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, present a basis for the review scope, and clarification of the regulatory 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, 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.

DEFINITIONS

Accident-Level. Term used to include both design-basis accidents and design-basis natural phenomenon events and conditions. [See also "Design-Basis ______."] Resistance, response limit, and functional capability requirements apply for conditions and events that exceed "off-normal" or "Design Event II," as described in ANSI/ANS 57.9.

Basic, or fundamental, safety criteria. The following minimal functions for nuclear safety in the design of an ISFSI or MRS facility:

- Maintain subcriticality.
- Prevent release of radioactive material above acceptable amounts.
- Ensure that radiation rates and doses do not exceed acceptable levels.
- Maintain retrievability of the stored radioactive materials throughout the life of the DCSS.

Benchmarking. Validation of the accuracy of a computer code by comparison of the calculated results with those of relevant experiments.

Bias. ANSI/ANS-8.1 defines bias as "a measure of systematic disagreement between the results calculated by a method and experimental data. The uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of he experimental data." See NUREG/CR-6361 for further discussion of bias. Bias defined as the average of the differences between results and measurements is acceptable, provided that one adequately considers the variation in the differences.

Code. Used generically to refer to national or "consensus" codes, standards, and specifications, or specifically to refer to the ASME Boiler and Pressure Vessel Code.

CDE. Committed Dose Equivalent, defined as the total radiation dose equivalent to the body (or specified part of the body) that will be accumulated over 50 years following an intake of radioactive material.

Confinement 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.

Containment The assembly of components of the packaging intended to retain the radioactive material during storage.

Confirmatory Calculations. Calculations made by the reviewer to determine whether the package design and specifications meet the regulations. These calculations do not replace the design calculations and are not intended to endorse the applicant's calculations.

Construction. The assembly, fabrication, or putting together of standard parts or components to form structures of systems of a DCSS

Controlled Area. That area immediately surrounding an ISFSI or MRS for which the licensee exercises authority over use and within which ISFSI operations are performed. See 10 CFR 72.3.

Design-Basis _______. The extreme level of an event or condition for which there is a specified resistance, limit of response, and requirement for a given of continuing capability. (Compares with "Design Events III and IV" as described in ANSI 57.9.)

Design Event (I, II, III, or IV). Conditions and events as defined and used for an ISFSI in ANSI/ANS 57.9 (also applicable to an MRS).

Exclusion Area. [Applies to sites with a reactor only] "That area surrounding the reactor, in which the reactor licensee has the authority to determine al activities including exclusion or removal of personnel and property from the area." [10 CFR 100, with additional descriptors included at 10 CFR 100.3.]

Gross Cladding Defect. A known or suspected cladding condition that results in the fuel not meeting its design-basis criteria for dry cask storage. The cask shielding, criticality, thermal, and radiological design analyses typically assume that the cladding provides sufficient structural integrity to retain the fuel pellets in the fuel assembly geometry for normal and accident conditions [1]. In addition, both individual fuel rods and fuel assemblies should be intact to preclude fuel handling or operational safety problems during loading and unloading operations. It is the responsibility of the licensee to ensure that fuel placed in dry storage meets the design-basis conditions. This definition is applicable to all phases of dry cask storage (from selection and inspection of the fuel before loading until the fuel is unloaded from the cask or the cask is placed in a permanent repository). Alternative means, such as canning, will be required for dry cask storage of fuel that does not meet design-basis conditions.

Hard Receiving Surface for a horizontal or vertical drop need not be an unyielding surface; rather the receiving surface may be modeled as a reinforced concrete pad on engineered fill.

Important Confinement Features. Term used in ANSI/ANS 57.9, but not acceptable to the NRC. (Per RG 3.60, "important to safety" should be substituted for "important confinement features" in the standard.)

Important to Safety [also "Important to Nuclear Safety"]. A function or condition required to store spent fuel of high-level waste safely. To prevent damage to the spent fuel or the high-level waste container during handling and storage, to provide reasonable assurance that spent fuel or high level radioactive waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Independent Calculation Calculations separate from the applicant's. Input data should be taken from primary sources such as the package drawings and manufacturer's specifications. Models should be developed separately by the reviewer. To the extent possible, different techniques, codes, and cross section sets or other derived data sets should be used.

Intact Cladding. Spent fuel cladding that does not have gross cladding defects (see Gross Cladding Defects).

Mixed waste. Waste material that is hazardous because it contains both radioactive material as well as chemical, toxic, incendiary, or other hazards.

MofS. Margin of safety, which may be defined as identical to factor of safety, f.s. = capacity/demand (with minimum acceptable MofS $\geq$1.0), or as a true margin, where MofS = f.s.-1 = (capacity/demand) - 1 (with minimum acceptable MofS $\geq$ 0.0.

NDE: Nondestructive examination: testing, examination, and/or inspection of a component which does not affect the use of the component. NDE can be broadly divided into three categories: visual, surface, and volumetric examinations. [Additional information may be found in the ASME B&PV Code, Section V, Nondestructive Examination, Appendix A.]

NDE related terms in order of increasing severity:

discontinuity:	an interruption in the normal physical structure of a material. Discontinuities may be unintentional, such as those formed inadvertently during the fabrication process, or intentional, such as a drilled hole.
indication:	detection of any discontinuity using an NDE method.
flaw:	detection of an imperfection or unintentional discontinuity using an NDE method.
defect:	a flaw which, due to its size, shape, orientation, location, or other properties, is rejectable to the applicable construction code. Defects may be detrimental to the intended service of a component and the component must be repaired or replaced.

Common NDE examination methods include:

[1] The statements of consideration for the standardized NUHOMS spent fuel storage cask FR 65898, comment F.4, provides an example of past staff practice regarding the allowable condition of fuel cladding for loading in a cask. "Licensees and Certificate of Compliance generally require that the fuel have no known or suspected gross cladding breaches to ensure the structural integrity of the fuel. Known or suspected failed fuel assemblies (rods) and fuel with cladding defects greater than pin holes and hairline cracks are not authorized in the Standardized NUHOMS."

LT leak testing
MT magnetic particle examination
PT liquid penetrant examination
RT radiographic examination
UT ultrasonic examination
VT visual examination

destructive examination: testing, examination, and/or inspection of a component which results in the destruction of the component.

Normal. The maximum level of an event or condition expected to routinely occur. The ISFSI or MRS is expected remain fully functional and to experience no temporary or permanent degradation from normal operations, events, and conditions. (Compares to "Design Event I" of ANSI/ANS 57.9.) Events and conditions that exceed "normal" levels are considered to be, and to have the response allowed for, "off-normal" or "accident-level" events and conditions.

Off-Normal. The maximum level of an event or condition that although not occurring regularly can be expected to occur with moderate frequency and for which there is a corresponding maximum specified resistance, limit of response, or requirement for a given level of continuing capability. (Similar to "Design Event II" of ANSI/ANS 57.9.) ISFSI SSC are expected to experience off-normal events and conditions without permanent deformation or degradation of capability to perform their full function (although operations may be suspended or curtailed during off-normal conditions) over the full license period.

Other radioactive wastes. Components generally associated with the spent fuel, e.g. Control Assemblies (Rods) BWR fuel channels etc.

Quality Group. NRC classification of SSCs by degree of importance to nuclear safety (NUREG-0800, §3.2.2, and Regulatory Guide 1.26) for reactor systems and adapted to use with ISFSI as follows:

- *Quality Group B* is accepted for "construction" [see term in Glossary] of ISFSI and MRS confinement vessels and their integral and contained components.

- *Quality Group C* is accepted for construction of fluid systems that may be connected to a penetration of the confinement barrier and used or located outside an NRC-licensed enclosing structure providing tertiary confinement (e.g., the fuel pool building).

- *Quality Group D* is accepted for construction of fluid systems that may be connected to a penetration of the confinement barrier and used or located outside an NRC-licensed enclosing structure providing tertiary confinement, if analysis shows that the maximum conservatively estimated offsite dose (the analysis procedure identified in RG 1.26, Subsection C.2.d, is acceptable) would not exceed 0.5 rem to the whole body or any equivalent part of the body.

Radwaste. Waste that is hazardous because it contains nuclear materials (may be high- or low-level.

Ready Retrievability. Capability to return the stored radioactive material to a safe condition without the release of radioactive materials to the environment or radiation exposures in excess of the limits defined by 10 CFR 20 [10 CFR 72.122(h)(5)]. ISFSI and MRS storage systems must be designed to allow ready retrieval of the stored spent fuel or high-level waste (MRS only) for compliance with 10 CFR 72.122(l).

Restricted Area. "Any area to which the licensee controls access to protect individuals from exposure to radiation and radioactive materials." [10 CFR 20]

Safety Analysis Report. In the context of the FSRP, the report submitted by the license applicant in compliance with 10 CFR 72, Subpart B or I. The fundamental contents of the report are described at 10 CFR 72.24. Guidance regarding the content of the report is provided by Reg. Guides 3.48, 3.61 and 3.62. For the staff review, the SAR is considered to constitute the actual SAR submitted with the application, along with supplemental data submitted with the application and supplemental data and responses submitted following the application during the NRC staff review and evaluation. The effective SAR is considered by the staff to be that submitted, as amplified and/or modified by the supplemental and later submissions that are docketed.

Safety Evaluation Report. In the context of the FSRP, the report prepared by the NRC staff to present findings and recommendations relating to the acceptability of the applicant's safety analysis and other required submissions. The SER also identifies the bases for those recommendations and the recommended technical specifications ("operating controls and limits" or "conditions of use").

Unrestricted Area. "Any area to which the licensee need not control access in order to protect individuals from exposure to radiation and radioactive materials." [10 CFR 20]

Volume %. The percent of a mole of the material that is present in a volume equal to the standard volume for the material as a gas; the volume occupied by one mole of the material as a gas at standard conditions for gases (760 mm Hg (760 torr) pressure and 0 °C temperature).

INTRODUCTION

This standard review plan (SRP) provides guidance for use by staff reviewers from the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards (NMSS), Spent Fuel Project Office, in performing safety reviews of applications for approval of spent fuel dry cask storage systems (DCSS). The principal purposes of the DCSS SRP are to ensure the quality and consistency of staff reviews and to establish a well-defined basis from which to evaluate proposed changes in the scope of reviews.

Other purposes of this SRP are to ensure wide availability of information about regulatory matters, to improve communication, and to help interested persons and the nuclear power industry better understand the staff review process.

The regulations (10 CFR Part 72) that govern the storage of spent nuclear fuel are largely performance based. An example of a performance based regulation can be found in 72.122 Overall requirements:

> (a) Quality Standards. 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.

This SRP describes the process and provides the reference documents reviewers need to evaluate what "commensurate with importance to safety" means and to evaluate it constantly with respect to the many different designs for DCSS that may be submitted for approval.

A DCSS may be used to store spent nuclear fuel under either a site-specific or general license to operate an independent spent fuel storage installation (ISFSI). At present, any holder of an active reactor operating license under Title 10, Part 50, of the *U.S. Code of Federal Regulations* (10 CFR Part 50), has the authority to construct and operate an ISFSI under the provisions of the general license. Requirements for construction and pre-operational activities of such an ISFSI are discussed in Subparts K and L of 10 CFR Part 72. The requirements for pursuing a site-specific ISFSI license are discussed in Subparts B and C of 10 CFR Part 72. Regardless of the license type, the NRC staff must review and approve the cask design that will be used in an ISFSI before spent fuel loading begins. This SRP describes the methods used by the NRC staff to conduct such a review.

The DCSS safety review 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 require inclusion of an SAR in each application for a license to store spent nuclear fuel or for approval of spent fuel casks. Before submitting an SAR, an applicant should have designed and analyzed the storage cask system in sufficient detail to conclude that it can be properly fabricated and safely operated without endangering the health and safety of the public. The SAR is the principal document in which the applicant provides the information that reviewers need in order to understand the bases for reaching the conclusion that the storage cask is acceptable for use.

Section 72.24 specifies, in general terms, the information to be supplied in an SAR. The specific information required by the staff for evaluation of an application is identified in Regulatory Guide (RG) 3.61, "Standard Format and Content of Topical Safety Analysis Reports for a Spent Fuel Dry Storage Facility." The sections of this SRP are keyed to the standard format defined in RG 3.61. Similar information is also provided in RG 3.62, "Standard Format and Content for the Safety Analyses Report for On-Site Storage of Spent Fuel Storage Casks."

This SRP is written to address a variety of site conditions and cask system designs. Each section presents the complete review procedure and all current acceptance criteria for all pertinent areas of review. However, for any given application, the staff reviewers may select and emphasize the particular aspects of each SRP section that are appropriate for a given 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 it necessary to ask additional questions or probe areas in greater depth, in order to adequately review a particular design. Review plans have not been included for SAR sections that consist of background or design data that are included for information or for use in reviewing other SAR sections.

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 states the purpose and scope of the review.

II. Areas of Review

This subsection describes the systems, components, analyses, data, or other information that are reviewed as part of the given SAR section. It also discusses 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 the applicable sections of 10 CFR Part 72 pertaining to the given SAR section. This list is not all inclusive (e.g., some parts of the regulations, such as 10 CFR Part 20, are assumed to apply to all sections of the SAR).

IV. Acceptance Criteria

This subsection addresses the design criteria and in some cases specific analytical methods that NRC staff reviewers have found to be acceptable for meeting the regulatory requirements, specified in 10 CFR Part 72, that apply to the given SAR section.

These acceptance criteria typically set forth the solutions and approaches that staff reviewers have previously determined to be acceptable 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 solutions and approaches are acceptable to the staff, but they are not the only possible solutions and approaches. Applicants should recognize that, as in the case of regulatory guides, substantial staff time and effort has gone into developing these acceptance criteria, and 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. Such an alternative proposal could consist of a submittal of a Topical Safety Analysis Report (TSAR). This type of application could form the basis for either a change in the staff interpretation of the regulatory requirements or support a request for rulemaking to change the requirements themselves.

V. Review Procedures

This subsection discusses how the review is to be accomplished, including the general procedure that reviewers follow to establish 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 given review area. For each area, a conclusion of this type is included in the safety evaluation report (SER) in which the staff reviewers publish their findings. The SER also describes 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.

VII. References

This subsection lists the references commonly used in the review process for the given subject area.

The SRP and RG 3.61 are directed toward storage cask systems designed for spent fuel with zircalloy cladding. Staff reviewers may adapt the SRP as needed for use in reviewing other storage designs and spent fuel types.

The SRP results from years of staff experience establishing and using regulatory requirements to review SARs and to evaluate the safety of spent fuel storage system designs. This SRP may be considered a part of the continuing regulatory standards development process and documents current review methods.

The SRP may be revised and updated as the need arises to clarify the content, correct errors, or incorporate modifications approved by the Director of the SFPO. Comments, suggestions for improvement, and notices of errors or omissions will be considered by and should be sent to the Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555.