expected that a significant release of radioactive material will not occur under normal, off-normal, and design basis accident conditions. The major mode of radiation exposure is from direct radiation. However, to demonstrate the overall safety of cask storage systems, and to illustrate compliance of regulatory limits, a non-credible event with an unspecified scenario is assumed to result in a release of radioactive material. This approach is normally applied in the Environmental Assessment, to illustrate compliance with regulatory requirements.

4. Corrective Course of Action

   This section should identify what action, if any, would be necessary to recover from the event. If various courses of action are possible, a general discussion should be presented on the criteria that will be used to select the most appropriate action. Because the fuel must be readily retrievable, returning the cask to the fuel handling building and reloading the spent fuel into a new cask is a viable option.

The primary emphasis in the review of Section 11 is to assess that information is complete. Coordination among all individual reviews should be accomplished, to ensure that all credible situations have been identified, their impact on each review area has been addressed, and compliance with regulatory criteria is satisfied.

VI. EVALUATION FINDINGS

Review the 10 CFR Part 72 acceptance criteria and provide a summary for the following:

- Structures, systems, and components of the [cask designation] are adequate for the prevention of accidents and the mitigation of the consequences of accidents and natural phenomena events.

- The spacing of casks, discussed in Section _____ of the SER and included as an operating limit in Section 12, will provide for accessibility to the equipment of on-site and available off-site emergency facilities and services, as required.

- Technical specifications for the [cask system designation] are included in the SER and are listed in table __ of the SER.

- The [cask designation] has been evaluated to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.

- An accident or natural phenomena event will not preclude the ready retrieval of spent fuel for further processing or disposal.

- The spent fuel will be maintained in a subcritical condition under accident conditions.

- Off-normal and accident conditions will not result in a dose to an individual outside the controlled area, that exceeds the limits of 10 CFR Part 72.104(a) or 72.106(b), respectively.

- No instruments or control systems are required to remain operational under accident conditions [as applicable].

The staff concludes that the accident design criteria for the [cask designation] are in compliance with 10 CFR Part 72 and that the accident design and acceptance criteria have been satisfied. The accident evaluation of the cask demonstrates that it will provide for safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

12.0 CONDITIONS FOR CASK USE - OPERATING CONTROLS AND LIMITS OR TECHNICAL SPECIFICATIONS

I. REVIEW OBJECTIVE

To ensure that the applicant's proposed operating controls and limits, or technical specifications, including their bases and justification, have been provided and are supported by the technical disciplines under review.

The term *technical specifications* may be considered synonymous with *operating controls and limits*. The technical specifications define the conditions that are deemed necessary and sufficient for safe cask system use and are attached to a cask's certificate of compliance or site-specific license, as applicable. The technical specifications will define operating limits and controls, monitoring instruments and control settings, surveillance requirements, design features, and administrative controls that ensure safe operation of the cask. Each specification should be clearly documented and justified in the technical review sections of the SAR and associated SER as necessary and sufficient for safe cask operation.

II. AREAS OF REVIEW

Technical specifications are discussed in 10 CFR Part 72.44(c), which presents the regulatory requirement for five categories, as listed below.

1. "Functional/Operating Limits and Monitoring Limits/Limiting Control Settings"
2. "Limiting Conditions"
3. "Surveillance Requirements"
4. "Design Features"
5. "Administrative Controls"

III. REGULATORY REQUIREMENTS

1. General Requirement for Technical Specifications

Technical specifications and bases must be proposed. These specifications must include: (1) functional and operating limits and monitoring instruments and limiting control settings, (2) limiting conditions, (3) surveillance requirements, (4) design features, and (5) administrative controls. [10 CFR Part 72.44(c); 10 CFR Part 72.24(g); 10 CFR Part 72.26] Subpart E, "Siting Evaluation Factors," and Subpart F, "General Design Criteria," to 10 CFR Part 72, provide the bases for the cask system design and hence are be applicable as bases for appropriate technical specifications.

2. Specific Requirements for Technical Specifications - Storage Cask Approval

As a condition of approval, the design, fabrication, testing, and maintenance of a spent fuel storage cask must comply with the requirements of 10 CFR Part 72.236. [10 CFR Part 72.234(a)]

Specifications must be provided for the spent fuel to be stored in the cask, such as, but not limited to: type of spent fuel (i.e., BWR, PWR, or both); maximum allowable enrichment of the fuel prior to any irradiation; burn-up (i.e., megawatt-days/MTU); minimum acceptable cooling time of the spent fuel prior to storage in the cask; the maximum heat that the cask system is designed to dissipate; maximum spent fuel loading limit; condition of the spent fuel (i.e., intact assembly or consolidated fuel rods); and the inerting atmosphere requirements. [10 CFR Part 72.236(a)]

Design bases and design criteria must be provided for structures, systems, and components important to safety. [10 CFR Part 72.236(b)]

The cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions. [10 CFR Part 72.236(c)]

Radiation shielding and confinement features must be provided sufficient to meet the requirements in 10 CFR Part 72.104 and 72.106 regarding radioactive material in effluents, direct radiation, and area control. [10 CFR Part 72.236(d); 10 CFR Part 20[1]]

The cask must be designed to provide redundant sealing of confinement systems. [10 CFR Part 72.236(e)]

The cask must be designed to provide adequate heat removal capacity without active cooling systems. [10 CFR Part 72.236(f)]

The cask must be designed to store the spent fuel safely for a minimum of 20 years and permit maintenance as required. [10 CFR Part 72.236(g)]

The cask must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR Part 72. 236(h)]

The cask must be designed to facilitate decontamination to the extent practicable. [10 CFR Part 72.236(i)]

The cask must be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness. [10 CFR Part 72.236(j)]

The cask and its systems important to safety must be evaluated, by appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR Part 72.236(l)]

IV. ACCEPTANCE CRITERIA

Because of the breadth and scope of the technical specifications (operating controls and limits), it is not possible to define each instance where a control or condition is necessary. For this reason, it is important that a detailed, thorough, and independent evaluation of each technical section is conducted by the staff. Specific operational controls and limits necessary to maintain criticality, confinement, shielding, heat removal, and structural integrity under normal and accident operations must be identified and supported in the pertinent SAR chapters.

V. REVIEW PROCEDURES

This section evaluates the operating controls and limits for cask use. The evaluation is based on information presented in Section 12 of the applicant's SAR, applicant commitments discussed in other technical sections of the SAR, applicant commitments made in subsequent correspondence after the initial application, and accepted practices. The fact that all commitments are not explicitly addressed in Chapter 12, does not negate the requirement of the cask system owner/operator to meet and comply with such commitments. Such commitments made be documented in SAR chapters other than Chapter 12, other pertinent applicant correspondence, or may be required by applicable regulations or included in site-specific license conditions. It is therefore, helpful to ensure that such commitments are included in Chapter 12 of the SAR and acknowledged in NRC SER.

Each section of the SAR should be evaluated with the goal of establishing the technical specifications and conditions of use for the cask. Each technical reviewer of the SAR should note all instances in which the SAR has either made an assumption or imposed a condition that should be identified as an operating control and limit. Reviewers should also note any instances where the SAR has requested exceptions or exemptions from regulatory requirements or other conditions that the reviewer has identified as an operational limit or condition. The reviewer of Chapter 12 should ensure that such limits and exemptions are clearly identified and documented in Chapter 12, and that such limits and exceptions are discussed in Section 12 of the SER.

The Licensing Project Manager is responsible for ensuring that all necessary conditions for cask use (technical specifications or operating controls) are explicitly delineated in the SER and in the certificate of compliance or site-specific license, as applicable. NRC technical reviewers assigned to each SAR chapter also have the responsibility to review the applicant's proposed conditions and to identify any additional necessary and sufficient conditions for cask use. The Licensing Project Manager should ensure that the conditions for use, as reviewed and approved by the technical reviewers, complement one another and are not internally contradictory. The Licensing Project Manager will coordinate the resolution of any disputed condition, limit or specification. The Licensing Project Manager is responsible for identifying any unique specifications (e.g., administrative) that may not be covered in the technical sections, although input may be solicited from the technical reviewers regarding any topic.

US NRC Regulatory Guide 3.61[2] provides a recommended format for use by an applicant for presenting operating controls and limits. However, this format may not be applicable to all controls. Since the basis for the control may have been extensively discussed earlier in the SAR, the applicant may use an abbreviated format.

Upon completion of the review, a separate table or appendix for Section 12 of the SER may be prepared by the Licensing Project Manager to designate explicitly those operating controls and limits applicable to the cask.

Note - Site-Specific License and Cask Certificate of Compliance

Reviewers should also note that additional conditions expected for cask system operation are identified in 10 CFR Part 72[3]. A spent fuel storage cask may be used under a site-specific license or under the provisions of the general license. Although the review mechanism for casks used under a site-specific license is similar to that for a cask system used under the general license, the regulatory requirements for a site-specific license are slightly different than the general license provisions of 10 CFR Part 72. As such, staff should review 10 CFR Part 72, Subpart C, "Issuance and Conditions of License;" Subpart E, "Siting Evaluation Factors;" Subpart F, "General Design Criteria;" Subpart G, "Quality Assurance;" Subpart H, "Physical Protection;" and Subpart I, "Training and Certification of Personnel," for applicable ISFSI operating constraints or conditions of operation that may not be appropriate as technical specifications, but may warrant inclusion as license conditions for use of the cask system.

VI. EVALUATION FINDINGS

The Licensing Project Manager (and each technical reviewer) should review the applicable 10 CFR Part 72 acceptance criteria and should be able to provide a summary statement similar to the following:

> The staff concludes that the conditions for cask use for the [cask designation] identify the necessary and sufficient operating controls and limits to satisfy 10 CFR Part 72, and that the applicable acceptance criteria have been satisfied. The evaluation of the operating controls and limits provides reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.
>
> Technical specifications for the [cask designation] have been identified in Section 12 of the SAR and are listed in Table _____ of the SER.

VII. REFERENCES

1. U.S. Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Facility," February 1989.

3. U.S. Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

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13.0 QUALITY ASSURANCE

REVIEW RESPONSIBILITIES

The review of the applicant's Quality Assurance program for activities under 10 CFR Part 72, Subpart G, is conducted independently from the SAR by Staff and is not included in the Safety Evaluation Report.

14.0 DECOMMISSIONING

I. REVIEW OBJECTIVE

At the end of an ISFSI operation, spent fuel casks will be removed from site for ultimate disposal. Any contamination remaining at the site will be cleaned-up to acceptable levels such that the facility can be released for unrestricted use and the ISFSI license under 10 CFR Part 72 will be terminated. Because the actual decommissioning will occur at a future date, perhaps more than 20 years after first use of the cask, and will employ site-specific procedures available at that time, 10 CFR Part 72 does not require the development of detailed decommissioning plans until near the time of license termination. Therefore, during the licensing of a proposed ISFSI, a conceptual decommissioning plan shall be submitted for evaluation. NRC requires a commitment from the site-specific licensee to submit a detailed decommissioning plan for review and approval before initiating decommissioning activities at the facility. For a general licensee, the ISFSI decommissioning should be included with the reactor decommissioning plan.

II. AREAS OF REVIEW

1. Projection and prediction of the decommissioning activities, types of waste generated, types of contamination, and waste disposal method

2. Commitment to decontaminate to applicable NRC criteria

3. Cost of decommissioning and financial assurance

4. Commitment to submit a timely detailed decommissioning plan for NRC's review and approval before decommissioning action

III. REGULATORY REQUIREMENTS

Financial assurance and record-keeping requirements associated with decommissioning of an ISFSI are found in 10 CFR Part 72.30.

Requirements for termination of an ISFSI license and decommissioning of ISFSI sites and buildings are addressed in 10 CFR Part 72.54.

Regulatory requirements for criteria to be used during decommissioning and ISFSI are found in 10 CFR Part 72.130.

Design of casks systems to provide for decommissioning are delineated in 10 CFR Part 72.236(i).

IV. ACCEPTANCE CRITERIA

1. Criteria for Decontamination of Buildings and Equipments are as specified in Regulatory Guide 1.86[1].

2. Criteria for Classification and Disposal of Wastes is discussed in 10 CFR Part 61.55[2].

V. REVIEW PROCEDURES

Review the general description and operating features of the cask system and its application to an ISFSI facility discussed in Section 1 of the SAR. Review and verify that the applicant's (1) projected decommissioning activities, (2) types of waste generated, (3) types of contamination, and (4) waste disposal method are generally accurate and acceptable.

After the casks have been decontaminated, the major radiation sources are those resulting from activation of the cask components themselves, such as concrete shielding material during the storage period. Verify that the activities of these nuclides have been properly estimated. Several of the activation products are short-lived, and the SAR may discuss their activity as a function of time after unloading. Although the specific activation products depend on the materials initially present in the cask components, the nuclides of interest are generally Cr-51, Mn-55, Fe-58, Co-58, Co-60, and Ni-63. A significant reduction in the total activation occurs in only 1 year after unloading.

Because of the low levels of spontaneous fission and subcritical multiplication in the spent fuel during the storage period, the activation of the cask components is generally very minor and can be approximated by simple, conservative methods. A typical approach is to use the same flux calculated from the deterministic shielding analysis performed in Chapter 5 of the SAR, along with appropriate cross-sections from the same calculation. For conservatism, activation of the cask body is determined from the flux at the inner surface. Equilibrium activities of the irradiated structures are generally calculated without considering the time dependence of the flux during the storage period.

Another common approach is to determine a conservative one-group flux, select a conservative cross-section library, and calculate the activities of resulting radioactive nuclides using ORIGEN2[3]. This evaluation is to ensure that the activated cask components can be disposed of in a low-level waste disposal site. The acceptable criteria are specified in the tables contained in 10 CFR Part 61.55[2]. Radiological survey information in the applicant's final decommissioning plan should provide verification of the waste classification and acceptable disposal methods. Other wastes generated from the applicant's decontamination activities should also be reviewed and evaluated to determine its quantity and its acceptability for disposal under 10 CFR Part 61.

Review the estimates on the cost of decommissioning and financial assurance to decontaminate the facility and site.

VI. EVALUATION FINDINGS

Review the 10 CFR Part 72 acceptance criteria and provide a summary for the following:

The cask is designed for decommissioning. As discussed in Section _____ of the SAR, provisions are made to facilitate decontamination of the cask and minimize the quantity of radioactive wastes, contaminated equipment, and contaminated materials at the time the cask is permanently decommissioned.

Information on proposed practices and procedures for the decontamination of the cask and for disposal of residual radioactive materials after all spent fuel has been removed is described in Section _____ of the SAR and provides reasonable assurance that decontamination and decommissioning will provide adequate protection to the health and safety of the public.

The staff concludes that the decommissioning considerations for the [cask designation] are in compliance with 10 CFR Part 72. This evaluation provides reasonable assurance that the [cask designation] will enable safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," June 1974.

2. U. S. Code of Federal Regulations, Title 10, Part 61, "Licensing Requirements for Land Disposal of Radioactive Wastes."

3. Oak Ridge National Laboratory, "ORIGEN2.: Isotope Generation and Depletion Code-Matrix Exponential Method," 1991.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Standard Review Plan (SRP) for Dry Cask Storage Systems provides guidance to the Nuclear Regulatory Commission staff in the Spent Fuel Project Office for performing safety reviews of dry cask storage systems. The SRP is intended to ensure the quality and uniformity of the staff reviews and present a basis for the review scope and requirements. Part 72, Supart B generally specifies the information needed in a license application for the independent storage of spent nuclear fuel and high level radioactive waste. Regulatory Guide 3.61 "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask" contains an outline of the specific information required by the staff. The SRP is divided into 14 sections which reflect the standard application format. Regulatory requirements, staff position, industry codes and standards, acceptance criteria, and other information are discussed. Comments on this draft, will be considered and incorporated into the SRP as appropriate. The SRP is scheduled for publication as an NRC NUREG document late in 1996. Comments, errors or omissions, and suggestions for improvement should be sent to the Director, Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001

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