



NUREG-1536
Revision 1A

Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility

Draft Report for Comment



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3 **ABSTRACT**

4 The Standard Review Plan (SRP) for dry storage systems (DSS) provides guidance to the U.S.
5 Nuclear Regulatory Commission (NRC) staff in the Division of Spent Fuel Storage and
6 Transportation (SFST) for reviewing applications for a Certificate of Compliance (CoC) of a dry
7 storage system (DSS) for use at a general license facility. This SRP is intended for use by the
8 NRC staff. Its objectives are to:

- 9
- 10 • provide a basis that promotes a consistent regulatory review of an application for
11 a DSS;
 - 12 • promote quality and uniformity of these reviews across each technical discipline;
 - 13 • present a basis for the review scope;
 - 14 • identify acceptable approaches to meeting regulatory requirements; and
 - 15 • develop a risk informed approach for review of each review procedure section of
16 each chapter to assist the staff in prioritization of its review.
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21 Title 10 of the *U.S. Code of Federal Regulations* (CFR) Part 72 (10 CFR 72), Subpart B,
22 specifies the information needed in a license application for the independent storage of spent
23 nuclear fuel for a site specific application. Subparts A specifies the information needed in an
24 application for a CoC for use at a general license facility. Regulatory Guide 3.61, *Standard*
25 *Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask*,
26 contains an outline of the information required by the staff. This SRP is divided into 14 chapters
27 with appendices that reflect the standard application format. Regulatory requirements, staff
28 positions, industry codes and standards, acceptance criteria, and other information are
29 discussed. However, the format used herein has evolved and, in some instances, superseded
30 Regulatory Guide 3.61 to better reflect current staff practice.
31

32 In conjunction with the SRP, the SFST developed several Interim Staff Guidance (ISG)
33 documents. An ISG addresses emergent review issues in a timely manner by staff and
34 applicants. These ISGs were developed to address changes in requirements, reflect lessons
35 learned and evolving technology, and document detailed technical positions. Current ISGs are
36 available on the NRC website. Although this SRP was revised to incorporate ISG 1 through ISG
37 22 as applicable, ISGs will continue to be developed as needed. This SRP will be revised
38 periodically to reflect current guidance to the staff.
39

40 The review procedures sections of each chapter of this SRP have been risk informed to assist
41 the NRC staff in prioritizing its review in an effort to increase efficiency. The method used to risk
42 inform the Review Procedures sections is documented in Appendix B. The priority of each
43 review procedure is shown in the applicable section of each chapter.
44

45 Comments are solicited on this document and applicable ISGs. Comments, errors or
46 omissions, and suggestions for improvement should be sent to the Director, Division of Spent
47 Fuel Storage and Transportation, U.S. Nuclear Regulatory Commission, Washington, DC
48 20555-0001.
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478

ACRONYMS AND ABBREVIATIONS

ACI	American Concrete Institute
ADE	annual dose equivalent
AISC	American Institute of Steel Construction
ALARA	as low as is reasonably achievable
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ANSI	American National Standards Institute
API	American Petroleum Institute
APSR	axial power shaping rod
ASD	allowable stress design
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWWA	American Water Works Association
AWS	American Welding Society
B&PV	boiler and pressure vessel
BPRA	burnable poison rod assembly
BR	breathing rate
BWR	boiling-water reactor
CDE	committed dose equivalent
CEA	control element assembly
CEDE	committed effective dose equivalent
CFD	computational fluid dynamics
CFR	U.S. Code of Federal Regulations
CoC	Certificate of Compliance
CSFM	Commercial Spent Fuel Management Program
DBA	design-basis accident
DBE	design-basis event
DCF	dose conversion factor
DSS	dry storage system
DDE	deep dose equivalent
DE	design earthquake

DLF	design load factor
DOE	U.S. Department of Energy
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
FR	Federal Registry
g	gram
Gr	Grashof
GTCC	greater than Class C
Gy	Gray
Gz	Graetz
HAC	hypothetical accident condition
HAZ	heat affected zone
HTGR	high-temperature gas-cooled reactor
H/U	hydrogen-to-uranium
IBC	International Building Code
ICBO	International Conference of Building Officials
ICC	International Code Council
ICRP	International Commission on Radiological Protection
INEL	Idaho National Engineering Laboratory
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
LANL	Los Alamos National Laboratory
LCO	limiting condition of operations
LDE	lens dose equivalent
LLNL	Lawrence Livermore National Laboratory
LRFD	load resistance factor design
LT	leak testing
LWR	light water reactor
mJ	milliJoule
mm	millimeter
MNOP	maximum normal operating pressure

MPa	megapascal
ms	millisecond
MT	magnetic particle examination
N	Newton
NDE	nondestructive examination
NDT	nil-ductility transition
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NOAA	National Oceanic and Atmospheric Administration
NRC	United States Nuclear Regulatory Commission
NRPB	National Radiation Protection Board
NRR	Office of Nuclear Reactor Regulation
OBE	operating-basis earthquake
OFA	optimized fuel assembly
ORNL	Oak Ridge National Laboratory
PNL	Pacific Northwest Laboratory
PT	liquid (dye) penetrant examination
PWHT	preheat and post-weld heat treatment
PWR	pressurized-water reactor
QA	Quality Assurance
QAPD	Quality Assurance Program Description
QC	quality control
RAI	request for additional information
RC	reinforced concrete
RCCA	rod cluster control assembly
RG	Regulatory Guide
RSICC	Radiation Safety Information Computational Center
RT	radiographic examination
SAR	Safety Analysis Report
SDE	shallow (skin) dose equivalent
SEM	scanning electron microscopy
SER	Safety Evaluation Report

SFST	Division of Spent Fuel Storage and Transportation
SNF	spent nuclear fuel
SNT	American Society for Nondestructive Testing
SRP	Standard Review Plan
SSC	structures, systems, and components
SSE	safe shutdown earthquake
Sv	Sievert
TEDE	total effective dose equivalent
TEM	transmission electron microscopy
TODE	total organ dose equivalent
TS	Technical Specification
TSAR	Topical Safety Analysis Report
UBC	Uniform Building Code
UK	United Kingdom
UT	ultrasonic examination
VT	visual examination

481
482
483

UNITS

Btu/hr.ft. °F	British thermal units per hour-foot-degree Fahrenheit
°C	degrees Centigrade
Ci/cm ³	Curies per cubic centimeters
Ci/s	Curies per second
cm ³ /s	cubic centimeters per second
°F	degrees Fahrenheit
ft	feet
ft/s	feet per second
ft ³	cubic feet
ft ³ /s	cubic feet per second
g/cm ³	grams per cubic centimeters
GWd/MTU	GigaWatt days per Metric Ton Uranium
in.	inches
K	Kelvin
kg	kilogram
kgf/cm ²	kilograms force per square centimeters
kPa	kiloPascal
ksi	thousand pounds per square inch
kW	kilowatts
lb	pounds
m	meters
m ²	square meters
m ³	cubic meters
m ³ /s	cubic meters per second
m/s	meters per second
mCi	millicuries (one-thousandth of a curie)
MeV	million electron volts
mg	milligram (one-thousandth of a gram)
mm	millimeters (one-thousandth of a meter)
MPa	MegaPascal (million Pascals)
mrem	millirem (one-thousandth of a rem)
mSv	millisievert (one-thousandth of a sievert)
MWd/MTU	MegaWatt days per Metric Ton Uranium
pCi/m ³	picoCurie (one-trillionth of a curie)/cubic meter
PM ¹⁰	particulate matter (less than 10 microns in diameter)
ppm	parts per million

psi	pounds per square inch
rem	roentgen equivalent man
s	second
Sv	sievert
μCi	microcurie (one-millionth of a curie)
$\mu\text{Ci}/\text{cm}^2$	microcurie per square centimeter
W/m.K	Watts per meter - Kelvin

GLOSSARY

The following terms are defined here by the staff for the purpose of this document.

Acceptance Test. Tests conducted by the applicant to ensure that material or component produced in a given production run is in compliance with the material or design requirements of the application. Acceptance tests are also used to ensure that the process is operating in a satisfactory manner by using statistical data for selected measurable parameters.

Accident-Level. A term used to include both design-basis accidents and design-basis natural phenomenon events and conditions.

Areal Density. Mass per unit area, usually expressed in grams per square centimeters (g/cm²). In this document, this term is used to describe the distribution of neutron absorber content in a material.

Adequate Margin. In the design of structures, systems, and components, the margin for safety is achieved by satisfying the acceptance criteria of the codes and standards for the specified design criteria loads, and the design basis (performance requirements). The reviewer must judge if the calculated design bases values require any margins with respect to the acceptance criteria of the codes and standards. This may depend on the uncertainties associated with the calculation of predicted design bases values (stress, displacements, etc.) used as reference for the performance of the structures.

As Low As is Reasonably Achievable (ALARA). Making every reasonable effort to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20 as is practical and consistent with the purpose for which the licensed activity is undertaken taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest (10 CFR 20.1003). Per 10 CFR 72.3, ALARA means as low as reasonably achievable taking into account the state of technology, and the economics of improvement in relation to: (1) benefits to the public health and safety, (2) other societal and socioeconomic considerations, and (3) the utilization of atomic energy in the public interest.

Benchmarking. Establishment of the bias of a computer code for a particular application by comparison of the calculated results with the measured results of relevant representative experiments. For purposes of criticality analyses, benchmarking is the process of establishing the bias of the calculational method, which includes aspects such as the computer code, cross sections set, analyst's technique, and analysis assumptions.

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

Burnable Poison Rod Assembly (BPR). An assembly of poison rods used to absorb neutrons created in the nuclear reactor to control the power produced in the associated fuel assembly during the early core life. The BPRs are inserted into the fuel assemblies through the upper end

535 fittings of the assembly and held in place against lift forces in the core by a retainer mechanism.
536 BPRs within the spent fuel assembly envelope may be approved for storage in a dry storage
537 system as part of the spent fuel assembly.
538

539 Burnup. The measure of the thermal power extracted from a specific amount of nuclear fuel
540 through fission, usually expressed in units of MWd/MTU (megawatt days per metric ton of
541 uranium). For the purpose of specifying the allowable content, the maximum burnup of the fuel
542 must be specified in term of peak rod average. To facilitate containment, criticality, shielding,
543 and thermal analyses, an equivalent assembly average burnup shall be specified.
544

545 Calculational Method. The calculational procedures – mathematical equations, approximations,
546 assumptions, and associated numerical parameters (e.g., cross sections) – that yield the
547 calculated results (ANSI/ANS-8.1-1998).
548

549 Canister. In a dry storage system for spent nuclear fuel, a metal cylinder that is sealed at both
550 ends and may be used to perform the function of confinement. Typically, a separate overpack
551 performs the radiological shielding and physical protection function.
552

553 Canning. To store damaged or consolidated spent nuclear fuel or nuclear fuel debris in a
554 separate container and confine it in such a way that degradation of the fuel during storage will
555 not pose operational safety problems with respect to its removal from storage
556 [10 CFR 72.122(h)(1)].
557

558 Cask. In a dry storage system using the cask design for spent nuclear fuel, a passive stand-
559 alone component that performs the functions of confinement, radiological shielding, decay heat
560 removal, and physical protection of spent fuel during normal, off-normal, and accident-level
561 conditions (NUREG-1571).
562

563 Certificate of Compliance. The certificate issued by the NRC that approves the design of a
564 spent nuclear fuel storage cask in accordance with the provisions of Subpart L of 10 CFR 72
565 (10 CFR 72.3).
566

567 Code. A generic reference to a national or “consensus” code, standard, and specification, or
568 specifically to the ASME Boiler and Pressure Vessel Code (ASME B&PV Code).
569

570 Committed Dose Equivalent ($H_T, 50$). The dose equivalent to organs or tissues of reference (T)
571 that will be received from an intake of radioactive material by an individual during the 50-year
572 period following the intake (10 CFR 20.1003).
573

574 Confinement. The ability to prevent the release of radioactive substances into the environment
575 (NUREG-1571).
576

577 Confinement System. Those systems, including ventilation, that act as barriers between areas
578 containing radioactive substances and the environment (10 CFR 72.3).
579

580 Confirmatory Calculations. Calculations made by the reviewer to determine whether the cask
581 design and specifications meet the requirements of the Code of Federal Regulations. These
582 calculations do not replace the design calculations and are not intended to endorse the
583 applicant’s calculations.
584

585 Construction. Includes materials, design, fabrication, installation, examination, testing,
586 inspection, and certification as required in the manufacture and installation of components.

587
588 Controlled Area. For an independent spent fuel storage installation (ISFSI), that area
589 immediately surrounding the ISFSI for which the licensee exercises authority over its use and
590 within which ISFSI operations are performed (10 CFR 72.3). For a nuclear power plant, that
591 area outside of a restricted area but inside the site boundary to which access can be limited by
592 the licensee for any reason (10 CFR 20.1003).

593
594 Criticality. A measurement of the state of a fission system.

595
596 Curie. The basic unit of radioactivity. A curie is equal to 37 billion (3.7×10^{10}) disintegrations
597 per second.

598
599 Damaged Fuel. Spent nuclear fuel is considered damaged for storage or transportation
600 purposes if it cannot fulfill its regulatory or design function. Specific conditions that define
601 damaged fuel are provided in Section 8.4.17.2 of this SRP. Section 8.6, Supplemental
602 Information for Methods for Classifying Fuel, provides methods for classifying spent nuclear fuel
603 as damaged.

604
605 Damaged-Fuel Can. A metal enclosure that is sized to confine one damaged spent fuel
606 assembly. A fuel can for damaged spent fuel with damaged spent-fuel assembly contents must
607 satisfy fuel-specific and system-related functions for undamaged SNF required by the applicable
608 regulations.

609
610 Degradation. Any change in the properties of a material that adversely affects the behavior of
611 that material; adverse alteration (ASTM C1174-97).

612
613 Design Bases. The information that identifies the specific functions to be performed by a
614 structure, system, or component (e.g., spent fuel storage cask) and the specific values or
615 ranges of values chosen for controlling parameters as reference bounds for design.

616
617 Design Earthquake. The design earthquake ground motion for a site where a cask system may
618 be used that is determined in accordance with 72.102 or 72.103.

619
620 Design Event (I, II, III, or IV). Conditions and events as defined and used for an independent
621 spent fuel storage installation in ANSI/ANS 57.9.

622
623 Double Contingency Principle. A design principle requiring that at least two unlikely,
624 independent, and concurrent or sequential changes in conditions essential to nuclear criticality
625 safety must occur before a criticality accident is possible (10 CFR 72.124(a)).

626
627 Exclusion Area. At a nuclear reactor site, the area surrounding the reactor in which the reactor
628 licensee has the authority to determine all activities including exclusion or removal of personnel
629 and property from the area. This area may be traversed by a highway, railroad, or waterway
630 provided these are not so close to the facility as to interfere with normal operations of the
631 facility, and provided appropriate and effective arrangements are made to control traffic on the
632 highway, railroad, or waterway, in case of emergency, to protect the public health and safety.
633 Residence within the exclusion area shall normally be prohibited. In any event, residents shall
634 be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor

635 may be permitted in an exclusion area under appropriate limitations, provided that no significant
636 hazards to the public health and safety will result (10 CFR 50.2).

637
638 Gray (Gy). The Standard International unit of absorbed dose. 1 Gy is equal to 100 rad.
639

640 Hard Receiving Surface. For a horizontal or vertical drop, need not be an unyielding surface;
641 rather, the receiving surface may be modeled as a reinforced concrete pad on engineered fill.
642

643 High Burnup Fuel. Spent nuclear fuel with burnups (see “Burnup”) generally exceeding
644 45 GWd/MTU.
645

646 Hoop Stress. The tensile stress in the cladding wall in the circumferential orientation.
647

648 Important Confinement Features. See “important to safety.”
649

650 Important to Safety, “Important to Nuclear Safety,” or “Structures, Systems, and Components
651 Important to Safety.” Those features of a dry storage system that have one or more of the
652 following functions: (1) maintain the conditions required to store spent nuclear fuel safely;
653 (2) prevent damage to the spent nuclear fuel cask during handling or storage; or (3) provide
654 reasonable assurance that spent nuclear fuel can be received, handled, containerized, stored,
655 and retrieved without undue risk to the health and safety of the public. ANSI/ANS 57.9 uses the
656 term “important confinement features”; however, NRC does not find this term acceptable. Per
657 Regulatory Guide 3.60, *Design of an Independent Spent Fuel Storage Installation (Dry Storage)*,
658 “important to safety” should be substituted for “important confinement features” in the standard.
659

660 Interim Staff Guidance (ISG). Supplemental information that clarifies important aspects of
661 regulatory requirements. An ISG provides NRC review guidance to NRC Staff in a timely
662 manner until standard review plans are revised accordingly.
663

664 Low Burnup Fuel. Spent nuclear fuel with burnups (see “Burnup”) generally less than
665 45 GWd/MTU.
666

667 Margin of Safety, or MofS. This term may be defined as being identical to a factor of safety,
668 $f.s. = \text{capacity/demand}$ (with minimum acceptable $\text{MofS} \geq 1$), or as a true margin, where
669 $\text{MofS} = f.s. - 1$ ($\text{capacity/demand} - 1$) (with minimum acceptable $\text{MofS} \geq 0.0$).
670

671 Misloading. The placement in a cask of spent nuclear fuel in a configuration not supported by
672 the cask’s design basis or technical specifications. Also, the placement in a cask of spent
673 nuclear fuel with characteristics that do not meet the characteristics of the cask’s allowable
674 contents.
675

676 Monitoring. Testing and data collection to determine the status of a dry storage system and to
677 verify the continued efficacy of the system on the basis of measurements of specified
678 parameters including temperature, radiation, and functionality and/or characteristics of
679 components of the system. With respect to radiation, per 10 CFR 20.1003, monitoring means
680 the measurement of radiation levels, concentrations, surface area concentrations or quantities
681 of radioactive material, and the use of the results of these measurements to evaluate potential
682 exposures and doses.
683

684 Neutron Absorber. Also known as “poison.” Materials that have high absorption cross section
685 and are used to absorb neutrons to make a fission system less reactive. They are used to

686 ensure subcriticality during normal/offnormal/accident-level conditions in containers for storing
687 and transporting fissile materials.

688
689 Nondestructive Examination (NDE). Testing, examination, and/or inspection of a component
690 that does not affect the functionality and performance of the component. NDE can be broadly
691 divided into three categories: visual, surface, and volumetric examinations. Additional
692 information may be found in the ASME B&PV Code, Section V, *Nondestructive Examination*,
693 Appendix A.

694
695 NDE-related terms in order of increasing severity:

696
697 Discontinuity: An interruption in the normal physical structure of a material.
698 Discontinuities may be unintentional (such as those formed inadvertently
699 during the fabrication process) or intentional (such as a drilled hole).

700
701 Indication: Sign of a discontinuity observed when using an NDE method.

702
703 Flaw: An imperfection in an item or material which may or may not be harmful.

704
705 Defect: A flaw that, due to its size, shape, orientation, location, or other
706 properties, is rejectable to the applicable construction code. Defects may
707 be detrimental to the intended service of a component and the component
708 must be repaired or replaced.

709
710 Common NDE examination methods include:

711
712 LT leak testing
713 MT magnetic particle examination
714 PT liquid penetrant examination
715 RT radiographic examination
716 UT ultrasonic examination
717 VT visual examination

718
719 Non-Fuel Hardware. Hardware that is not an integral part of a fuel assembly. Burnable Poison
720 Rod Assembly (BPRA), Control Element Assembly (CEA), Thimble Plug Assembly (TPA), etc.
721 are typical non-fuel hardware.

722
723 Normal Events and Conditions. The maximum level of an event or condition expected to
724 routinely occur. The cask system is expected to remain fully functional and to experience no
725 temporary or permanent degradation from normal operations, events and conditions. Note:
726 Specific normal conditions to be addressed have been evaluated for each dry storage system
727 that has received a Certificate of Compliance and are documented in a safety analysis report for
728 that system.

729
730 Off-Normal Events or Conditions. The maximum level of an event or condition that although not
731 occurring regularly can be expected to occur with moderate frequency and for which there is a
732 corresponding maximum specified resistance, limit of response, or requirement for a given level
733 of continuing capability. "Off-Normal" events and conditions are similar to "Design Event II" of
734 ANSI/ANS 57.9. An independent spent fuel storage installation structure, system, or component
735 is expected to experience off-normal events and conditions without permanent deformation or

736 degradation of capability to perform its full function (although operations may be suspended or
737 curtailed during off-normal conditions) over the full license period.

738
739 Preferential Loading. A configuration of spent nuclear fuel assemblies within a dry storage
740 system that is used as a method of controlling thermal conditions.

741
742 Qualification Test. A test, or series of tests, that is conducted at least once for a given
743 manufacturing process and set of material specifications to demonstrate the quality and
744 durability of the component such as neutron absorber product over its licensed service life.

745
746 Rad. The unit of absorbed dose. 1 rad is equal to the absorption of 100 ergs per gram.

747
748 Ready Retrievability. Capability to return the stored radioactive material to a safe condition
749 without the release of radioactive materials to the environment or radiation exposures in excess
750 of the limits defined by 10 CFR Part 20 [10 CFR 72.122(h)(5)]. An independent spent fuel
751 storage installation must be designed to allow ready retrieval of the stored spent nuclear fuel for
752 compliance with 10 CFR 72.122(l).

753
754 Real Individual. A person who is not a nuclear worker and who is at or beyond the controlled
755 area of an independent spent fuel storage installation, a nuclear power plant, or other nuclear
756 facility. For example, a real individual may be anyone living, working, or recreating close to the
757 facility for a significant portion of the year.

758
759 Reasonable Assurance. NRC staff base their decisions on the adequacy of a dry storage
760 system design to protect public health and safety on a variety of factors including: technical
761 evaluations, test and operational data, compliance with NRC requirements, and insights from
762 operational safety events.

763
764 Rem. The special unit of any of the quantities expressed as dose equivalent. See “Sievert” for
765 the unit conversion. The dose equivalent in rem is equal to the absorbed dose in rad multiplied
766 by the quality factor associated with a type (beta, gamma and neutron) of radiation (10 CFR
767 20.1004).

768
769 Restricted Area. An area to which access is limited by the licensee for the purpose of protecting
770 individuals against undue risks from exposure to radiation and radioactive materials. Restricted
771 area does not include areas used as residential quarters, but separate rooms in a residential
772 building may be set apart as a restricted area (10 CFR 20.1003).

773
774 Safety Analysis Report (SAR). In the context of this standard review plan, the report submitted
775 to the NRC staff by a certificate applicant to present information related to the design of a dry
776 storage system. This document provides the justification and analyses to demonstrate that the
777 design meets the requirements and acceptance criteria.

778
779 Safety Evaluation Report (SER). In the context of this standard review plan, the report prepared
780 by the NRC staff to present findings and recommendations relating to the acceptability of an
781 applicant’s safety analysis and other required documents submitted as part of a certificate
782 application. The SER also identifies the bases for those recommendations and the
783 recommended technical specifications (“operating controls and limits” or “conditions of use”).

784

785 Safety Functions. The functions that dry storage system structures, systems, and components
786 important to safety are designed to maintain include:

- 787
- 788 • Protection against environmental conditions,
 - 789 • Content Temperature Control,
 - 790 • Radiation Shielding,
 - 791 • Containment,
 - 792 • Sub-criticality control,
 - 793 • Retrievability.
- 794

795 Sievert (Sv). The Standard International unit of any of the quantities expressed as dose
796 equivalent. 1 Sv equals 100 rem. The dose equivalent in sieverts equals the absorbed dose in
797 grays multiplied by the quality factor (10 CFR 20.1004).

798

799 Spent Nuclear Fuel, (SNF). Nuclear fuel that has been withdrawn from a nuclear reactor
800 following irradiation, has undergone at least one year's decay since being used as a source of
801 energy in a power reactor, and has not been chemically separated into its constituent elements
802 by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source
803 material, and other radioactive materials associated with fuel assemblies (10 CFR 72.3).

804

805 Subcritical. The state at which the number of fission neutrons decreases with time and the
806 effective neutron multiplication factor (k_{eff}) is less than unity.

807

808 Supplemental Shielding. At an independent spent fuel storage installation, an engineered
809 radiation shield (principally neutron and gamma radiation) such as an earthen berm or concrete
810 wall. Supplemental shielding is classified as an important to safety component.

811

812 Total Effective Dose Equivalent (TEDE). The sum of the deep-dose equivalent for external
813 exposures and the committed effective dose equivalent for internal exposures
814 (10 CFR 20.1003).

815

816 Unrestricted Area. An area to which access is neither limited nor controlled by the licensee
817 (10 CFR 20.1003).

818

819 Validation. Demonstration of the validity of a computer code for use in a general area of
820 application by comparison of the code's calculational results with the measured results from a
821 variety of experiments spanning the area of intended applications.

822

823 Volume Percent. The percent of a mole of the material that is present in a volume equal to the
824 standard volume for the material as a gas; the volume occupied by one mole of the material as
825 a gas at standard conditions for gases (760 mm Hg [760 torr] pressure and 0°C [32°F]
826 temperature).

827

INTRODUCTION

This document is a Standard Review Plan (SRP). It is intended to provide guidance to the NRC staff conducting the safety review of an application for a spent fuel dry storage system (DSS) for facilities storing spent fuel under the general license authorized by 10 CFR 72.210. A general license authorizes a nuclear power plant licensee to store spent nuclear fuel (SNF) in NRC-approved casks at a site that is licensed to operate a power reactor under 10 CFR Part 50.

This SRP was developed to promote a consistent regulatory review of an application for a DSS, present a basis for the review scope, and identify acceptable approaches to meeting regulatory requirements.

This introduction provides an overview of the DSS and the Safety Analysis Report (SAR) review process, and assists the project manager in the coordination of the review effort. It is also designed to help individual technical reviewers understand how their specific review should be coordinated and integrated with other disciplines to produce a complete Safety Evaluation Report (SER).

This 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 Division of Spent Fuel Storage and Transportation (SFST). Comments, suggestions for improvement, and notices of errors or omissions will be considered by and should be sent to the Director, Division of Spent Fuel Storage and Transportation, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Use of Dry Storage Systems

In accordance with the requirements set forth in 10 CFR 72.212, a DSS may be used to store SNF in an independent spent fuel storage installation (ISFSI) under a general license. 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 using NRC-approved cask designs under the provisions of the general license.

The DSS safety review is primarily based on the information provided by an applicant, or cask vendor, in a SAR. Section 72.230 of 10 CFR Part 72 requires inclusion of a SAR in each application for approval of SNF cask storage design. Before submitting a SAR, an applicant should have designed the DSS considering as-low-as-is-reasonably-achievable (ALARA) principles for radiation protection and analyzed it 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 on the design and operational features and their associated technical bases. The reviewers need to understand the design and operational features and their technical bases, including but not limited to the selection of materials and geometries, mathematical models and equations used, computer models and calculated results in order to be able to draw conclusions that the storage cask is acceptable for use.

Technical Review Oversight

Cask designers are responsible for the safety of the cask design, and the cask users are responsible for safely operating the cask system at Part 50 reactor sites and complying with appropriate safety regulations. The mission of the regulator is to license and regulate the use of

879 each DSS and ensure adequate protection of public health and safety. The value of the NRC
880 review team is its independent expertise in identifying and resolving potential design or
881 operational deficiencies; potential analytical errors; significant uncertainties in novel design
882 approaches; or other non-compliance problems. If otherwise left unchecked by the designer,
883 user and regulator, these issues could potentially lead to the unsafe or non-compliant use of the
884 DSS.

885
886 Several considerations may influence the depth and rigor that is needed for a reasonable
887 assurance determination of both safety and compliance. These include the novelty of the
888 design (as compared to existing designs); safety margins; operational experience; defense-in-
889 depth, and the relative risks that have been identified for normal operations and potential
890 accident conditions. Consideration should also be given to the design parameters and
891 methodology approved in the SAR and their possible use in subsequent 10CFR 72.48(c)
892 changes to the design or procedures by the licensee or certificate holder. Any aspect of the
893 design or procedures that the NRC determines should not be changed by either the certificate
894 holder or general licensee, without prior NRC approval, must be placed in the CoC conditions or
895 in the attached technical specifications.

896
897 As described further below, each review procedure is prioritized using a graded approach that
898 factored in many of these considerations for a typical review. The prioritization was developed
899 with the expertise of NRC reviewers within each discipline, who have several years of regulatory
900 experience with the current fleet of certified spent fuel storage cask designs. These priorities
901 are intended to serve as a guidepost to the depth and rigor that is expected for a typical review;
902 but should not be treated as absolutes for every case. It is the responsibility of the individual
903 reviewer to assess the design and determine the ultimate rigor needed to make a safety
904 determination, with reasonable assurance, in each review area. In other words, reviewers
905 should consistently apply these review procedures for each case, but may need to adjust the
906 scope of review in some areas based on safety margins, operational experience, defense-in-
907 depth considerations, design novelty, or other issues that are unique to each proposed design.
908

909 **Review Process**

910
911 The purpose of the staff review is to evaluate the proposed cask design, contents and
912 operations, and provide regulatory confirmation of reasonable assurance of safe design and
913 construction of the cask.

914
915 The reviews are performed by project management and technical review staff with expertise in
916 the technical discipline areas described in the review plan. Due to the complexity of the
917 technical information in the application, coordination among the different disciplines is important
918 to ensure a consistent, uniform, and quality review. As described in the flow charts of each
919 chapter, technical issues can overlap between the disciplines and many rely on input from other
920 areas.

921
922 This SRP is guidance meant to be used in unison with the current ISGs. ISGs provide guidance
923 concerning specific, important issues that either are not currently addressed in the SRP or need
924 clarification beyond that in the present SRP text and may delineate specific review procedures.
925 For this reason, the staff should be familiar with ISGs that may supersede this guidance and
926 these new ISGs should be used together with this SRP in the review of a DSS application.
927 ISGs may be discontinued if they are fully incorporated into all applicable regulatory guidance
928 documents. Appendix C lists the ISGs from 1 to 22, and identifies which ones have been
929 incorporated in this revision of the SRP.

930
931 The staff may consult the SERs of previous CoC amendments, if reviewing an amendment to a
932 currently approved design, as well as the SERs for approved systems of similar design to
933 understand past NRC determinations regarding analyses affecting or similar to those in the
934 application under review.

935
936 In case the reviewer finds that the information provided in the SAR is not properly justified, the
937 reviewer may develop and then forward to the applicant questions requesting clarification of
938 technical issues via a Request for Additional Information (RAI). The applicant's response to the
939 RAI should be reviewed for accuracy as well as the need to update the applicant's SAR. The
940 RAI process is repeated as necessary, consistent with NRC's in-office instructions, until the
941 application is deemed technically acceptable, or until the application review is terminated by the
942 NRC or withdrawn by the applicant.

943
944 Once the technical review is complete, a draft SER is written that summarizes the results of the
945 review and the cognizant NRC Project Manager approves the SER. If the NRC intends to
946 approve the application, the staff prepares *Federal Register* notices for a direct final rule and a
947 companion proposed rule. The rulemaking notices identify the ADAMS numbers for the draft
948 CoC, TSs and SER. During the rulemaking process, stakeholders and members of the public
949 are allowed to comment on the draft CoC, TSs and SER. After addressing and responding to
950 any public comments, the NRC staff modifies the proposed CoC, TS and preliminary SER, if
951 necessary, and issues the Final CoC, TS, and SER. The rulemaking adds the CoC, or in the
952 case of an amendment to an existing CoC, the CoC amendment, to the list of approved cask
953 designs in 10 CFR 72.214.

954
955 **Safety Evaluation Report and Content**

956
957 The results of a SAR review are documented in an SER. The final determination of the
958 organization of an SER is determined by the review project manager, but the SER typically is
959 organized in the same manner as this SRP and contains the following information:

- 960
- 961 • A general description of the system, operational features, and SNF
962 specifications.
 - 963
 - 964 • A summary of the approach used by the applicant to demonstrate compliance
965 with the regulations, and a description of the reviews that the staff performed to
966 confirm compliance.
 - 967
 - 968 • Comparison of systems, components, analyses, data, or other information
969 important in the review analysis to the acceptance criteria, in addition to,
970 conclusions regarding the acceptability, suitability, or appropriateness that this
971 information provides reasonable assurance the acceptance criteria has been
972 met.
 - 973
 - 974 • Summary of aspects of the review that were selected or emphasized; matters
975 that were modified by the applicant: aspects of the cask's design that deviates
976 from the criteria stated in the SRP; and the bases for any deviations from the
977 SRP.
 - 978
 - 979 • Summary statements for evaluation findings at the end of each chapter.
 - 980

981 **Content of SRP**

982

983 Each chapter of the SRP is organized into the following sections:

984

- 985 • Review Objective
- 986 • Areas of Review
- 987 • Regulatory Requirements
- 988 • Acceptance Criteria
- 989 • Review Procedures
- 990 • Evaluation Findings

991

992 Review Objective. This section provides the purpose and scope of the review and establishes
993 the major review objectives for the chapter. The reviewer should obtain reasonable assurance
994 during the review that the objectives are met. It also discusses the information needed or
995 coordination expected from reviewers of other SAR chapters to complete the subject technical
996 review.

997

998 Areas of Review. This section describes the systems, components, analyses, data, or other
999 information and their sequence in the discussion of acceptance criteria and review procedures
1000 sections of each chapter.

1001

1002 Regulatory Requirements. This section summarizes the regulatory requirements from
1003 10 CFR Part 72 pertaining to the given SAR section. This list is not all inclusive (e.g., some
1004 parts of the regulations, such as 10 CFR Part 20, are assumed to apply to all chapters of the
1005 SAR). 10 CFR Part 72 sections applicable to a DSS are listed in 10 CFR 72.13(d). In addition,
1006 10 CFR 72.13(c) is important to the applicant to ensure that the general licensee does not
1007 violate those conditions. The reviewer should read the complete language of the current
1008 version of 10 CFR Part 72 to determine the proper set of regulations for the section being
1009 reviewed.

1010

1011 Acceptance Criteria. This section addresses the design criteria and in some cases specific
1012 analytical methods that NRC staff reviewers have found to be acceptable for meeting regulatory
1013 requirements, specified in 10 CFR Part 72, that apply to the given SAR chapter. The
1014 acceptance criteria are organized in accordance with the review areas established in Section 2
1015 of the specific chapter and identify the type and level of information that should be in the
1016 application.

1017

1018 These acceptance criteria typically set forth the solutions and approaches that staff reviewers
1019 have previously determined to be acceptable in addressing a specific safety concern or design
1020 area that is important to safety. These solutions and approaches are discussed in the SRP so
1021 that staff reviewers can implement consistent and well-understood positions as similar safety
1022 issues arise in future cases. These solutions and approaches are acceptable to the staff, but
1023 they are not the only possible solutions and approaches.

1024

1025 Substantial staff time and effort has gone into developing these acceptance criteria.
1026 Consequently, a corresponding amount of time and effort may be required to review and accept
1027 new or different solutions and approaches. Thus, applicants proposing solutions and
1028 approaches to new safety issues or analytical techniques other than those described in the SRP
1029 may experience longer review times and more extensive staff questioning in these areas. An
1030 alternative for the applicant is to propose new methods on a generic basis, apart from a specific
1031 license application. Such an alternative proposal could consist of a submittal of a Topical Safety

1032 Analysis Report (TSAR). This type of application could form the basis for either a change in the
1033 staff interpretation of the regulatory requirements or support a request for rulemaking to change
1034 the requirements themselves.

1035
1036 Review Procedures.
1037

1038 This section presents a general approach that reviewers typically follow to establish reasonable
1039 assurance that the applicable acceptance criteria have been met. As an aid to the reviewer, this
1040 section may also provide information on what has been found acceptable in past reviews.
1041 Standards that have been found acceptable in specific licensing reviews, or are desirable, but
1042 not specifically identified in existing regulatory documents, are identified in this section. Since
1043 many of the reviews are interdisciplinary, the reviewer should coordinate with other reviewers,
1044 as necessary, for identification of issues in other SAR chapters.

1045
1046 Each review procedure has been assigned a HIGH, MEDIUM or LOW priority, following
1047 application of the prioritization process described in Appendix B. These priorities are intended
1048 to provide guidance to the reviewer regarding the relative level of effort typically applied in
1049 implementing each procedure. As previously discussed, unique aspects of an application may
1050 result in an adjustment to the scope of review in a specific technical area. Specifically, the
1051 following can be used as general guidance on the implications of the priorities for the staff
1052 review:

1053
1054 **HIGH** priority means the NRC staff review should ensure all items in the applicant's
1055 submittal are complete and correct as specified in the review procedure. This
1056 represents the most comprehensive review where many of the analytical methods,
1057 assumptions, and supporting references are evaluated. The reviewer may need to
1058 perform independent confirmatory analysis to validate the results of the safety analysis
1059 calculations. It is expected a reviewer would spend approximately 60 percent of his or
1060 her review time focused on the high priority review procedures.

1061
1062 **MEDIUM** priority means the NRC staff should review the applicant's submittal for
1063 completeness and correctness in key areas. This represents a review in which key
1064 analytical methods, key assumptions, and key supporting references are checked and
1065 evaluated. It is expected a reviewer would spend approximately 30 percent of his or her
1066 review time focused on the medium priority review procedures.

1067
1068 **LOW** priority means the NRC staff review should ensure that the applicant's submittal
1069 contains all of the requested information. A limited review of selected portions of the
1070 application for correctness would be performed. Given its relative significance, the
1071 reviewer should generally consider the applicant's analysis to be complete and accurate
1072 and forego independent confirmation, unless there is a reason to believe otherwise.
1073 However, if a problem is detected, the reviewer must thoroughly evaluate and resolve
1074 the issue. It is expected a reviewer would spend approximately 10 percent of his or her
1075 review time focused on the low priority areas.

1076
1077 The risk-informed procedures are intended to ensure that reviewers focus most of their effort on
1078 the areas considered to have the greatest impact on safety and compliance with regulatory
1079 limits. While some issues could possibly escape detection and resolution through this audit
1080 review, they would be of lower regulatory significance. It is important to remember that the
1081 priority designations were developed on a generic basis and may need to be adjusted
1082 depending upon the characteristics of specific applications. It is the responsibility of the

1083 individual reviewer to assess the design and determine the ultimate rigor needed to make a
1084 safety determination, with reasonable assurance, in each review area.

1085
1086 Finally it should be noted that a low or medium priority review procedure does not mean an
1087 application is exempted from any associated regulatory requirement, design requirement, or
1088 safety analyses that is expected within the review objectives and acceptance criteria in this
1089 SRP.

1090
1091 Evaluation Findings. This section provides example summary statements for evaluation
1092 findings to be incorporated into the SER for each area of review. The evaluation findings are
1093 prepared by the reviewer based on the satisfaction of the regulatory requirements. The findings
1094 are published in the SER.

GENERAL INFORMATION EVALUATION

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1.1 Review Objective

The purpose of reviewing the general description of the Spent Fuel dry storage system (DSS) is to ensure that the applicant has provided a non-proprietary description, or overview, that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.2 Areas of Review

The general description should be reviewed by all reviewers, regardless of their specific review assignments, to obtain a basic understanding of the DSS, its components, and the protections afforded for the health and safety of the public. Because much of the information relevant to this initial aspect of the DSS review is presented in more detail in other chapters of this SRP, this chapter focuses on familiarization with the DSS and consistency of the DSS general description with the remaining chapters of the safety analysis report (SAR). The SAR should be reviewed for adequacy of the DSS and DSS support system descriptions and drawings. Areas of review addressed in this chapter include the following:

DSS Description and Operational Features

Drawings

DSS Contents

Qualifications of the Applicant

Quality Assurance

Consideration of 10 CFR Part 71 Requirements Regarding Transportation

1.3 Regulatory Requirements

This section presents a summary matrix of the portions of U.S. Code of Federal Regulations (CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste and Reactor-Related Greater Than Class C Waste," Title 10, "Energy" (10 CFR Part 72) that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should read the exact regulatory language. Table 1-1 matches the relevant regulatory requirements associated with this chapter to the areas of review.

Table 1-1 Relationship of Regulations and Areas of Review						
Areas of Review	10 CFR Part 72 Regulations					
	72.2(a)(1), (b)	72.122 (a), (h)(1)	72.140 (c)(2)	72.230 (a)	72.230 (b)	72.236(a), (c), (h),(m)
DSS Description and Operational Features	•	•		•		
Drawings	•			•		
DSS Contents	•					•
Qualifications of the Applicant	•					
Quality Assurance	•		•			
Consideration of 10 CFR Part 71 Certified Transportation Cask System Requirements	•				•	•

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1.4 Acceptance Criteria

This section identifies the acceptance criteria for the material provided in the introduction. This initial aspect of the DSS review should contain sufficient information to allow all reviewers, regardless of their specific review assignments, to understand the principal functions and design features of the DSS.

1.4.1 DSS Description and Operational Features

The application should contain a broad overview and a general, non-proprietary description (including engineering drawings, sketches, and illustrations) of the DSS. This information should clearly identify the functions of all principal components and principal auxiliary equipment, and provide a list of those components classified as being “important to safety.” Important aspects from all of the disciplinary areas should be summarized. If there are several versions of the cask because of design limitations of nuclear power plants and ISFSIs, the differences between the versions should be delineated. Typical operational sequences for loading and unloading procedures should be described.

If the potential exists that the DSS will be used to store damaged fuel, the SAR should include a discussion of how the sub-criticality requirement of 10 CFR 72.236(c) and the wet or dry loading and unloading requirements of 10 CFR 72.236(h) will be maintained. Additionally, a discussion should be included of the planned decommissioning process.

The reviewer should verify that any documents submitted to the NRC in other applications and incorporated in whole or in part have been tabulated, and a summary has been included in the appropriate section of the SAR.

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1.4.2 Drawings

Drawings should be included in the first chapter of the SAR. The drawings should contain sufficient detail to allow the reviewer to understand the operation of the DSS and any special equipment used for loading, unloading, transportation, or long-term storage of the DSS. Also, the drawings should provide enough detail to allow the reviewer the option of developing an analysis model for confirmatory calculations.

Ideally, the drawings should be non-proprietary. However, in some cases, the applicant may request to have certain specific portions of the drawings classified as proprietary. Reviewers should note that any drawings relied on as the technical basis for adding the DSS design to the list of approved DSSs 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 2.390].

Any request for withholding from public disclosure subject to the provisions of 10 CFR 2.390 should be accompanied by an affidavit and must include information to support the claim that the material is proprietary. The NRC Project Manager will develop and administer public disclosure determinations, and the Office of the General Counsel will review them for compliance with the requirements of 10 CFR 2.390.

1.4.3 DSS Contents

The reviewer should ensure specifications are provided for the contents expected to be stored in the DSS (normally spent nuclear fuel [SNF]). These specifications may include, but not be limited to, type of SNF (i.e., boiling-water reactor [BWR], pressurized-water reactor [PWR], or both); number of SNF assemblies the cask can accommodate; maximum allowable enrichment of the fuel before any irradiation; burnup (i.e., MWd/MTU); minimum acceptable cooling time of the SNF before storage in the DSS (e.g., aged at least 1 year); maximum heat designed to be dissipated; maximum SNF loading limit; condition of the SNF (i.e., intact, undamaged, damaged, etc.); weight and nature of non-SNF contents; and inert atmosphere requirements.

1.4.4 Quality Assurance

Reviewers should verify that the application describes the proposed quality assurance (QA) program and cites the applicable implementing procedures. This description should satisfy all requirements of 10 CFR Part 72, Subpart G. A detailed review of the QA program to be described in the SAR is presented in Chapter 14, "Quality Assurance Evaluation," of this SRP.

1.4.5 Consideration of 10 CFR Part 71 Requirements Regarding Transportation

If the DSS has previously been evaluated for use as a transportation cask, the submittal should include the Part 71 Certificate of Compliance (CoC) and associated documents in accordance with 10 CFR 72.230(b). If application for storage is submitted, the transportability, per 10 CFR 72.236(m) should be addressed. (See Section 1.5.5).

1.5 Review Procedures

Figure 1-1 presents an overview of the evaluation process and a complete bulleted listing of pertinent information for each chapter. Figure 1-1 and the corresponding figures in each

1211 chapter of this Standard Review Plan (SRP) provide a means to coordinate the review among
1212 the NRC staff disciplines.

1213
1214 Regulatory requirements of 10 CFR Part 72 applicable to the general description review are
1215 delineated in the following subsections. Since the review of the General Description of the SAR
1216 is interdisciplinary, the reviewer should coordinate with other reviewers (e.g., structural, thermal,
1217 shielding, criticality, materials), as necessary, for identification of related issues.

1218
1219 **1.5.1 DSS Description and Operational Features (MEDIUM Priority)**

1220
1221 Reviewers should verify that the application provides a broad overview of the DSS design that is
1222 non-proprietary and may be used as a tool to familiarize interested parties with the features of
1223 the proposed DSS. This description should present the principal characteristics of the DSS
1224 including its dimensions, weight, and construction materials. In addition, the description should
1225 clearly identify all components considered important to safety. Features such as the
1226 confinement vessel, fuel basket, valves, lids, seals, penetrations, trunnions, closure
1227 mechanisms, shielding safety features, criticality control features, impact limiters, and cask
1228 identification should be identified and described. A clear definition of the primary confinement
1229 system is particularly important. Special design features of the DSS such as a non-passive
1230 heat-removal system, neutron poisons or monitoring instrumentation should be discussed.

1231
1232 Sketches and diagrams found throughout the SAR should be compared with the detailed
1233 drawings presented in SAR Chapter 1, "General Information". If the application includes
1234 proprietary drawings and descriptions that will remain proprietary upon approval of the license
1235 or certificate, the sketches, drawings, and diagrams that provide the general description and
1236 operational features need not show the proprietary features. This may be achieved by depicting
1237 less detail or by illustrating generic components that fulfill the design function. However, these
1238 representations should show the operational concept and features important to safety in
1239 sufficient detail to form an acceptable basis for public review and comment.

1240
1241 In addition to information on a single DSS, the application should describe any limitations on the
1242 arrangement of DSS arrays. For a particular DSS, these limitations may include the minimum
1243 spacing between the casks, maximum density of casks in an array, and/or total number of casks
1244 or amount of SNF that may be stored at a single ISFSI. The acceptable limitations should be
1245 included among the technical specifications in the Safety Evaluation Report (SER) (see Chapter
1246 13, "Technical Specifications and Operating Controls and Limits Evaluation," of this SRP). For a
1247 DSS such as those with metal confinement vessels stored in a concrete vault, information on
1248 the configuration of vault compartments and horizontal/vertical arrangement is necessary. The
1249 operational sequences for loading and unloading the cask should be described.

1250
1251 Damaged fuel may require canning for storage and transportation. The purpose of canning is to
1252 confine gross fuel particles to a known, subcritical volume during off-normal and accident
1253 conditions, and to facilitate handling and ready retrievability. Therefore, the reviewer should
1254 verify that a description of how damaged fuel would be canned, the characteristics of the can,
1255 and the means in which the can would be placed in the cask and retrieved is in the application.
1256

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1259

Chapter 1 – General Information Evaluation			
DSS Design Information	DSS Description		Compliance with 10 CFR Part 72
<ul style="list-style-type: none"> Purpose of Application Quality Assurance Program Proposed Use and Contents of DSS DSS Category, Type, and Model Number Thermal Loading Fabrication and Welding Criteria 	<ul style="list-style-type: none"> Cask and Overpack Operating Features Contents of DSS 		<ul style="list-style-type: none"> Condition of DSS after Testing per Applicable Portions of §72.122 Structural, Thermal, Confinement, Shielding, Criticality Requirements, and Materials Operating Procedures, Acceptance Tests, and Maintenance
→	Chapter 2 – Principal Design Criteria Evaluation		
	<ul style="list-style-type: none"> Items Important to Safety DSS Design Basis 	<ul style="list-style-type: none"> Spent Fuel Design Basis External Conditions 	
→	Chapter 3 – Structural Evaluation		
	<ul style="list-style-type: none"> Internal and External Structure Codes and Standards 	<ul style="list-style-type: none"> Component Materials Dimensions and Weights 	
→	Chapter 4 – Thermal Evaluation		
	<ul style="list-style-type: none"> Spent Fuel Cladding Configuration 	<ul style="list-style-type: none"> Component Materials Dimensions 	<ul style="list-style-type: none"> Decay Heat Heat Dissipation
→	Chapter 5 – Containment Evaluation		
	<ul style="list-style-type: none"> Dimensions Component Materials 	<ul style="list-style-type: none"> Containment Boundary DSS Contents 	<ul style="list-style-type: none"> Allowable Leak Rate Accident Conditions Penetrations
→	Chapter 6 – Shielding Evaluation		
	<ul style="list-style-type: none"> Dimensions Configuration 	<ul style="list-style-type: none"> Component Materials Content Limits 	
→	Chapter 7 – Criticality Evaluation		
	<ul style="list-style-type: none"> Fissile Content Materials Dimensions and Tolerance 	<ul style="list-style-type: none"> Component Materials Neutron Poison Contents 	
→	Chapter 8 – Materials Evaluation		
	<ul style="list-style-type: none"> Material Selection 	<ul style="list-style-type: none"> Corrosion 	<ul style="list-style-type: none"> Cladding Integrity
→	Chapter 9 – Operating Procedures Evaluation		
	<ul style="list-style-type: none"> General Restrictions Operational Sequences 		
→	Chapter 10 – Acceptance Tests and Maintenance Program Evaluation		
	<ul style="list-style-type: none"> Codes and Standards Dimensions and Tolerances 	<ul style="list-style-type: none"> Fabrication Materials Contents 	<ul style="list-style-type: none"> Maintenance Tasks Instrumentation
→	Chapter 11– Radiation Protection Evaluation		
	<ul style="list-style-type: none"> ALARA Radiation Protection Features 	<ul style="list-style-type: none"> Dose Assessment Health Physics Program 	
→	Chapter 12 – Accident Analysis Evaluation		
	<ul style="list-style-type: none"> Accident Identification DSS Performance Analysis 	<ul style="list-style-type: none"> Corrective Action Program 	
→	Chapter 13 – Technical Specifications and Operating Controls Evaluation		
	<ul style="list-style-type: none"> Operating, Monitoring, and Safety Limits Loading and Unloading 	<ul style="list-style-type: none"> Transport Surveillance 	
→	Chapter 14 – Quality Assurance Evaluation		
	<ul style="list-style-type: none"> Program Description National Standards 	<ul style="list-style-type: none"> Items Important to Safety Document Control 	

Figure 1-1 Overview of Safety Evaluation

1260 **1.5.2 Drawings (MEDIUM Priority)**

1261
1262 Drawings are usually presented in Chapter 1, "General Information" of the SAR. Reviewers
1263 should be familiar with NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package
1264 Approval." While NUREG/CR-5502 was written for transportation packages, the criteria in
1265 NUREG/CR-5502 for drawings can be applied to applications for storage casks.

1266
1267 Although some applications may contain drawings designated as "proprietary," reviewers should
1268 note that any drawings relied on as the technical basis for adding the DSS design to the "list of
1269 approved spent-fuel storage DSS" contained in Subpart K of 10 CFR 72 become part of the
1270 public record. Such drawings will not be treated as proprietary and will be made available to the
1271 public [10 CFR 2.390(a)]. Applicants may submit additional drawings showing greater detail to
1272 support their evaluations, and these may be exempted from the public record if they are not
1273 relied on by the staff as part of the technical basis for DSS design approval. The reviewer
1274 should verify that all structures, systems, and components (SSC) important to safety are
1275 sufficiently detailed to enable reviewers to evaluate their effectiveness. In addition, information
1276 on non-safety items may also be necessary to ensure they do not impede the safety systems.

1277
1278 Each reviewer should evaluate the level of detail furnished with the application. The drawings
1279 should specify those details of the cask design that affect its evaluation. Those design features
1280 that have a significant effect on safety if altered or modified, should be considered for inclusion
1281 into the technical specifications directly or by reference. If size reduction has rendered any
1282 information unclear or illegible, the Project Manager in the Division of Spent Fuel Storage and
1283 Transportation (SFST) should request that the applicant provide larger or full-size drawings.

1284
1285 Particular attention should be devoted to ensuring that dimensions, materials, and other details
1286 on the drawings are consistent with those described in both the text of the SAR and those used
1287 in supplementary analysis. The dimensions shown on the general arrangement drawing should
1288 specify the overall size of the cask and the location and configuration of the contents. All
1289 dimensions indicated on drawings should include tolerances that are consistent with the cask
1290 evaluation.

1291
1292 **1.5.3 DSS Contents (MEDIUM Priority)**

1293
1294 The application should present a general description of the contents proposed for storage in the
1295 DSS. Because a very detailed description of the proposed DSS contents or SNF is typically
1296 provided in Chapter 2, "Principal Design Criteria," of the SAR, the information presented in
1297 Chapter 1, "General Information" of the SAR is important only to the extent that it permits overall
1298 familiarization with the DSS. Key parameters for SNF include the type of fuel (i.e., PWR, BWR,
1299 or both), number of fuel assemblies, the radiation source terms associated with these fuel
1300 assemblies, preferential loading, and condition of the fuel assemblies (i.e., intact or
1301 consolidated). Chapter 1 may also include additional characteristics such as maximum burnup,
1302 initial enrichment, heat load, and cooling time as well as the assembly vendor and configuration
1303 (e.g., Westinghouse 17x17). These characteristics may also be repeated in Chapter 2. In
1304 addition, the cover gas, if any, should be identified.

1305
1306 If the applicant proposes the storage of damaged fuel or components that are associated with or
1307 integral to the fuel assembly that do not have an integral confinement boundary, the range of
1308 permissible conditions for the stored material should be defined. If the DSS system is intended
1309 to be used to store damaged fuel or components that are associated with or integral to the fuel
1310 assembly with an integral confinement boundary when placed in the confinement DSS, the

1311 possible range of conditions of the fuel or components should be stated. 10 CFR 72.122(h)(1)
1312 requires “canning” or use of other acceptable means for storing fuel with cladding that is not or
1313 may not remain intact and for unconsolidated assemblies (without intact cladding).
1314 10 CFR 72.236(c) requires the damaged fuel be maintained in a subcritical condition, while
1315 10 CFR 72.236(h) requires the damaged fuel to be compatible with wet or dry loading and
1316 unloading facilities. If damaged fuel is to be stored, the application should address how the
1317 following basic requirements will be met:

- 1318
- 1319 • Maintain subcriticality;
- 1320 • Prevent unacceptable release of contained radioactive material;
- 1321 • Avoid excessive radiation dose rates and doses;
- 1322 • Maintain ready retrievability of the contents.
- 1323

1324 If the application requests approval to use the DSS system to store components that are
1325 associated with or integral to the fuel assembly (i.e., control spiders, burnable poison rod
1326 assemblies, control rod elements, thimble plugs, fission chambers, and primary and secondary
1327 neutron sources, or BWR channels that are an integral part of the fuel assembly that do not
1328 require special handling), the application should present summary descriptions of those
1329 components in Chapter 1, “General Information” of the SAR. The SFST staff has made a
1330 practice of carefully characterizing components as being “associated with or integral to” the fuel
1331 assembly because only those components listed above are acceptable at a geologic repository
1332 per 10 CFR 961.11, Appendix E, Section B.2. Components that are associated with or integral
1333 to the fuel assembly are reviewed in more detail as part of Chapter 2, “Principal Design Criteria
1334 Evaluation,” of this SRP. Also, if the components are degraded (e.g., the component does not
1335 provide adequate confinement under design basis conditions to contain radioactive gas or other
1336 dispersible radioactive materials), the application should describe the possible conditions and
1337 alternative confinement methods, if any.

1338

1339 **1.5.4 Quality Assurance Program (See Chapter 14 for Priority)**

1340

1341 The application should describe the proposed QA program, citing all implementing procedures
1342 in a manner that satisfies the 18 criteria defined in 10 CFR Part 72, Subpart G, “Quality
1343 Assurance” (10 CFR §§ 72.142-72.176). The description need only refer to procedures that
1344 implement the QA program, and these procedures need not be explicitly included in the
1345 application. The QA program should address design, fabrication, construction, testing,
1346 operation, and modification activities regarding the SSCs that are important to safety. The
1347 application should also discuss the activities to be performed under the QA program and how
1348 these activities will be controlled to ensure compliance with all of the requirements of Subpart G.
1349 These controls may be applied to the various activities using a graded approach as presented in
1350 NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage
1351 System Components According to Importance to Safety” (i.e., QA efforts expended for a given
1352 activity should be consistent with that activity's system classification and function).

1353

1354 Per 10 CFR 72.140(d), a QA program previously approved by the NRC and established,
1355 maintained, and executed for another DSS will be accepted as satisfying the requirements for a
1356 QA program for the purpose of this application. Additionally, previously approved QA programs
1357 that meet the requirements of Appendix B to 10CFR 50 or Subpart H to 10 CFR 71, will be
1358 acceptable provided they also meet the recordkeeping requirements of §72.174. Any reference
1359 to a previously approved QA program should identify the program by date of submittal to the
1360 NRC, docket number, and date of NRC approval. The reviewer should coordinate with the
1361 Chapter 14, “Quality Assurance Evaluation,” review of this SRP.

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1.5.5 Consideration of 10 CFR Part 71 Requirements (MEDIUM Priority)

Casks that have been certified for transportation of SNF under 10 CFR Part 71 may be approved for the storage of SNF under 10 CFR Part 72 provided the application contains:

- A copy of the CoC issued under 10 CFR Part 71,
- Copies of all drawings and other documents referenced in the 10 CFR Part 71 CoC, and
- Sufficient information in the SAR to demonstrate that the cask is suitable for the storage of SNF for at least 20 years [10 CFR 72.230(b)].

Because applications for dual-purpose certification under 10 CFR Parts 71 and 72 are sometimes submitted jointly, the final (approved) version of such documents may not be available at the time of initial DSS SAR submission. Nonetheless, applicable documentation of the Part 71 certification (or application), including questions and responses from the related review, should be 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 effort. The reviewer should verify that a process for promptly informing each of the review teams about DSS system design changes precipitated by any concurrent safety reviews has been identified by the applicant. Provisions for communicating these changes should be addressed by, and discussed with, the applicant. In addition, transportability of storage-only or dual purpose casks, per 10 CFR 72.236(m) should be addressed. The applicant should address how it is planning to address the transportation requirements. The reviewer should verify that such considerations have been made and described in the SAR, when the SAR and/or accompanying documentation indicate plans to use the cask system for transportation purposes.

1.6 Evaluation Findings

The evaluation findings are prepared by the reviewer on satisfaction of the regulatory requirements in Section 1.3. These statements should be similar to the following examples, if the documentation submitted with the application supports positive findings for each of the regulatory requirements (the finding number is for convenience in reference within the SRP and SER):

- F1.1 A general description and discussion of the DSS is presented in Section(s) of the SAR, with special attention to design and operating characteristics, unusual or novel design features, and principal considerations important to safety.
- F1.2 Drawings for SSCs important to safety are presented in Section _____ of the SAR. A listing of those drawings (including dates and revision numbers) that were relied upon as a basis for approval appears in Section _____ of the SER.
- F1.3 Specifications for the SNF to be stored in the DSS are provided in SAR Section _____. Additional details concerning these specifications are presented in Chapter _____ of both the SAR and SER.

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F1.4 The quality assurance program and implementing procedures are described in Section _____ of the SAR.

F1.5 The [DSS system designation] [has been/is/is not being] certified under 10 CFR Part 71 for use in transportation. A copy of the SAR and CoC issued under 10 CFR Part 71 is on file with the NRC under Docket No. _____ [if applicable].

A summary statement similar to the following should be made:

“The staff concludes that the information presented in Chapter 1, “General Information” of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, Regulatory Guide 3.61, and accepted practices.”

1429 **2 PRINCIPAL DESIGN CRITERIA EVALUATION**

1430
1431 **2.1 Review Objective**

1432
1433 The objective of evaluating the principal design criteria related to structures, systems, and
1434 components (SSCs) important to safety is to ensure that, in the view of the U.S. Nuclear
1435 Regulatory Commission (NRC) staff, the principal design criteria comply with the relevant
1436 general criteria established in U.S. Code of Federal Regulations (CFR) Part 72, "Licensing
1437 Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive
1438 Waste and Reactor-Related Greater Than Class C Waste," Title 10, "Energy" (10 CFR Part 72).
1439 Further guidance can be found in NUREG/CR-6407, "Classification of Transportation Packaging
1440 and Dry Spent Fuel Storage System Components According to Importance to Safety." Material
1441 provided in this chapter will form the basis for accepting the safety analysis report (SAR) for
1442 NRC staff review.

1443
1444 With regard to reviewing the principal design criteria, the applicant may take one of two
1445 approaches: (1) SAR Chapter 2, "Principle Design Criteria" may discuss these criteria in general
1446 terms with details provided in later sections or (2) SAR Chapter 2 may present detailed
1447 discussions of selected (or all) criteria. Past applicants have generally selected the latter
1448 approach. Subsequent chapters of this Standard Review Plan (SRP) provide detailed
1449 discussions of the design criteria applicable to each functional area (e.g., structural, thermal)
1450 without regard to those that may have been presented in SAR Chapter 2.

1451
1452 **2.2 Areas of Review**

1453
1454 The review of the principal design criteria should provide reasonable assurance that all design
1455 criteria are addressed in the SAR. The following areas of review have been adopted by the
1456 NRC staff:

1457 ***Structures, Systems, and Components Important to Safety***

1458
1459 ***Design Basis for Structures, Systems, and Components Important to Safety***

- 1460 Spent Nuclear Fuel (SNF) Specifications
- 1461 External Conditions

1462
1463 ***Design Criteria for Safety Protection Systems***

- 1464 General
- 1465 Structural
- 1466 Thermal
- 1467 Shielding/Confinement/Radiation Protection
- 1468 Criticality
- 1469 Material Selection
- 1470 Operating Procedures
- 1471 Acceptance Tests and Maintenance
- 1472 Decommissioning

1473
1474
1475 **2.3 Regulatory Requirements**

1476
1477 This section presents a summary matrix of the portions of U.S. Code of Federal Regulations
1478 (CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel,
1479 High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste" Title 10,

1480 “Energy” (10 CFR Part 72) that are relevant to the review areas addressed by this chapter. The
 1481 NRC staff reviewer should read the exact regulatory language. Table 2-1 matches the relevant
 1482 regulatory requirements associated with this chapter to the areas of review.
 1483

Table 2-1 Relationship of 10 CFR Part 72 Regulations and Areas of Review										
Areas of Review	10 CFR Part 72 Regulations									
	72.2 (a)(1)	72.104 (a), (b), (c)	72.106 (a), (b), (c)	72.122 (a), (b) (1)(2) (3), (c), (f)	72.122 (h)(1) (4)	72.122 (i), (l)	72.124 (a), (b)	72.126 (a)(1) (2)(3) (4)(5) (6)	72.236 (a), (b), (c), (d)	72.236 (e), (f), (g), (h), (i), (l), (m)
SSCs Important to Safety									•	
Design Bases for SSCs Important to Safety	•			•					•	
Design Criteria for Safety Protection Systems		•	•	•	•	•	•	•	•	•

1484
 1485 **2.4 Acceptance Criteria**
 1486

1487 The reviewer should verify that the applicant has provided either sufficient general or summary
 1488 discussions of the SSC design features and of both operational and accident conditions. This
 1489 demonstrates a clear and defensible case that they have met the design criteria. In evaluating
 1490 the principal design criteria related to DSS SSCs that are important to safety, reviewers should
 1491 seek to ensure that the given design fulfills the following acceptance criteria.
 1492

1493 **2.4.1 SSCs Important to Safety**
 1494

1495 The reviewer should verify that the applicant presents the general configuration of the DSS and
 1496 provides an overview of specific components and their intended functions. In addition, the
 1497 reviewer should ensure the applicant identifies those components deemed to be important to
 1498 safety and addresses the safety functions of these components in terms of how they meet the
 1499 general design criteria and regulatory requirements discussed above. Additional information
 1500 concerning specific functional requirements for individual DSS components is addressed in
 1501 subsequent chapters of this SRP.
 1502

1503 **2.4.2 Design Bases for SSCs Important to Safety**

1504
1505 Detailed descriptions of each of the items listed below are generally found in specific sections of
1506 the SAR. However, a brief description of these areas, including a summary of the analytical
1507 techniques used in the design process, should also be captured in Chapter 2, "Principle Design
1508 Criteria" of the SAR. This description gives reviewers a perspective of how specific DSS
1509 components interact to meet the regulatory requirements of 10 CFR Part 72. This discussion
1510 should be non-proprietary since it may be used to familiarize interested persons with the design
1511 features and bounding conditions of operation of a given DSS.

1512
1513 **2.4.2.1 SNF Specifications**

1514
1515 The range and types of SNF or other radioactive materials that the DSS is designed to store
1516 should be specified. In addition, these specifications should include, but are not limited to:

- 1517
1518 • The type of SNF (i.e., boiling-water reactor (BWR), pressurized-water reactor
1519 (PWR), or both),
- 1520
1521 • Cladding material,
- 1522
1523 • Maximum assembly uranium mass loading,
- 1524
1525 • Weights of the stored materials,
- 1526
1527 • Dimensions and configurations of the fuel,
- 1528
1529 • The identification and limits on amount and position of damaged fuel, if damaged
1530 fuel is to be stored, and the dimensions of the "damaged-fuel can,"
- 1531
1532 • Maximum allowable enrichment of the fuel before any irradiation for criticality
1533 safety and minimum enrichment for the shielding evaluation,
- 1534
1535 • Assigned Burnup Loading Value (i.e., MWd/MTU),
- 1536
1537 • Loading Curves for each set of licensing conditions if Burnup Credit is used
1538 (required minimum burnup versus enrichment curve),
- 1539
1540 • Operational history parameters (e.g., average in-core soluble boron
1541 concentration, average moderator temperature, etc.) if burnup credit is used,
- 1542
1543 • Minimum acceptable cooling time of the SNF before storage in the DSS,
- 1544
1545 • Maximum heat to be dissipated,
- 1546
1547 • Maximum number of SNF elements,
- 1548
1549 • Condition of the SNF (i.e., intact assembly, damaged fuel or consolidated fuel
1550 rods),
- 1551

- 1552 • Inerting atmosphere requirements and the maximum amount of fuel permitted for
1553 storage in the DSS.

1554
1555 For DSSs that will be used to store components that are associated with or integral to fuel
1556 assemblies (e.g., control rods and BWR fuel channels), the reviewer should ensure the
1557 applicant specifies the types and amounts of radionuclides, heat generation, and the relevant
1558 source strengths and radiation energy spectra permitted for storage in the DSS. For other
1559 radioactive materials to be stored with the SNF assemblies, the SAR should specify the
1560 following:

- 1561 • The design basis source term;
1562
1563 • The effects of gas generation on the cask internal pressure;
1564
1565 • The effects of the additional weight and length of the proposed material on
1566 structural and stability analyses;
1567
1568 • The impact of the added heat from these components, including the impact on
1569 heat transfer characteristics; and
1570
1571 • Credit for any negative reactivity from residual neutron absorbing material
1572 remaining in the control components.
1573

1574
1575 2.4.2.2 External Conditions
1576

1577 The SAR should define the bounding conditions under which the DSS is expected to operate.
1578 Such conditions include both normal and off-normal environmental conditions as well as
1579 accident conditions. In addition, the reviewer should verify that the applicant has considered the
1580 effects of natural events such as tornadoes, earthquakes, floods, and lightning strikes.

1581
1582 **2.4.3 Design Criteria for Safety Protection Systems**

1583
1584 2.4.3.1 General
1585

1586 The SAR should define an expected lifetime for the cask design. The minimum licensing period
1587 is 20 years. The reviewer should verify that the applicant has provided a brief description of the
1588 proposed quality assurance (QA) program, and applicable industry codes and standards, which
1589 will be applied to the design, fabrication, construction, and operation of the DSS. The applicant
1590 should also describe how the cask design reflects consideration of compatibility with removal
1591 from a reactor site, transportation, and ultimate disposition of the stored spent fuel.

1592
1593 In establishing normal and off-normal conditions applicable to the design criteria for DSS
1594 designs, applicants should account for actual facility operating conditions. Therefore, design
1595 considerations should reflect normal operational ranges, including any seasonal variations or
1596 effects.

1597
1598 2.4.3.2 Structural
1599

1600 The SAR should define how the DSS structural components are designed to accommodate
1601 combined normal, off-normal, and accident loads while preserving retrievability and protecting
1602 the DSS contents from significant structural degradation, criticality, and loss of confinement.

1603 This discussion is generally a summary of the analytical techniques and calculation results from
1604 the detailed analysis discussed in SAR Chapter 3, "Structural Evaluation," and should be
1605 presented in a non-proprietary form.

1606
1607 2.4.3.3 Thermal

1608
1609 The SAR should contain a general discussion of the proposed heat-removal systems, including
1610 the reliability and verifiability of such systems, and any associated limitations. All heat-removal
1611 systems should be passive and independent of intervening actions under normal and off-normal
1612 conditions.

1613
1614 2.4.3.4 Shielding/Confinement/Radiation Protection

1615
1616 The reviewer should ensure that the applicant describes those features of the cask that protect
1617 occupational workers and members of the public against direct radiation dosages and releases
1618 of radioactive material, and minimize the dose after any off-normal or accident-level conditions.

1619
1620 2.4.3.5 Criticality

1621
1622 The SAR should address the mechanisms and design features that enable the DSS to maintain
1623 SNF in a subcritical condition under normal, off-normal, and accident-level conditions.

1624
1625 2.4.3.6 Material Selection

1626
1627 The materials selected for the DSS must demonstrate adequate corrosion performance during
1628 normal operation, off-normal, and accident-level conditions in the environmental conditions of
1629 the ISFSI for the duration of the license.

1630
1631 The spent fuel cladding must be protected during storage against degradation that leads to
1632 gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during
1633 storage will not pose operational problems with respect to its removal from storage.

1634
1635 2.4.3.7 Operating Procedures

1636
1637 The reviewer should ensure that the applicant provides potential licensees with guidance
1638 regarding the content of normal, off-normal, and accident response procedures. Cautions
1639 regarding both loading, unloading, and other important procedures should be mentioned here.
1640 Retrievability should be provided for normal and off-normal conditions. Applicants may choose
1641 to provide model procedures to be used as aids in preparing detailed site-specific procedures.

1642
1643 2.4.3.8 Acceptance Tests and Maintenance

1644
1645 The reviewer should verify that the applicant identifies the general commitments and industry
1646 codes and standards used to derive acceptance, maintenance, and periodic surveillance tests
1647 used to verify the capability of DSS components to perform their designated functions. In
1648 addition, the reviewer should ensure the applicant discusses the methods used to assess the
1649 need for such tests with regard to specific components.

1650

1651 2.4.3.9 Decommissioning

1652

1653 Casks should be designed for ease of decontamination and eventual decommissioning. The
1654 reviewer should examine the SAR to ensure the applicant describes the features of the design
1655 that support these two activities.

1656

1657 **2.5 Review Procedures**

1658

1659 Chapter 2, "Principle Design Criteria" applies to all review disciplines. Figure 2-1 presents an
1660 overview of the evaluation process and may be used as a guide for the coordination of the
1661 review among review disciplines.

1662

1663 Reviewers for each section of the SAR should consider SAR Chapter 2 in combination with
1664 additional details presented later in the SAR. In this SRP, evaluations of design criteria
1665 applicable to each of the relevant chapters of the SAR are discussed in detail. Reviewers
1666 should coordinate the review of each chapter with the applicable disciplines to ensure that multi-
1667 disciplinary issues, which impact more than one chapter, have been addressed.

1668

1669 **2.5.1 SSCs Important to Safety (MEDIUM Priority)**

1670

1671 Reviewers should verify that the applicant has clearly identified all SSCs important to safety
1672 (see Glossary for the definition of "important to safety") and documented the rationale for this
1673 designation. Such information may be provided in tabular form. Reviewers should review the
1674 general DSS description presented in SAR Chapter 1, "General Description." Reviewers should
1675 ensure that the applicant has provided adequate justification for excluded SSCs.

1676

1677 Reviewers should pay particular attention to instrumentation and other equipment (e.g., lifting
1678 devices and transport vehicles). In general, the NRC staff accepts that monitoring systems
1679 need not be classified as being important to safety. For example, a failure in the functioning of
1680 the pressure monitoring system does not directly result in a release of radionuclides. Additional
1681 justification for not considering such systems as being important to safety may be presented in
1682 later sections of the SAR and summarized in SAR Chapter 2, "Principle Design Criteria".

1683

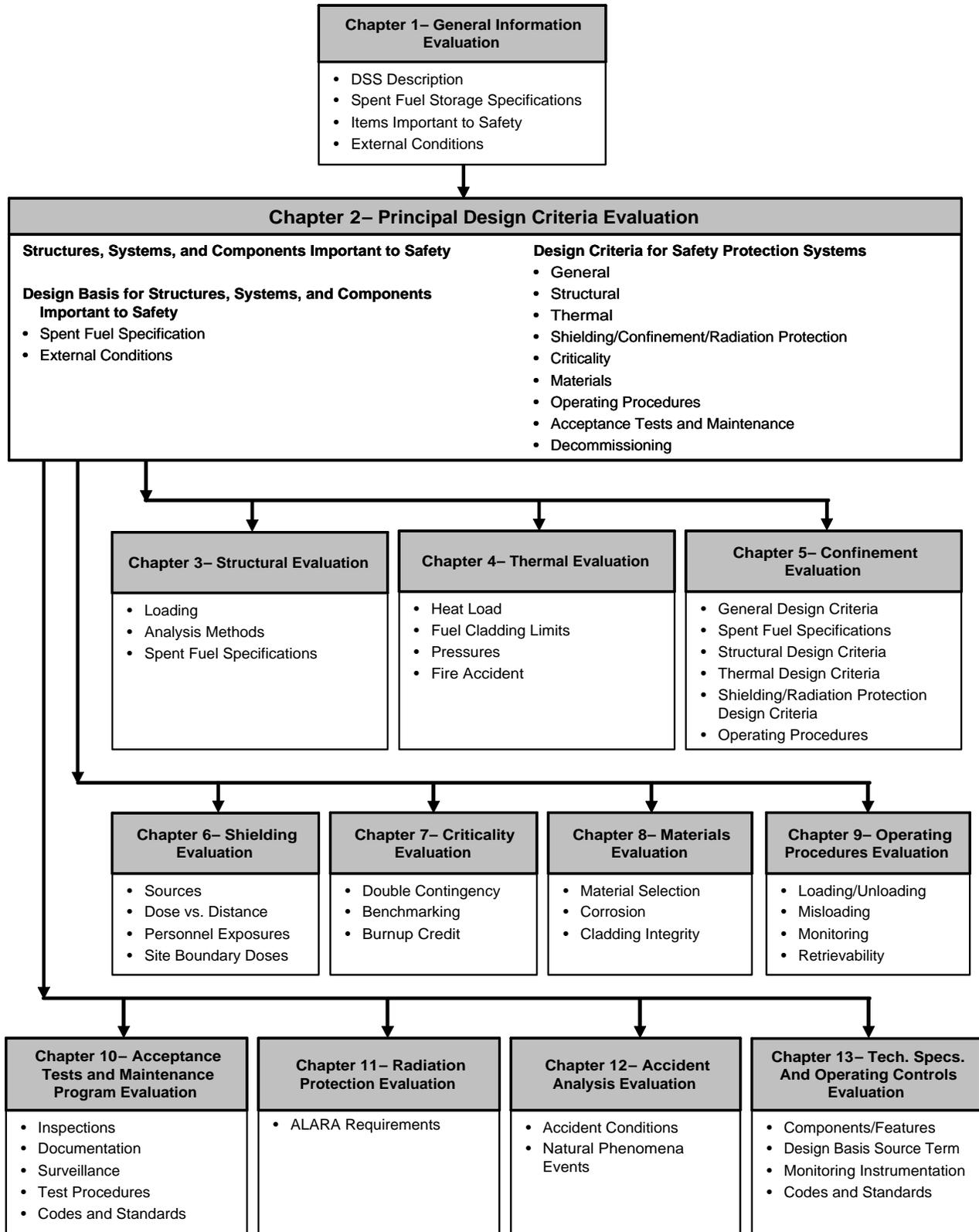
1684 Reviewers should consider adding to SAR Chapter 13 "Technical Specifications and Operating
1685 Controls and Limits" any design features that would have a significant effect on safety if altered
1686 or modified. Any such additions to Chapter 13 should be thoroughly discussed in their
1687 respective sections of the SER.

1688

1689 **2.5.2 Design Bases for SSCs Important to Safety**

1690

1691 The reviewer should verify that the applicant's design basis for DSS approval accurately
1692 identifies the range of SNF configurations and characteristics, the enveloping conditions of use,
1693 the bounding site characteristics, and is consistent with or bounds the DSS's Technical
1694 Specifications. These factors determine the bounds within which an ISFSI owner may use the
1695 SAR rather than provide additional proof regarding suitability of the covered topics.



1696
1697

Figure 2-1 Overview of Principal Design Criteria Evaluation

1698 2.5.2.1 SNF Specifications (MEDIUM Priority)

1699

1700 The reviewer should review the detailed specifications for the SNF to be stored in the DSS as
1701 presented in SAR Chapter 2, "Principle Design Criteria" and ensure that they are consistent with
1702 those specifications discussed in SAR Chapter 1, "General Information" and later in the SAR.
1703 The description of the range of SNF to be stored should include the type (PWR, BWR, or both);
1704 configuration (e.g., 17x17, 15x15, or 8x8); fuel vendor; number of assemblies per cask;
1705 enrichment; burnup and burnup profiles; minimum cooling time; decay heat generation rate;
1706 type of cladding; physical dimensions; total weight per assembly; and uranium weight per
1707 assembly. In addition, if components associated with fuel assemblies (e.g., control assemblies)
1708 will be stored with the fuel, the reviewer should ensure that combined weight, dimensions, heat
1709 load, and other appropriate information (e.g., number per cask) are specified.

1710

1711 The reviewer should examine any limitations regarding the condition of the SNF. If damaged
1712 fuel is allowed, the effects of such damage should be assessed in later sections of the SAR.
1713 Specific conditions that define damaged fuel are provided in Section 8.6, "Supplemental
1714 Information for Methods for Classifying Fuel," of this SRP with methods for classifying fuel
1715 identified in Section 8.4.17.2 of this SRP. If damaged rods have been removed from a fuel
1716 assembly, and they have/have not been replaced with solid dummy rods, the criticality reviewer
1717 should determine whether the intended loading configuration has been adequately analyzed to
1718 show sub-criticality. Note, the presence of additional moderating material will need to be
1719 addressed in the criticality analysis in SAR Chapter 7, "Criticality". Coordination with the
1720 structural reviewer is necessary if there are structural defects in the assembly hardware.

1721

1722 The release of fill and fission product gases from failed fuel rods increases the pressure in the
1723 cask cavity and the potential source term in the event of confinement failure. Consequently, the
1724 reviewer should verify that the applicant provides information regarding the fill/fission product
1725 gas present in the fuel as well as the free volume in the cask cavity to enable reviewers to
1726 evaluate the pressure in the cask cavity resulting from cladding failure during storage. For the
1727 purpose of calculating internal cask pressures, the NRC staff has accepted the bounding
1728 assumptions given in SRP Section 4.5.4.6, "Pressure Analysis" regarding the minimum
1729 percentages of fuel rods have failed (and released their gases).

1730

1731 The reviewer should pay particular attention to the specification of burnup, cooling time, and
1732 decay heat generation rate. These parameters are generally not independent, and the manner
1733 in which they are specified and combined can significantly affect the maximum allowed cladding
1734 temperature as discussed in SRP Chapter 4, "Thermal Evaluation."

1735

1736 The SAR will typically list various fuel assemblies that can be stored in the DSS. It is not
1737 expected that one type of fuel assembly will be bounding for all analyses. The reviewer should
1738 ensure that the applicant has justified which specifications are bounding for each of the
1739 evaluations presented in subsequent sections of the SAR. Specifications used in these
1740 analyses should also be clearly identified or referenced in SAR 13, "Technical Specifications
1741 and Operational Controls and Limits".

1742

1743 If the applicant requests permission for the storage of components that are associated with or
1744 integral to the fuel assembly in the cask, the reviewer should examine the relevant detailed
1745 specifications, conditions, and constraints presented in the SAR. These specifications should
1746 be as detailed as the applicable information presented for the fuel designs to provide the
1747 reviewer with a basis for determining that the relevant safety functions of the DSS will be

1748 maintained. The reviewer should ensure that the applicant also considers the storage of these
1749 components in the analyses.

1750
1751 If the applicant requests burnup credit, the reviewer should examine the relevant detailed
1752 specifications of the contents to which burnup credit is being applied. These specifications
1753 include those that are already considered in criticality analyses for fresh fuel (e.g., maximum
1754 initial enrichment). Additional specifications that must be reviewed include the cooling time, the
1755 burnup, the requested amount of credit (i.e., the specific actinides), and operational history
1756 parameters (e.g., core average boron concentration and assembly average moderator
1757 temperature).

1759 2.5.2.2 External Conditions (MEDIUM Priority up to Natural Phenomena Events)

1760
1761 The SAR should identify those external conditions that significantly affect, or could potentially
1762 affect, the performance of the DSS. These design-basis conditions will generally restrict either
1763 the sites at which the DSS can be used for SNF storage or the manner in which the DSS can be
1764 handled. For example, by selecting the design earthquake, the SAR limits the use of the cask
1765 being reviewed to sites that are bounded by this seismic limit. By establishing a design-basis
1766 drop, the SAR defines the maximum height to which a cask can be lifted without additional
1767 safety analysis or design changes (e.g., addition of impact limiters) by the applicant.

1768
1769 Reviewers should note that movement of cask system components within a reactor building
1770 may not meet the NRC's criteria described in the NRC Bulletin 96-02, "Movement of Heavy
1771 Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety Related Equipment," for
1772 movement of heavy loads within the reactor building. As such, if a potential user (licensee) has
1773 been identified, coordination with the appropriate project manager or technical lead from the
1774 NRC's Office of Nuclear Reactor Regulation (NRR) should occur during the early stages of DSS
1775 design review.

1776
1777 At a minimum, the NRC staff has generally addressed the conditions discussed below; however,
1778 other conditions may be relevant depending on specific details of the DSS design. Reviewers
1779 should pay particular attention to special design features and how these might be affected both
1780 by other external conditions and other DSS components. Reviewers should ensure all required
1781 information is provided in the SAR for the design earthquake accident analysis.

1782
1783 "Normal" conditions (including conditions involving handling and transfer) and the extreme
1784 ranges of normal conditions are presumed to exist during design-basis accidents or design-
1785 basis natural phenomena with the exception of irrational or readily avoidable combinations. For
1786 example, an earthquake or tornado may occur at any time and in combination with any "normal"
1787 condition. By contrast, it can be presumed that transfer, loading, and unloading operations
1788 would not be conducted during a flood.

1789
1790 "Off-normal" conditions and events are presumed to occur in combination with normal conditions
1791 that are not mutually exclusive. Nonetheless, it is not required that the SAR analyze or the
1792 system be designed for the simultaneous occurrence of independent off-normal conditions or
1793 events, design-basis accidents, or design-basis natural phenomena.

1794
1795 Conditions involving a "latent" equipment or instrument failure or malfunction (that is, one that
1796 occurs and remains undetected) should be presumed to exist concurrently with other off-normal
1797 or design-basis conditions and events. Typical latent malfunctions include a misreading
1798 instrument that is not detected as part of routine procedures, an undetected ventilation

1799 blockage, or undetected damage from an earlier design-basis event or condition if no provisions
1800 exist for detection, recovery, or remediation of such conditions.

1801
1802 For normal, off-normal, and accident-level conditions, reviewers should verify that the applicant
1803 has defined appropriate operating and accident scenarios. For these scenarios, the reviewer
1804 should verify the applicant includes in the SAR a comprehensive evaluation of the effects of
1805 such scenarios on the SSCs important to safety. The analyses of such events are addressed in
1806 individual chapters of the SRP. For example, the analyses of an earthquake on the DSS
1807 structural components are addressed in SRP Chapter 3, "Structural Evaluation." The
1808 applicant's evaluations should demonstrate that the requirements of 10 CFR §§ 72.104 and
1809 72.106 as well as 10 CFR Part 20 have been met.

1810
1811 If appropriate, the following design bases should be included as operating controls and limits in
1812 SAR Chapter 13, "Technical Specifications and Operational Controls and Limits Evaluation":

1813
1814 (1) Normal Conditions
1815
1816 For a given SNF specification, the primary external conditions that affect DSS
1817 performance are the ambient temperatures, insolation, and the operational
1818 environment experienced by the DSS.

1819
1820 The NRC accepts as the maximum and minimum "normal" temperatures the
1821 highest and lowest ambient temperatures recorded in each year, averaged over
1822 the years of record. For the SAR, the applicant may select any design-basis
1823 temperatures as long as the restrictions they impose are acceptable to both the
1824 applicant and the NRC. If the cask is also designed for transportation, the
1825 temperature requirements of 10 CFR Part 71 could determine the design-basis
1826 temperatures for storage.

1827
1828 For storage casks, the NRC staff accepts a treatment of insolation similar to that
1829 prescribed in 10 CFR Part 71.71 for transportation casks. If the applicant selects
1830 another design approach, the alternative approach should be justified in the SAR.

1831
1832 The operational environment experienced by the DSS under normal conditions
1833 includes the manner in which the cask is loaded, unloaded, and lifted.
1834 Occupational dose rates will, in part, depend on whether the cask is sealed in a
1835 wet or a dry environment. Fuel cladding temperatures may also be affected.
1836 The manner in which the cask is lifted will determine the load on the trunnions
1837 and/or lifting yoke. The orientation of the cask (vertical or horizontal) and its
1838 height above ground during transport to the ISFSI will establish initial conditions
1839 for the drop accidents discussed below.

1840
1841 (2) Off-Normal Conditions
1842
1843 An applicant's SAR generally addresses several off-normal conditions. These
1844 should include variations in temperatures beyond normal, failure of 10 percent of
1845 the fuel rods combined with off-normal temperatures, failure of one of the
1846 confinement boundaries, partial blockage of air vents, human error, out-of-
1847 tolerance equipment performance, equipment failure, and instrumentation failure
1848 or faulty calibration.

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(3) Accident Conditions

The staff has generally considered that the following accidents should be evaluated in the SAR. These do not constitute the only accidents that should be addressed if the SAR is to serve as a reference for accidents for a specific application. Other credible accidents that may be derived from a hazard analysis could include accidents resulting from operational error, instrument failure, lightning, and other occurrences. Post-accident recovery of damaged fuel may require such systems as overpacks or dry-transfer systems since ready retrieval of the fuel is required only for normal and off-normal conditions. Accident situations that are not credible because of design features or other reasons should be identified and justified in the SAR. Chapter 3, "Structural Evaluation" of this SRP provides more detail regarding accidents.

(a) Cask Drop

The SAR should identify the operating environment experienced by the cask as well as the drop events (i.e., 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 SNF building or in terms of the maximum acceleration that the cask could experience in a drop.

(b) Cask Tipover

Although cask system supporting structures may be identified and constructed as being important to safety (i.e., designed to preclude cask tipovers), the NRC considers that cask tipover events should be analyzed. In some cases, cask tipover may be determined to be a credible hazard, and the associated analysis should reflect the conditions (e.g., heights and accelerations) associated with that hazard.

The NRC staff has accepted an unyielding surface for determining the bounding cask deceleration loads. Prototype or scale model testing and analytical modeling can be used. In the analytical approach, the hard receiving surface need not be unyielding.

(c) Fire

The fire conditions postulated in the SAR should provide an "envelope" for subsequent comparison with site-specific conditions. The NRC accepts the methods discussed in 10 CFR 71.73(c)(4). In addition, the NRC staff accepts that the applicant may consider a fire based upon the limited availability of flammable material at an ISFSI (e.g., only that associated with vehicles transporting or lifting the cask, or sources of nearby combustible materials). Regardless of which approach the applicant takes, the SAR should specify and justify the bounding conditions for a "design-basis" fire.

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(d) Fuel Rod Rupture

The regulations require that the cask be designed to withstand the effects of accident conditions and natural phenomena events without impairing its capability to perform safety functions. Consequently, during the cask analysis for conditions resulting from design-basis accidents and natural phenomena, the NRC has asserted and the applicant should assume a release of 100 percent of the initial rod fill gases and a release of 30 percent of the fission product gases from the fuel rods into the cask interior. The remaining 70 percent of the fission product gases is presumed to be retained within the fuel pellet.

(e) Leakage of the Confinement Boundary

Casks are designed to provide the confinement safety function under all credible conditions. Nevertheless, for assessment purposes and to demonstrate the overall safety of the storage cask system, the NRC staff considers that the DSS should be evaluated for the effects of a confinement boundary failure.

(f) Explosive Overpressure

The conditions under which a DSS may be exposed to the effects of an explosion vary greatly among individual sites. Generally, explosive overpressure is postulated to originate from an industrial accident. The NRC separately evaluated the effects of various sabotage methods on cask systems in developing appropriate regulations in 10 CFR Part 73. Consequently, this SRP does not consider explosive overpressures from sabotage events.

The extent to which explosive overpressure is addressed in the SAR directly affects the degree of site-specific review required. The principal concern in the SAR should be the effects of explosive overpressure on the storage system rather than 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 discussion 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 reviewer should verify the applicant considers blockage of air inlets and outlets in an accident condition. The NRC staff considers that the effects of such an assumption should be utilized in determining the appropriate inspection intervals, and/or monitoring systems, for the DSS.

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(4) Natural Phenomena Events (LOW Priority)

The NRC staff has generally considered that the following events should be evaluated in the SAR:

(a) Flood

The SAR should establish a design-basis flood condition. This condition may be determined on the basis of the presumption that the cask cannot tip over and the yield strength of the cask will not be exceeded. Alternatively, the SAR can show that credible flooding conditions have negligible impact on the cask design.

If the SAR establishes parameters for a design-basis flood, all of the potential effects of flood water and ravine flood byproducts should be recognized. 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 resulting from rapid cooling from immersion.

(b) Tornado

The NRC staff accepts design-basis tornado wind loading as defined by RG 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants" (Region 1) and RG 1.117, "Tornado Design Classification." Design criteria should be established for the cask on the basis of these wind-loading and missile-impact definitions. The cask should not tip over, and the capability to perform the confinement safety function should not be impaired. The NRC staff considers that tornados and tornado missiles may occur without warning. The review should note that, in general, the effects of a tornado missile bound those of a light general aviation aircraft directly impacting a DSS.

(c) Earthquake

The SAR should state the parameters of the design earthquake. For use of a DSS at reactor sites, this is equivalent to the SSE used for analysis of nuclear facilities under 10 CFR Part 50. An analysis for an Operating-Basis Earthquake (OBE) is not required for a DSS SAR prepared in accordance with 10 CFR Part 72. Cask tipover accidents are analyzed, but tipover caused by an earthquake may not be a credible event. The reviewer should verify that the SSCs meet appropriate guidance in RG 1.29, "Seismic Design Classification," RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," and RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".."

(d) Burial Under Debris

Debris resulting from natural phenomena or accidents that may affect cask system performance may be addressed in the SAR or left to the site-

1998 specific application. Such debris can result from floods, wind storms, or
1999 land slides. The principal effect is typically on thermal performance.

2000
2001 (e) Lightning

2002
2003 Lightning typically has a negligible effect on cask systems; however, the
2004 requirements of the Lightning Protection Code and National Electric Code
2005 should be applied to the design of the cask system structures. The
2006 applicant should cite these codes as part of the general design criteria for
2007 the cask system (see Section 2.4.3.1). In addition, the SAR should
2008 address lightning as a natural phenomenon if cask-system performance
2009 may be impacted by the effect of lightning on a component that is
2010 important to safety.

2011
2012 (f) Other

2013
2014 10 CFR Part 72 identifies several other natural phenomena events
2015 (including seiche, tsunami, and hurricane) that should be addressed for
2016 SNF storage. The SAR may include these natural phenomena as design-
2017 basis events or show that their effects are bounded by other events. If
2018 these events are not addressed in the SAR and they prove to be
2019 applicable to a specific site, a safety analysis is required prior to approval
2020 for use of the DSS under either a site-specific or general license.

2021
2022 **2.5.3 Design Criteria for Safety Protection Systems (MEDIUM Priority)**

2023
2024 Cask system components that are to be used in facility areas subject to review under 10 CFR
2025 Part 50 should satisfy both the requirements in 10 CFR Part 72 (with review guided by this SRP)
2026 and 10 CFR Part 50 (with review guided by NUREG-0800). Acceptance of the cask system in
2027 areas covered by 10 CFR Part 50 license requirements is not addressed in this SRP for
2028 approval under 10 CFR Part 72. If the applicant states that the storage system will be used at a
2029 specific reactor site, then the Division of Spent Fuel Storage and Transportation (SFST) project
2030 manager should inform the appropriate NRR project manager. The reviewer is reminded that
2031 heavy loads are a likely matter of interest to NRR.

2032
2033 Table 2-2 presents a summary of design criteria (and design bases) that should generally be
2034 identified during the initial stages of the review. The applicability of Table 2-2 may vary
2035 depending on the details of the storage system design.

2036
2037 Regardless of where the descriptions and associated criteria are located in the SAR, reviewers
2038 should include a description and evaluation of the safety protection systems in SER Chapter 2,
2039 "Principal Design Criteria." The system descriptions should address the functions of the various
2040 system components in providing confinement, cooling, subcriticality, radiation protection of the
2041 public and workers, and SNF retrieval. Summary criteria for the performance of the system as a
2042 whole in providing for these capabilities or functions should also be described and evaluated.
2043 Reviewers should verify that the design-basis assumptions presented are consistent with and
2044 reasonable for actual site or facility conditions. Reviewers should also include a description and
2045 evaluation of the cask system design's compatibility with removal from a reactor site,
2046 transportation, and ultimate disposition of the stored spent fuel.

2047

Table 2-2 Outline of Design Criteria and Bases for DSS

Design Life	<ul style="list-style-type: none"> • Limited to the requested term in the application
Design Bases	<ul style="list-style-type: none"> • SNF Specifications <ol style="list-style-type: none"> (1) Type (2) Configuration/Vendor (3) Enrichment (Maximum and Minimum) (4) Weight or Range of Weights (5) Burnup (6) Type of Cladding (7) Assemblies/Cask (8) Dimensions • Decay Heat/Assembly <ol style="list-style-type: none"> (1) Minimum Decay/Cooling Time (e.g., 5 years, 10 years, etc.) (2) Maximum Kilowatts per assembly • Gas Volume (at Temperature) • Fuel Condition/Damage Allowed • Burnup Credit <ol style="list-style-type: none"> (1) Credit Amount (specific actinides) (2) Operational History Parameters • Non-Fuel Hardware
Normal Design Event Conditions	<ul style="list-style-type: none"> • Ambient Temperature <ol style="list-style-type: none"> (1) Maximum (2) Minimum • Loading <ol style="list-style-type: none"> (1) (Wet/Dry) • Storage Handling Orientation <ol style="list-style-type: none"> (1) (Vertical/Horizontal) • Maximum Lift Height • Maximum Cladding Temperature • Other Conditions Considered in 2.5.2.2 (1)
Off-Normal Design Event Conditions	<ul style="list-style-type: none"> • Summarize Events Considered in 2.5.2.2 (2)
Design-Basis Accident Design Events and Conditions	<ul style="list-style-type: none"> • End Drop <ol style="list-style-type: none"> (1) Lift Height (or Maximum Acceleration) • Side Drop <ol style="list-style-type: none"> (1) Lift Height (or Maximum Acceleration) • Tip-Over <ol style="list-style-type: none"> (1) Acceleration (if applicable) • Fire <ol style="list-style-type: none"> (1) Duration (2) Temperature • Other Events Considered in 2.5.2.2 (3) (As Applicable)

Table 2-2 Outline of Design Criteria and Bases for DSS

<p>Design-Basis Natural Phenomena Design Events and Conditions</p>	<ul style="list-style-type: none"> • Flood • Earthquake • Tornado • Other Events Considered in 2.5.2.2 (4) (As Applicable)
<p>Structural</p>	<ul style="list-style-type: none"> • Design Code (e.g., ASME, AISC) <ul style="list-style-type: none"> (1) Containment (2) Noncontainment (3) Basket (4) Trunnions (5) Storage Radiation and Protective Shielding and Enclosure (6) Transfer Radiation and Protective Shielding and Enclosure (7) Cooling Structure or System • Design Weight • Design Cavity Pressure <ul style="list-style-type: none"> (1) Normal/Off-Normal/Accident • Response and Degradation Limits <ul style="list-style-type: none"> (1) Normal/Off-Normal/Accident
<p>Thermal</p>	<ul style="list-style-type: none"> • Maximum Design Temperatures <ul style="list-style-type: none"> (1) Cladding (2) Other Components • Insolation (Side/Top/Bottom) • Fill Gas <ul style="list-style-type: none"> (1) Type (e.g., helium, etc.) (2) Initial Fill Pressure (at temperature) • Modes of Heat Transfer Utilized in the Design
<p>Confinement</p>	<ul style="list-style-type: none"> • Description of Confinement Boundary • Redundant Seals for Closure • Maximum Leak Rate for Confinement Boundary <ul style="list-style-type: none"> (1) Normal/Off-Normal/Accident (2) Justification of Leakage Rate if not Leaktight • Monitoring System Specifications

Table 2-2 Outline of Design Criteria and Bases for DSS

Radiation Protection/Shielding	<ul style="list-style-type: none"> • Confinement Cask <ul style="list-style-type: none"> (1) Surface Position Normal/Off-Normal/Accident • Exterior of Shielding <ul style="list-style-type: none"> (1) Transfer Mode Position (2) Storage Mode Position Normal/Off-Normal/Accident • ISFSI Controlled Area Boundary <ul style="list-style-type: none"> (1) Dose Rate (2) Annual Dose Normal/Off-Normal/Accident
Criticality	<ul style="list-style-type: none"> • Method of Control Geometry, Fixed Poison, Soluble Poison • Minimum Boron Concentration (Fixed and/or Soluble Poison) • Maximum k_{eff} • Burnable Neutron Absorber Credit • Burnup Credit Analysis
Materials	<ul style="list-style-type: none"> • Cladding Hoop Stress • Corrosion
Operating Procedures	<ul style="list-style-type: none"> • Normal and Off-Normal • After Accident-level Conditions
Acceptance Tests and Maintenance	<ul style="list-style-type: none"> • Industry codes and standards
Tech Specs	<ul style="list-style-type: none"> • Operational Controls and Limits

2048
 2049 Criteria relating to redundancy and allowable levels of response by the DSS under normal, off-
 2050 normal, and accident-level conditions and events should be described and evaluated. In
 2051 general, no unacceptable degradation in physical condition or functional performance should
 2052 result from normal or off-normal conditions. The design criteria regarding limits of permissible
 2053 system response and degradation resulting from an accident condition should be evaluated
 2054 against the SSC capabilities to perform the principal safety functions. Considerations of
 2055 permissible responses should include detectability and corrective actions that may be proposed
 2056 as conditions of system use.

2057
 2058 The staff accepts that both routine surveillance programs and active instrumentation meet the
 2059 intent of “continuous monitoring” as required in 10 CFR 72.122(h)(4).
 2060

2061 Reviewers should note that some DSS designs may contain a component or feature whose
 2062 continued performance over the licensing period has not been demonstrated to staff with a
 2063 sufficient level of confidence (e.g., rubber “O” rings). Therefore, staff may require the use of
 2064 active instrumentation if the failure of that system or component causes an immediate threat to
 2065 the public health and safety, and if that failure would not be detected by any other means. In
 2066 some cases, to demonstrate compliance with 10 CFR 72.122(h)(4), the vendor or NRC staff

2067 may propose a technical specification requiring such instrumentation as part of the first use of a
2068 cask system. After first use, and if warranted and approved by staff, such instrumentation may
2069 be discontinued or modified.

2070
2071 The staff should verify that the applicant has met the intent of continuous monitoring so that the
2072 applicant can determine when corrective action needs to be taken to maintain safe storage
2073 conditions.

2074

2075 **2.6 Evaluation Findings**

2076

2077 The reviewer will prepare evaluation findings on satisfaction of the regulatory requirements in
2078 Section 2.3. If the documentation submitted with the application supports positive findings for
2079 each of the regulatory requirements (the finding number is for convenience in reference within
2080 the SRP and SER), these statements should be similar to the following examples:

2081

2082 F2.1 The SAR and docketed materials adequately identify and characterize the SNF
2083 to be stored in the DSS in conformance with the requirements given in
2084 10 CFR 72.236.

2085

2086 F2.2 The SAR and the docketed materials relating to the design bases and criteria
2087 meet the general requirements as given in 10 CFR 72.122(a), (b), (c), (f), (h)(1),
2088 (h)(4), (i), and (l).

2089

2090 F2.3 The SAR and docketed materials relating to the design bases and criteria for
2091 structures categorized as important to safety meet the requirements given in
2092 10 CFR 72.122(a), (b)(1), (b)(2) and (b)(3), (c), (f), (h)(1), (h)(4), and (i); and 10
2093 CFR 72.236.

2094

2095 F2.4 The SAR and docketed materials meet the regulatory requirements for design
2096 bases and criteria for thermal consideration as given in 10 CFR 72.122 (a),
2097 (b)(1), (b)(2) and (b)(3), (c), (f), (h)(1), (h)(4), and (i).

2098

2099 F2.5 The SAR and docketed materials relating to the design bases and criteria for
2100 shielding, confinement, radiation protection, and ALARA considerations meet the
2101 regulatory requirements as given in 10 CFR 72.104(a) and (b); 10 CFR
2102 72.106(b); 10 CFR 72.122(a), (b), (c), (f), (h)(1), (h)(4), and (i); 10 CFR
2103 72.126(a).

2104

2105 F2.6 The SAR and docketed materials relating to the design bases and criteria for
2106 criticality safety meet the regulatory requirements as given in 10 CFR 72.124(a)
2107 and (b).

2108

2109 F2.7 The SAR and docketed materials relating to the design bases and criteria for
2110 retrieval capability meet the regulatory requirements as given in 10 CFR
2111 72.122(a), (b)(1), (b)(2), and (b)(3), (c), (f), (h)(1), (h)(4), and (l).

2112

2113 F2.8 The SAR and docketed materials relating to the design bases and criteria for
2114 other SSCs not important to safety but subject to NRC approval meet the general
2115 regulatory requirements as given in the following subparts of

2116 10 CFR Part 72: Subpart E, "Siting Evaluation Factors" 72.104 and 72.106;
2117 Subpart F, "General Design Criteria" 72.122, 72.124, and 72.126; and Subpart L,
2118 "Approval of Spent Fuel Storage Casks."
2119

2120 The reviewer should provide a summary statement similar to the following:
2121

2122 "The staff concludes that the principal design criteria for the [cask designation] are
2123 acceptable with regard to meeting the regulatory requirements of 10 CFR Part 72. This
2124 finding is reached on the basis of a review that considered the regulation itself,
2125 appropriate regulatory guides, applicable codes and standards, and accepted
2126 engineering practices. A more detailed evaluation of the design criteria and an
2127 assessment of compliance with those criteria are presented in Chapters 3 through 14 of
2128 the SER."
2129

2130

3 STRUCTURAL EVALUATION

3.1 Review Objective

In this portion of the dry storage system (DSS) review, the U.S. Nuclear Regulatory Commission (NRC) evaluates aspects of the DSS design and analysis related to structural performance under normal and off-normal operations, accident conditions, and natural phenomena events. In conducting this evaluation, the NRC staff seeks a high degree of assurance that the cask system will maintain confinement, subcriticality, radiation shielding, and retrievability of the fuel under all credible loads for normal and off-normal conditions, accidents, and natural phenomenon events.

3.2 Areas of Review

This chapter of the DSS Standard Review Plan (SRP) provides guidance for use in evaluating the design and analysis of the proposed cask system with regard to its structural performance. All DSSs include a confinement cask that may have both internal components and integral external components. In addition, some DSSs have a variety of other components that are subject to NRC evaluation and approval under the cask certification provisions of Subpart L of 10 CFR Part 72.

Recognizing the diversity of the various cask system components, the NRC has broadly categorized the applicable review procedures and acceptance criteria as follows:

- Structural Capability of the Confinement boundary and Internals,
- Other structural system components important to safety,
- Other structural components subject to NRC approval.

Within these broad categories, the NRC focuses the DSS structural evaluation, as described in Section 3.5, "Review Procedures," using the following areas of review as appropriate:

Scope

Structural Design Criteria and Design Features

- Design Criteria
 - General Structural Requirements
 - Applicable Codes and Standards
- Structural Design Features

Materials Related to Structural Evaluation

Structural Analysis

- Load Conditions
 - Normal Conditions
 - Off-normal Conditions
 - Natural Phenomena and Accident Conditions
- Structural Analysis Methods
 - Finite-element Analysis
 - Closed-form Calculations
 - Structural Analysis for Specific Cask Components
- Structural Evaluation

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Structural Capability
 Fabrication and Construction

3.3 Regulatory Requirements

Table 3-1 presents a matrix that shows the primary relationship of the regulations provided in this section to the specific areas of review associated with this SRP chapter. The NRC staff reviewer should verify the association of regulatory requirements with the areas of review presented in the matrix to ensure that no requirements are overlooked as a result of unique applicant design features.

Table 3-1 Relationship of Regulations and Areas of Review				
Areas of Review	10 CFR Part 72 Regulations			
	72.124(a)	72.234(a), (b)	72.236(b),(c), (d), (l)	72.236(g), (h)
Scope	•	•	•	
Structural Design Criteria and Design Features	•	•	•	•
Materials Related to Structural Evaluation			•	
Structural Analysis		•	•	
Structural Evaluation		•	•	•

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3.4 Acceptance Criteria

The most important function of the structural analysis is to ensure sufficient structural capability for every applicable section of the cask system to withstand the worst-case loads under accident conditions and natural phenomena events. Withstanding such loads enables the cask system to successfully preclude the following negative consequences:

- Unacceptable risk of criticality,
- Unacceptable release of radioactive materials,
- Unacceptable radiation levels,
- Impairment of retrievability.

Because of the diversity of cask system components and various materials that are subject to NRC evaluation and approval, it is not possible to define objective structural review criteria that address all possible component configurations. No single structural code currently accepted by the NRC (such as the American Society of Mechanical Engineers [ASME] Boiler and Pressure Vessel [B&PV] Code, Section III, Division 1 [ASME B&PV]) or Section III, Division 2 may cover the design of all spent nuclear fuel (SNF) storage systems. Consequently, the acceptability of any given structure will be contingent upon a combination of adherence to applicable portions of

2213 multiple codes and a review of the functional performance of the structure taken as a whole.
2214 This combined approach allows the designer to request relief, or provide alternatives, and the
2215 reviewer to impose additional restrictions when warranted by specific design features.
2216

2217 In general, the DSS structural evaluation seeks to ensure that the proposed design and analysis
2218 fulfill the following acceptance criteria that reflect the industry codes and standards the NRC
2219 staff has accepted in past DSS structural evaluations. The American National Standards
2220 Institute's (ANSI) "Design Criteria for an Independent Spent Fuel Storage Installation (Dry
2221 Storage Type)" (ANSI/ANS-57.9) generally applies to the design and construction of an ISFSI
2222 but contains some criteria/design requirements relative to dry storage systems.
2223

2224 **3.4.1 Confinement Cask and Metallic Internals**

2226 3.4.1.1 Steel Confinement Cask

2227
2228 The structural design, fabrication, and testing of the confinement system and its redundant
2229 sealing system should comply with an acceptable code or standard such as ASME B&PV Code.
2230 (The NRC has accepted use of either Subsection NB or Subsection NC of Section III, Division 1
2231 of this code.) Division 3 of Section III of the ASME B&PV Code, addressing storage of spent
2232 nuclear fuel, has been published, but currently no NRC position has been established on that
2233 standard. Other design codes or standards may be acceptable depending on their application.
2234 An applicant must justify the use of new criteria where no NRC staff position has been
2235 established.
2236

2237 i. The NRC staff evaluates the proposed limitations on allowable stresses and
2238 strains in the confinement cask, steel parts important to safety and subject to
2239 review by comparison with those specified in applicable codes and standards.
2240 Where certain proposed load combinations will produce values that exceed the
2241 accepted limits for localized points on the structure, the applicant should provide
2242 adequate justification to show that the deviation will not affect the functional
2243 integrity of the structure. Under certain conditions limiting strains and limiting
2244 deformations may form part of the acceptance criteria.
2245

2246 ii. The NRC has accepted the use of applicable subsections of the ASME B&PV
2247 Code, Section III, Division 1, such as Subsections NF and NG, for components
2248 used in the cask system. This includes the "basket" structure used in casks to
2249 restrain and position multiple fuel elements in the storage system in which
2250 Subsection NG has been used.
2251

2252 3.4.1.2 Steel-Lined Concrete Confinement Cask

2253
2254 i. The American Concrete Institute (ACI) and ASME's "Code for Concrete Reactor
2255 Vessels and Containments" (ACI 359), also known as Section III, Division 2 of
2256 the ASME B&PV Code, constitutes an acceptable standard for prestressed and
2257 reinforced concrete structures that are an integral component of a steel-lined
2258 concrete confinement cask that must withstand internal pressure in operation or
2259 testing and constitutes a confinement cask. The minimum functional
2260 requirements of ANSI/ANS-57.9 for subject areas not specifically addressed in
2261 ACI 359 shall be met.
2262

2263 ii. The NRC will review the use of applicable subsections of the ASME B&PV Code,
2264 Section III, Division 1, such as Subsections NF and NG, for components used
2265 within the confinement cask but not integrated with it. This includes Subsection
2266 NG for the “basket” structure used in casks to restrain and position multiple fuel
2267 elements in the storage system.
2268

2269 **3.4.2 Other Structural System Components and Structures Important to Safety**

2270

2271 The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited
2272 therein) as the basic reference for the cask system structures important to safety that are not
2273 designed in accordance with accepted provisions or alternatives to applicable portions of
2274 Section III, Division 1 or 2 (ACI-359) of the ASME B&PV Code. Structures and components that
2275 are important to safety which are related to lifting and handling cask systems should comply
2276 with American National Standards Institute (ANSI) Standard, “American National Standards for
2277 Radioactive Material Lifting Devices for Shipping Containers Weighing 10,000 lbs (4500 kg) or
2278 More” (N14.6). American Society of Civil Engineers, “Minimum Design Loads for Buildings and
2279 Other Structures,” (ASCE 7) that can be used when load combinations established on the basis
2280 of ANSI/ANS-57.9 are used.
2281

2282 **3.4.2.1 Steel Structures**

2283

2284 The principal codes and standards include the following references that may be applied to steel
2285 structures and components:
2286

- 2287 a. American Institute of Steel Construction (AISC), “Specification for Structural Steel
2288 Buildings — Allowable Stress Design and Plastic Design.”
- 2289 b. AISC, “Load and Resistance Factor Design Specification for Structural Steel
2290 Buildings.”
- 2291 c. American Welding Society, “Structural Welding Code Steel,” (AWS D1.1).
2292

2293 **3.4.2.2 Reinforced Concrete Structures**

2294

2295 ACI’s “Code of Requirements for Nuclear Safety Related Concrete Structures,” ACI 349 can be
2296 applied to reinforced concrete structures and components.
2297

2298 **3.4.3 Other Structural Components Subject to NRC Approval**

2299

2300 For structural design and construction of other components subject to NRC approval, the
2301 principal codes and standards include the following:
2302

- 2303 a. American Society of Civil Engineers (ASCE), “Minimum Design Loads for
2304 Buildings and Other Structures” (ASCE 7).
- 2305 b. International Building Code (IBC) 2006 from International Code Council.
- 2306 c. AISC, “Specification for Structural Steel Buildings—Allowable Stress Design and
2307 Plastic Design.”
- 2308 d. AISC, “Code of Standard Practice for Steel Buildings and Bridges.”
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- e. ASME B&PV Code, Section VIII.
- f. ACI 318, "Building Code Requirements for Structural Concrete."

3.5 Review Procedures (HIGH Priority)

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

The SAR should clearly identify the confinement system and other structures important to safety, and each component should 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 retrievability of stored nuclear materials (the NRC has accepted some degradation of retrievability under accident conditions).

This position does not necessarily require that all confinement system and other structures important to safety survive all design-basis accidents and extreme natural phenomena without any permanent deformation or other damage. Some load combination expressions for the design basis event (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 accident deformations and any permanent deformations, degradation, or other damage that may occur. The reviewer should verify that the applicant has performed computations, analyses, and/or tests and that both the tests and results are acceptable to the NRC to clearly demonstrate that any permanent deformations, degradation, or other damage that may occur does not render the system performance unacceptable.

Structures important to safety are not required to survive accidents to the extent that they remain suited for use for the life of the cask system without inspection, repair, or replacement. If the service life of structures important to safety may be degraded by accident-level conditions, there must be SAR commitments and procedures for determining and correcting the degradation and performing other acceptable remedial action.

The proposed technical specifications should be reviewed to ensure that they include adequate restrictions on cask handling and operations to preclude the possibility of damage to the structure or the confined nuclear material. Operating controls and limits of the technical specifications (reviewed under Chapter 13 of this SRP) should be included in both the SAR and the SER, and should describe actions to be taken and inspections to be conducted upon occurrence of events that may cause such damage.

2364 Figure 3-1 presents an overview of the evaluation process and can be used as a guide to assist
2365 in coordinating with other review disciplines.

2366
2367 In evaluating the structural design and performance of a proposed DSS, the reviewer should
2368 select and emphasize aspects of the following review procedures, as appropriate for the
2369 particular DSS, in relation to the acceptance criteria summarized in Section 3.4.

2370

2371 Description of Structures, Systems, and Components Important to Safety

2372

2373 The reviewer should verify that the applicant's safety analysis report (SAR) clearly identifies the
2374 proposed structural design and construction of structures, systems, and components (SSCs)
2375 that are important to safety and necessary for effective functional performance and safety of the
2376 DSS. The SAR and supplemental material submitted by the applicant should be reviewed to
2377 assess compliance with the applicable scope and content requirements defined in 10 CFR
2378 72.230. The reviewer should focus in particular on requirements and conditions of use related
2379 to design, construction, implementation, operation, and maintenance of structural SSCs.

2380 Applicable Codes, Standards, and Specifications

2381

2382 NRC guidelines recommend that the safety evaluation report (SER) prepared by the NRC staff
2383 include a table (in the design criteria evaluation section) summarizing the applicable reference
2384 sources. This table should identify all source documents cited in the SAR, their usage (e.g.,
2385 description of model, prior NRC approval of cask system elements, design code, construction
2386 code), and acceptability for that usage. The sources of interest include documents directly
2387 referenced in the SAR; sources of material incorporated by reference; and codes, standards,
2388 specifications, and other sources of criteria that further define the design and construction of the
2389 proposed structures. If not tabulated, the consolidated review and assessment of reference
2390 sources should otherwise be included in the SER.

2391

2392 Loads and Load Combinations

2393

2394 The reviewer should verify that the loads and load combinations are as specified in Chapter 2,
2395 "Principal Design Criteria Evaluation," of this SRP. If the applicant has not adequately justified
2396 any deviations from the acceptance criteria for loads and load combinations, the reviewer
2397 should identify the deviations as unacceptable and transmit them to the applicant for further
2398 justification. If components associated with or integral to the fuel assembly are to be stored in
2399 the cask, then the reviewer should ensure these components are considered by the applicant in
2400 the structural analyses.

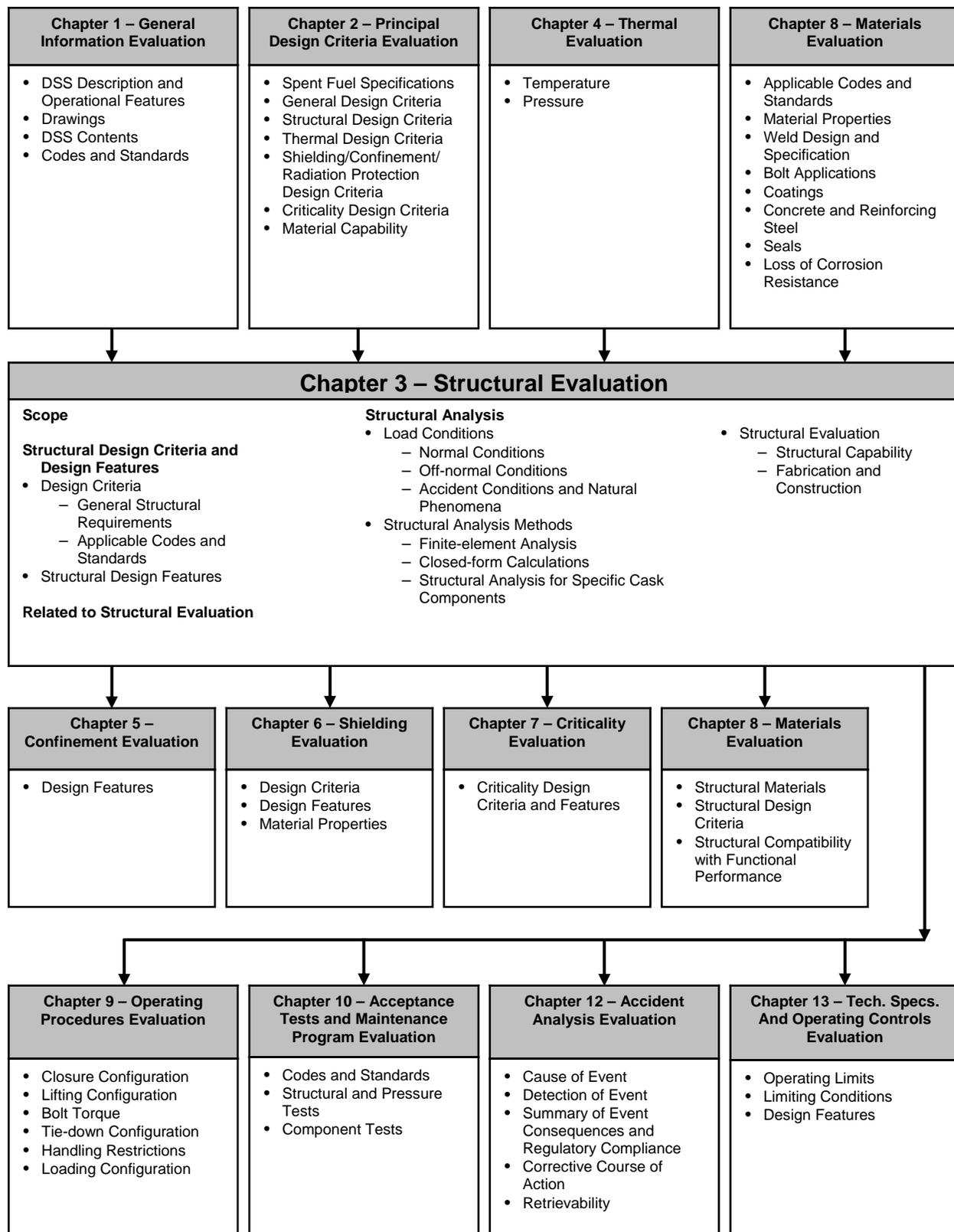


Figure 3-1 Overview of the Structural Evaluation

2401
2402
2403

2404 The SAR should include a comprehensive table of load combinations and safety margins for
2405 selected structural sections of components important to safety (or otherwise subject to NRC
2406 evaluation). The summary table should include sufficient structural sections and forms of
2407 loadings (e.g., shear, flexure, axial, and combined stress situations) to verify that the lowest
2408 margins of safety are represented for the various components. In addition, this table can be
2409 used to summarize the structural capacity evaluation.

2410
2411 Design and Analysis Procedures

2412
2413 The reviewer should determine whether the applicant's design and analysis procedures and
2414 assumptions are conservatively defined on the basis of accepted engineering practice. The
2415 behavior of the structure under various loads, and the manner in which these loads are treated
2416 in conjunction with other coexistent loads should be reviewed, while compliance with the
2417 acceptance criteria, defined in Section 3.4 of this SRP should be assessed.

2418
2419 Structural Acceptance Criteria

2420
2421 The proposed limitations on allowable stresses, strains, or deformations in the confinement
2422 cask, its internals, system components important to safety, and other components subject to
2423 review should be analyzed. The reviewer should compare the proposed limitations with those
2424 specified in the applicable codes and standards. Where the applicant proposes to exceed the
2425 accepted limits for certain load combinations at localized points on the structure, the reviewer
2426 should evaluate the justification provided to ensure that the deviation will not affect the
2427 functional integrity of the structure. If the justification is not acceptable, the reviewer should
2428 request additional justification and bases.

2429
2430 Materials, Quality Control, and Special Fabrication Techniques

2431
2432 Information provided in the SAR regarding materials is reviewed under the guidance of Chapter
2433 8, "Materials Evaluation" of this SRP. Quality control methods, and special fabrication
2434 techniques, if any, related to the structural evaluation should be reviewed in coordination with
2435 the materials discipline and QA. The QA program is reviewed under Chapter 14 "Quality
2436 Assurance Evaluation" of this SRP. If the applicant proposes to use a new material not
2437 addressed in prior approvals, the applicant must provide sufficient data regarding the material's
2438 structural properties to establish the acceptability of the material. Similarly, the reviewer should
2439 evaluate any new quality control programs or construction techniques to ensure that they will
2440 not degrade the structural quality, integrity, or function of the DSS.

2441
2442 Testing and In-Service Surveillance Requirements

2443
2444 The proposed pressure test procedures for the confinement cask should be reviewed in
2445 comparison with the procedures described in ASME Code, Section III, Division 1, Subsection
2446 NB-6000, and in conjunction with Chapter 10, "Acceptance Tests and Maintenance Program
2447 Evaluation" of this SRP. Also, the proposed acceptance test and maintenance requirements for
2448 trunnions should be reviewed in comparison with those described in the ASME Code and ANSI
2449 N14.6, as applicable for load bearing components. Any other proposed testing and in-service
2450 surveillance programs should be reviewed on a case-by-case basis. Also, the reviewer should
2451 read SAR Section 10 to verify that the applicant has included all appropriate acceptance tests
2452 and addressed all required evaluations in Section 10 of the SER.

2453

2454 Conditions for Use of Structures

2455
2456 The structural evaluation should be reviewed to determine if conditions of use or technical
2457 specifications should be associated with the structural design or proposed fabrication and
2458 construction methods. The reviewer should determine the appropriateness of and need for any
2459 proposed technical specifications related to structural design and construction. Also, the
2460 reviewer should determine whether any additional technical conditions related to structural
2461 performance are needed and, if so, provide input to the conditions of use discussed in the SER.
2462 In addition, the reviewer should describe the basis for the suggested conditions in the structural
2463 evaluation section of the SER. Structure-related conditions of use may be linked to evaluations
2464 performed under other sections (such as a field verification that maximum concrete
2465 temperatures predicted from thermal analysis will not be exceeded).

2466
2467 The remainder of this section provides specific review procedures for each of the three
2468 categories of cask system components including the confinement cask and steel internals, other
2469 structures important to safety, and other components subject to NRC approval. Within each of
2470 these broad categories, the specific review procedures focus the DSS structural evaluation
2471 using the areas of review identified in Section 3.2 of this SRP.

2472
2473 **3.5.1 Confinement Cask and Metallic Internals**

2474
2475 The structural review of the confinement cask addresses drawings, plans, sections, supporting
2476 computations, and specifications for those structural components comprising confinement
2477 barriers. The review also addresses structural and sealing interfaces, and connections that are
2478 necessary to complete the confinement system (as defined in 10 CFR Part 72). In addition, this
2479 review includes evaluation of components that serve no structural function to confirm that they
2480 do not impair the functioning of the confinement cask. The review also encompasses the
2481 evaluation of the metallic internals that constitute the “basket” structure.

2482
2483 3.5.1.1 Scope

2484
2485 The SAR must describe all components of the confinement cask and internals important to
2486 safety in sufficient detail to allow evaluation of their structural behavior and effectiveness under
2487 the imposed design conditions. In addition, the SAR must identify all codes and standards
2488 applicable to the components.

2489
2490 The discussion in the SAR must demonstrate that all components of the confinement cask and
2491 internals important to safety will be designed and fabricated to quality standards commensurate
2492 with the importance to safety of the function to be performed. In addition, components of the
2493 confinement cask and internals important to safety must be designed to accommodate the
2494 combined loads anticipated during normal, off-normal, accident, and natural phenomenon
2495 events with an adequate margin of safety.

2496
2497 3.5.1.2 Structural Design Criteria and Design Features

2498
2499 i. Design Criteria (MEDIUM Priority)

2500
2501 The cask-related design criteria presented in SAR Chapter 2, “Principle Design
2502 Criteria Evaluation” should be reviewed as well as the criteria provided herein.
2503 The NRC generally considers the following design criteria to be acceptable to
2504 meet the structural requirements of 10 CFR Part 72:

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(1) General Structural Requirements

The proposed cask must maintain confinement of radioactive material under normal and off-normal operations, accident conditions, and natural phenomenon events. In addition, neither the cask nor any basket within the cask may deform under credible loading conditions in a manner that would jeopardize the subcritical condition or retrievability of the fuel.

The design must adequately protect the fuel cladding against gross rupture caused by degradation resulting from design or accident conditions. In addition, the design must ensure that the SNF will not experience accelerations/decelerations that would damage its structural integrity or jeopardize its subcritical condition or retrievability under normal and off-normal design conditions.

The applicant must analyze the cask to show that it will not tip over or drop in its storage condition as a result of a credible natural phenomenon event. A tipover or drop is always assessed as a bounding condition during handling operations.

Radiation shielding in the cask system is required to protect the public and workers involved with spent fuel handling and storage, and such shielding must not degrade under normal or off-normal conditions or events. The shielding function may degrade as a result of an accident (e.g., displacement of source or shielding, reduction in shielding). However, the loss of function must be readily visible, apparent, or detectable. (Any permissible degradation in shielding must be shown to result in dose rates sufficiently low to permit recovery of the damaged cask including unloading, if necessary). The necessary structural criteria to assure adequate shielding remains in-place should be clearly identified.

(2) Applicable Codes and Standards

The structural design, fabrication, and testing of the confinement system and any necessary redundant sealing system should comply with acceptable codes or standards. Use of codes and standards previously accepted by the NRC expedites the evaluation process. Use of other codes and standards, definition of criteria composed of extracts from multiple 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 a custom NRC review and may delay the evaluation process.

Section III, Division 1, of the ASME B&PV Code is an accepted code for design, fabrication, and test of steel confinement casks. Specifically, the NRC has accepted use of either Subsection NB or NC. Other design codes or standards may be acceptable depending on their application. The NRC has accepted use of the applicable subsections of the ASME Code, Section III, Division 1, for cask system components used within the

2556 confinement cask but not integrated with it. This includes the “basket,”
2557 which is a structure used in casks to restrain and position multiple fuel
2558 elements. Section III, Division 3 of the ASME B&PV Code is also
2559 available and addresses storage cask systems, but NRC has not
2560 endorsed its use at the current time.

2561
2562 Also, the NRC has accepted applicable subsections of Division 1, of the
2563 ASME Code, for structural external integral elements of the confinement
2564 (e.g., Subsection NF for integral supports) cask.

2565
2566 Commitments for structures important to safety to ASME Code Section III,
2567 with proposed alternatives to the Code, should be documented in the
2568 application. Likewise, NRC staff-approved alternatives to the Code
2569 should be incorporated, either directly or referenced, in the certificate of
2570 compliance (or in the technical specifications attached to the certificate)
2571 issued by the NRC. In the event that alternatives to codes are required
2572 during fabrication and the alternatives do not impact the quality or safety
2573 of the component, an alternative to the requirements of the certificate of
2574 compliance or technical specification may be granted with approval of the
2575 NRC.

2576
2577 Applicants should propose a condition to the certificate of compliance or
2578 technical specification, either directly or referenced, describing the
2579 alternatives to the referenced codes. The condition or technical
2580 specification should also describe a process to address one-time
2581 alternatives from the ASME Code that may occur during fabrication. The
2582 information provided should include the identification of the component,
2583 the reference to the ASME Code (code edition, addenda, section or
2584 article), description of the Code requirement, and a description of the
2585 alternative. In addition, the applicant should justify the alternative,
2586 including a description of how the alternative would provide an acceptable
2587 level of quality and safety. Additionally, the applicant should describe
2588 how compliance with the code provisions would result in hardship or
2589 difficulty without a compensating increase in the level of quality or safety.

2590
2591 For a steel-lined concrete confinement cask system, NRC accepts ACI
2592 359, also designated Section III, Division 2, of the ASME Boiler and
2593 Pressure Vessel Code. This Code is acceptable for prestressed and
2594 reinforced concrete that is an integral component of a radioactive material
2595 containment vessel that must withstand internal pressure in operation or
2596 testing. ACI 359, as endorsed by RG 1.136, Rev. 3, “Design Limits,
2597 Loading Combinations, Materials, Construction, and Testing of Concrete
2598 Containments,” and Section 3.8.1, “Concrete Containments” of NUREG-
2599 0800, “Standard Review Plan for Review of Safety Analysis Reports for
2600 Nuclear Power Plants,” should be applied on the basis of containment
2601 function regardless of whether the concrete structure is fixed or portable
2602 and regardless of where the concrete structure is fabricated. ACI 359
2603 also applies to structural concrete supports constructed as an integral
2604 part of the containment. If ACI 359 and RG 1.136 apply to the structure,
2605 the Code applies to the entire design, material selection, fabrication, and
2606 construction of that structure.

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As an alternative to the requirements of Section CC-3440 of ACI 359, the NRC also accepts the following. These criteria are an alternative to the temperature requirements of ACI 349, A.4, but only for the specified uses and temperature ranges:

- a. If concrete temperatures of general or local areas are 93°C (200°F) in normal or off-normal conditions/ occurrences, no tests to prove capability for elevated temperatures or reduction of concrete strength are required.
- b. If concrete temperatures of general or local areas exceed 93°C (200°F) but would not exceed 149°C (300°F), no tests to prove capability for elevated temperatures or reduction of concrete strength are required if Type II cement is used and aggregates are selected which are acceptable for concrete in this temperature range. The following criteria for fine and coarse aggregates are acceptable:
 - 1) Satisfy ASTM C33, ("Standard Specification for Concrete Aggregates," 2002) requirements and other requirements referenced in ACI 349 for aggregates.
 - 2) Have demonstrated a coefficient of thermal expansion (tangent in temperature range of 20°C to 38°C [70°F to 100°F]) no greater than 11×10^{-6} mm/mm/°C (6×10^{-6} in./in./°F), or be one of the following minerals: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.
- c. If concrete temperatures of general or local areas in normal or off-normal conditions or occurrences do not exceed 107°C (225°F), the requirements of 1 and 2 apply to the coarse aggregate, but fine aggregate that meets 1 and is composed of quartz sands or sandstone sands may be used in place of compliance with 2.

ii. Structural Design Features (LOW Priority)

The cask-related descriptive information presented in SAR Chapter 1, "General Information Evaluation" should be reviewed as well as any related information provided in SAR Chapter 3 "Structural Evaluation". The drawings, figures, tables, and specifications included in the SAR should fully define the structural features of the cask. These may include the cask system that could include an inner shell, an outer shell, and a gamma shield, inner and outer lids and bolts, port covers and bolts, vent port covers to be welded in place, neutron shields and shell, trunnions, fuel basket, and impact limiters (if used).

The reviewer should coordinate with the confinement review (Chapter 5, "Confinement Evaluation," of this SRP) to verify that the SAR clearly identifies the confinement boundaries. These boundaries include the primary confinement

2656 vessel; its penetrations, seals, welds, and closure devices; and the redundant
2657 sealing system as provided by the system.

2658
2659 The list of weights and calculation of the cask center of gravity should be
2660 reviewed. The reviewer should verify that the applicant used the appropriate
2661 limiting cases in the structural evaluations.

2662
2663 3.5.1.3 Materials Related to Structural Evaluation (HIGH Priority)
2664

2665 The structural reviewer should coordinate with the materials reviewer to determine the impact of
2666 corrosion, reviewed in Chapter 8, "Materials Evaluation" of this SRP, on structural integrity. The
2667 reviewer should ensure that the applicant used appropriate corrosion allowances for the
2668 structural analyses. The reviewer should consider the static and dynamic (where appropriate)
2669 stresses, strains, deformations, and response, and the limits used for the structural design and
2670 evaluations.

2671
2672 A DSS serves to confine and maintain safe storage conditions throughout its service life.
2673 Design and construction codes (e.g., ASME B&PV Code Section III) give reasonable assurance
2674 that the as-fabricated material will provide the necessary integrity. It is noted that the ASME
2675 Code Section III, Division 1, applies specifically to maintaining pressure boundaries and
2676 supporting structures in nuclear power plants, and may not necessarily be totally applicable to
2677 all DSS. However, designers may choose to cite it as the code to which selected components
2678 are to be fabricated. Codes such as the ASME B&PV are not likely to address all the potential
2679 performance conditions (e.g., cracking, creep, corrosion, etc.) that may arise from
2680 environmental, electrochemical, or dynamic-loading. These and other effects are specific to the
2681 individual application and should be addressed to meet the guidance of Chapter, 8, "Materials
2682 Evaluation" of this SRP.

2683
2684 The reviewer should verify that the properties used are appropriate for the load conditions of
2685 interest (e.g., static or dynamic, impact loading, hot or cold temperature, wet or dry conditions).
2686 SAR Chapter 12, "Accident Analyses Evaluation" should be reviewed to ensure that the
2687 applicant considered any appropriate restrictions regarding temperature or environmental
2688 conditions for the materials under accident conditions.

2689
2690 The reviewer should coordinate with the thermal and material disciplines to determine the
2691 appropriate temperatures at which allowable stress limits should be defined. For most cask
2692 materials, the stress limits should be defined at the maximum temperature for each material as
2693 established by the SAR thermal analysis. Further discussion of the background for the
2694 temperature limits can be found in Chapters 4, "Thermal Evaluation" and 8, "Materials
2695 Evaluation" of this SRP.

2696
2697 The reviewer should coordinate with the materials, criticality, and shielding reviews to ensure
2698 that, during storage and accident conditions, any structural materials considered as neutron
2699 absorbers and/or gamma shields will perform safety functions as intended.

2700
2701 If the cask has impact limiters, used in the transfer and storage operations, the applicant should
2702 thoroughly evaluate and verify their nonlinear impact characteristics. In addition, the applicant
2703 should tabulate and describe the crush characteristics and properties of the limiters in the
2704 directions that are to be used.

2705

2706 3.5.1.4 Structural Analysis
2707
2708 i. Load Conditions
2709
2710 (MEDIUM Priority) To meet the structural requirements of 10 CFR Part 72, the
2711 DSS design must accommodate the full spectrum of load conditions including all
2712 anticipated normal, off-normal, and accident-level conditions (including natural
2713 phenomenon events). The system should not experience any permanent
2714 deformation or loss of safety function capability during normal or off-normal
2715 operating conditions. However, the system may experience some permanent
2716 deformation, but no loss of safety function capability, in response to an accident.
2717
2718 (1) Normal Conditions (LOW Priority)
2719
2720 Normal conditions and events are those associated with cask system
2721 operations, including storage of nuclear material, under the normal range
2722 of environments. The SAR should state the assumed limits of normal use
2723 environments to support evaluation by a user of the certified cask system
2724 suitability for use at a specific site under a general license.
2725
2726 Loads normally applicable to a confinement cask include weight, internal
2727 and external pressures, and thermal loads associated with operating
2728 temperature. The loads experienced may vary during loading,
2729 preparation for storage, transfer, storage, and retrieval operations. The
2730 weight is the maximum or design weight (including tolerances) of the cask
2731 as it is stored and loaded with SNF. However, depending on the
2732 operation and procedures, the weight should also include water fill. The
2733 applicant should evaluate all orientations of the cask body and closure
2734 lids during normal operations and storage conditions including loads
2735 associated with loading, transfer, positioning, and retrieval of the
2736 confinement cask.
2737
2738 Internal pressures result from hydrostatic pressure, cask drying and
2739 purging operations, filling with non-reactive cover gas, out-gassing of fuel,
2740 refilling with water, radiolysis, and temperature increases. Temperature
2741 variations and thermal gradients in the structural material may cause
2742 additional stresses in the cask and closure lids. The reviewer should
2743 coordinate with the thermal review (Chapter 4, "Thermal Evaluation," of
2744 this SRP) to determine the conservative (or enveloping) values and
2745 combinations of the cask internal pressures and temperatures for both hot
2746 and cold conditions. The reviewer should use the temperature gradients
2747 calculated in SAR Chapter 4 to determine thermal stresses. Note that if
2748 the confinement system has several enclosed areas; all areas may not
2749 have the same internal pressures. In some casks, enclosed areas
2750 consist of the cask cavity and the region between the inner and outer lids.
2751
2752 Required evaluations include weight plus internal pressures and thermal
2753 stresses from both hot and cold conditions. The reviewer should verify
2754 that the applicant included the maximum thermal gradient as determined
2755 in the thermal analysis, when evaluating thermal stresses.
2756

2757 (2) Off-Normal Conditions (LOW Priority)
2758
2759 The review should identify and evaluate all off-normal events and
2760 conditions described in Chapter 12, "Accident Analyses Evaluation," of
2761 this SRP. The off-normal conditions and events should be reviewed for
2762 those that affect the confinement cask structure. The confinement cask
2763 components should satisfy the same structural criteria required for normal
2764 conditions, as discussed above.
2765
2766 The SAR should clearly identify anticipated off-normal conditions and
2767 events that may reasonably be expected to occur during the life of the
2768 cask system at the proposed site. In addition, the SAR should state the
2769 environmental limits to support comparison of the cask system design
2770 bases with specific site environmental data. Off-normal conditions and
2771 events can involve potential mishandling, simple negligence of operators,
2772 equipment malfunction, loss of power, and severe weather (short of
2773 extreme natural phenomena).
2774
2775 (3) Accident-Level Events and Conditions
2776
2777 The reviewer should follow the guidance below in reviewing the structural
2778 response to accident-level conditions. Note that the SAR must address,
2779 at a minimum, each of the following accidents. However, this discussion
2780 may not address all of the potential events or accidents that apply to a
2781 cask (Chapter 12 of this SRP addresses the identification and evaluation
2782 of accidents).
2783
2784 (a) Cask Drop and Tipover (HIGH Priority)
2785
2786 The reviewer should ensure the applicant performs a cask drop
2787 and tipover analysis or demonstrates that this scenario is not
2788 credible. The SAR should identify the operating environment
2789 experienced by the cask and the drop events (end/side/tipover)
2790 that could result. Generally, applicants establish the design basis
2791 in terms of the maximum height to which the cask is lifted outside
2792 the building or the maximum deceleration that the cask could
2793 experience in a drop. The design-basis drops should be
2794 determined on the basis of the actual potential handling and
2795 transfer accidents.
2796
2797 If the analytical approach described in the LLNL report
2798 UCID-21246 (Chun, R., et al., 1986) for axial buckling is used to
2799 assess fuel integrity for the cask drop accident, the analysis
2800 should use the irradiated material properties and should include
2801 the weight of fuel pellets.
2802
2803 Alternatively, an analysis of fuel integrity which considers the
2804 dynamic nature of the drop accident and any restraints on fuel
2805 movement resulting from cask design is acceptable if it
2806 demonstrates that the cladding stress remains below yield. If a
2807 finite element analysis is performed, the analysis model may

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consider the entire fuel rod length with intermediate supports at each grid support (spacer). Irradiated material properties and weight of fuel pellets should be included in the analysis.

The NRC will accept cask tipover 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 cask drops with the longitudinal axis horizontal which, together with analysis of a vertical drop, could bound a non-mechanistic tipover case.

NRC staff has accepted an unyielding surface for determining the bounding cask deceleration loads that can far exceed the decelerations experienced by a cask dropping onto or tipping over the concrete storage pad that will bend and deform. Prototype or scale model testing can be used to obtain more realistic cask deceleration or equivalent load for quasi-static analyses. Alternatively, applicants can develop an analytical model to calculate cask deceleration loads. In the analytical approach, the hard receiving surface for a drop or tipover accident need not be an unyielding surface, and its flexibility may be included in the modeling.

The structural discipline should review validation of the analytical model. The staff has completed a series of low-velocity impact tests of a steel billet from which a model validation approach and corresponding acceptance criteria have been developed. These tests and analytical evaluations are summarized in NUREG/CR-6608, *Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet Onto Concrete Pads* (Witte, 1998). On the basis of the report, the following model validation acceptance criteria apply to a cask-pad-soil analytical model for predicting impact responses of the cask:

- When solid steel billet is used to replace the cask in the cask-pad-soil analytical model, it should predict a pulse amplitude slightly higher than the recorded pulse amplitude from the billet test.
- The calculated pulse duration and shape should be similar, but not necessarily identical, to those recorded from the billet test.

The validated billet-pad-soil model is considered adaptable to a cask-pad-soil analysis model if relevant attributes of the cask are used to replace those of the billet.

(b) Explosive Overpressure (LOW Priority)

Explosion-induced overpressure and reflected pressure may result from explosion hazards associated with explosives and chemicals transported by rail or on public highways, natural gas pipelines,

2859 and vehicular fires of equipment used in the transfer of casks.
2860 Explosions may result from detonation of an air-gaseous fuel
2861 mixture. With the exception of transfer vehicle accidents, the
2862 explosion hazards are typically similar to those for facilities subject
2863 to reviews under 10 CFR Part 50, "Domestic Licensing of
2864 Production and Utilization Facilities."

2865
2866 The SAR should state the level of overpressure that the cask
2867 system can withstand for this accident condition. This
2868 overpressure level would then serve as the quantitative envelope
2869 for future comparison with hazards for specific site installations.
2870 The pressure criteria for the assumed design-basis wind or
2871 tornado may also serve as an envelope for the explosive
2872 pressures for comparison with actual site hazards of a general
2873 licensee's facility.

2874
2875 If the SAR includes bounding explosion effects for which the cask
2876 system is to be approved, the reviewer should verify that the
2877 applicant also provided structural analyses of those effects for
2878 cask system structures that may be affected. The SAR should
2879 identify the maximum response determined. That response
2880 should be sufficiently low such that while damage may occur, it
2881 would not impair the capability of the component to perform its
2882 safety functions. In addition, the SAR should identify any post-
2883 event inspection and remedial actions that may be necessary.

2884
2885 (c) Fire (LOW Priority)

2886
2887 Chapter 4, "Thermal Evaluation" of this SRP addresses potential
2888 fire conditions. Fire-related structural evaluation considerations
2889 include increased pressures in the confinement cask, changes in
2890 material properties, stresses caused by different coefficients of
2891 thermal expansion and/or temperatures in interacting materials,
2892 and physical destruction.

2893
2894 The reviewer should evaluate the discussion in the SAR
2895 concerning the treatment of structural effects associated with the
2896 presumed fire. The reviewer should evaluate the appropriateness
2897 of the applicant's analysis of those structural effects for the
2898 assumed parameters of the design-basis fire. The reviewer
2899 should confirm that the applicant defined the confinement cask
2900 pressure capacity on the basis of the cask material properties at
2901 the temperature resulting from the fire.

2902
2903 (d) Flood (LOW Priority)

2904
2905 The applicant's evaluation of the DSS design should be reviewed
2906 with regard to the structural consequences of a flood event. The
2907 SAR may stipulate an assumption that the DSS not be used at
2908 any site where there is potential for flooding. In this case, the
2909 DSS would have to be placed at an ISFSI site above the

2910 maximum probable flood level (SAR Chapter 12, "Accident
2911 Analyses Evaluation" should state this condition). Alternatively, an
2912 application for a certificate of compliance to use a DSS at a site
2913 with flooding potential would require a full analysis for a defined
2914 flood event.

2915
2916 If a design flood event is defined for the certificate of compliance
2917 the reviewer should verify that the SSCs meet appropriate
2918 guidance in RG 1.59, Rev. 2 and 1.102, Rev. 1 for that level of
2919 flood protection.

2920
2921 One possible structural consequence of a flood is that a vertically
2922 stored cask may tip over or translate horizontally (slide) because
2923 of the water velocity. Another possible consequence is that
2924 external hydrostatic pressure will exceed the capacity of the cask.
2925 The applicant may state the critical water velocity and hydrostatic
2926 pressure as bounds for the SAR flood analysis.

2927
2928 The NRC accepts the evaluation for flooding events when the
2929 flood conditions for overturning and sliding of stored confinement
2930 casks and other cask system structures (with a safety factor of 1.1
2931 for accidents cases) have been applied. The applicant should
2932 state the basis for estimation of lateral pressure on a structure as
2933 a result of water velocity.

2934
2935 The NRC accepts the use of Hoerner's *Fluid-Dynamics Drag*
2936 (Hoerner, 1965) for estimating drag coefficients and net lateral
2937 water pressure. An approach for calculating the velocity
2938 corresponding to the cask stability limit is to assume that the cask
2939 is pinned at the outer edge of the cask bottom and rotates about
2940 that outer edge, and the pinned edge does not permit sliding. The
2941 overturning moment from the velocity of the flood water can be
2942 compared to the stability moment of the cask (with buoyancy
2943 considered). The structural consequences of the flood event are
2944 typically bounded by analyses for the drop or tipover accident
2945 cases.

2946
2947 The analysis of the confinement cask should be reviewed for
2948 flood-related hydrostatic pressure. The analysis should include
2949 the combined effects of weight, external hydrostatic pressure,
2950 internal pressure(s), and thermal stress. Resistance of the
2951 confinement cask to flood-related hydrostatic pressure should be
2952 analyzed in accordance with Section III, Subsection NB or NC, of
2953 the ASME B&PV Code (depending on the subsection used for
2954 design).

2955
2956 Additional flood consequences include potential scouring under a
2957 foundation, damage to access routes, temporary blockage of
2958 ventilation passages with water, blockage of ventilation passages
2959 and interstitial spaces between the confinement cask and
2960 shielding structure with mud, and steep temperature gradients in

2961 the shielding structure and confinement cask. The consequences
2962 of these conditions should be analyzed in the SAR and identified
2963 in the certificate of compliance so a general licensee will be able
2964 to consider these factors when siting an ISFSI.

2965 (e) Tornado Winds (LOW Priority)

2966 The reviewer should verify that the SAR addresses the potential
2967 structural consequences of design-basis tornado or extreme wind
2968 effects. The load combination analyses should be reviewed for
2969 acceptable inclusion of tornadoes and tornado missiles. Current
2970 NRC guidance provided in RG 1.76, Rev. 1, recognizes three
2971 regions in the contiguous United States each with distinct design-
2972 basis tornado parameters. The applicant for a certificate of
2973 compliance must clearly define the boundary conditions of the
2974 proposed cask system with respect to these regions or utilized
2975 Region 1.

2976 Confinement casks may be vulnerable to overturning and/or
2977 translation caused by the direct force of the drag pressure while in
2978 storage or during transfer operations. Criteria for resistance to
2979 overturning or sliding should be provided in the SAR.

2980 Confinement casks are generally not vulnerable to damage from
2981 overpressure or negative pressure associated with tornadoes or
2982 extreme winds. However, they may be vulnerable to secondary
2983 effects, such as wind-borne missiles (see (f), below) or collapse of
2984 a weather enclosure, if used. The capability and behavior of the
2985 cask system under the collapse of any such external structure, if
2986 allowed by the Certificate of Compliance should be identified in
2987 the SAR.

2988 Tornadoes typically produce the greatest “design-level” wind
2989 effects for American sites. However, there are some potential
2990 American sites at which high winds may be more severe than the
2991 credible tornado. The SARs for a limited set of potential sites
2992 could reflect high wind effects as a basis for structural analysis. If
2993 the certificate is to include proven design resistance to tornadoes
2994 or extreme winds, the SAR documentation must identify the wind
2995 levels (e.g., in miles or kilometers per hour), source (tornado or
2996 high wind), and specific wind-driven missiles (shape, weight, and
2997 velocity) for which the design is to be evaluated.

2998 RG 1.76, Rev. 1, “Design-Basis Tornado for Nuclear Power
2999 Plants,” provides applicable tornado-related parameters. The
3000 NRC accepts the use of ASCE 7 for conversion of wind speed to
3001 pressure and for typical building shape factors. Conversion of
3002 tornado or other wind speeds to pressure in the SAR
3003 documentation should assume that the cask system is at sea
3004 level.

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The reviewer should verify that the cask system design meets appropriate guidance in the RG 1.76, Rev. 1, and 1.117, Rev. 1, and NUREG-0800 "Standard Review Plan for Power Reactors," Section 3.3.2, Rev. 3 for tornado protection.

Tornadoes and high winds can produce a significant negative pressure differential between interior spaces and the outside in a storage cask system that should be considered. 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 the SAR to separately state negative pressure to establish an envelope for approval since negative pressure is insignificant with regard to confinement cask accident pressure analysis.

The NRC does not accept the presumption 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 onsite transfer is considered by the staff to be a design-basis event. The tornado analysis should determine if tornado-induced overturning is bounded by drop and tipover cases. In addition, the SAR should show that the cask system will continue to perform its intended safety functions (i.e., criticality, radioactive material release, heat removal, radiation exposure, and ready retrievability).

(f) Tornado Missiles (LOW Priority)

The applicant's evaluation of the cask system design should be reviewed with regard to the structural consequences of wind-driven missile impact (RG 1.76, Rev. 1 and NUREG-0800, "Standard Review Plan for Power Reactors," Section 3.5.1.4 (Rev. 3) and Section 3.5.3 (Rev. 3) describe the effects of tornado missiles). The SAR should define the missile parameters for which the cask system is to be evaluated based on the three tornado regions currently identified in the RG 1.76, Rev. 1. Among the possible missile effects, the SAR should address those that may result in a tipover and those that may cause physical damage as a result of impact. The damage should not result in unacceptable radiation dose or significantly impair either criticality control, heat removal, or the ready retrievability of the fuel.

The NRC has accepted use of the analytical approaches given in U.S. Reactor Containment Technology, ORNL-NSIC-5, Volume 1, Chapter 6 (Cottrell and Savolainen), for estimating the potential effects of missile impact on steel sheets, plates, and other structures. Further guidance on analytical acceptable approaches for use in ISFSI design is provided in NUREG-0800, Section 3.5.3, "Barrier Design Procedures." In addition, for analysis and design regarding the ability of reinforced concrete structures to resist

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missiles, the NRC has accepted use of “Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects” (Kennedy, 1975).

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. Moreover, the capability of the SSC to perform their safety functions should not be impaired.

(g) Earthquake (MEDIUM Priority)

The applicant’s evaluation of the cask design should be reviewed with regard to the structural consequences of the earthquake event. Cask designs must satisfy the load combinations that encompass earthquake, including those for sliding and overturning. The applicant should demonstrate that no tipover or drop will result from an earthquake. In addition, impacts between casks should either be precluded, or should be considered an accident event for which the cask must be shown to be structurally adequate.

Appendix H of ANSI/ANS-57.9-1992 provides guidance for seismic analysis. Implicit in this guidance is the assumption that the ISFSI concrete pad, supported by soil, behaves as a rigid mat and therefore possesses no out-of-plane flexibility. This is valid for the majority of nuclear power plant structures where relatively thick mats support integral reinforced concrete walls. However, ISFSI pads are usually relatively thin structures (i.e., small thickness to length ratio) and generally do not incorporate integral walls to stiffen the pad. While the cask itself is relatively rigid, the rigid cask resting on a flexible pad has a lateral mode frequency that is generally low enough to fall within the amplified range of most design earthquake spectra. Thus, in determining the inertia forces that act at the center of gravity of the cask for the purpose of evaluating the onset of sliding or tipping, the reviewer should ensure that the applicant has either accounted for the out-of-plane flexibility of the pad in the seismic analysis or demonstrated that it is not an important parameter in determining the response of the cask, (“Influence of ISFSI Design Parameters on the Seismic Response of Dry Storage Casks,” Bjorkman & Moore, 2001).

The reviewer should verify that the cask system design meets appropriate guidance in RGs 1.29, Rev. 4, 1.60, Rev. 1, 1.61, Rev. 1, and 1.92, Rev. 2, for protection against seismic events.

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 outer shield and the rocking and fall back of a vertically or horizontally oriented confinement cask on its supports.

3114
3115 Cask systems are not required to survive a design earthquake
3116 without permanent deformation. However, the maximum extent of
3117 damage from a design earthquake must be predicted, and the
3118 capability to provide principal safety functions should not degrade.
3119

3120 ii. Structural Analysis Methods
3121

3122 (LOW Priority) The applicant's structural analysis of various loading combinations
3123 and the resulting stresses, strains, and deformations from different loads should
3124 be reviewed. The reviewer should verify that the applicant properly used
3125 acceptable analytical approaches and tools. In addition, the applicant should
3126 have performed and reviewed the associated computations internally under an
3127 acceptable independent design review (equivalent to ASME NQA-1) and quality
3128 assurance procedures. The scope of the staff's review may include performing
3129 detailed parallel computations (such as finite element analyses) to validate
3130 submitted computations or their results. The reviewer may perform separate,
3131 less extensive calculations when these could most readily evaluate any
3132 suspected problems.
3133

3134 The applicant's analysis of loads and load combinations resulting from different
3135 structural conditions should be consistent with the code or criteria requirements
3136 used in designing the component.
3137

3138 Subsection NB or NC of the ASME B&PV Code defines the requirements for
3139 categorizing stresses and determining allowable stress limits for confinement
3140 casks. These references also provide definitions of stress categories and stress
3141 intensity limits for normal and off-normal operating conditions. For Level D or
3142 accident conditions, Appendix F to the ASME B&PV Code provides definitions of
3143 the stress intensity limits.
3144

3145 In accordance with these references, stress intensity is defined on the basis of
3146 the maximum shear stress theory for ductile materials. Since the maximum
3147 shear stress is not identical to the maximum octahedral shear stress, octahedral
3148 shear stresses should not be compared with the stress intensity limits. Values
3149 for the stress intensity limits are defined in Appendices I and III of the ASME
3150 Code. Stresses resulting from inertial and pressure loads should be considered
3151 primary stresses. Thermal stresses resulting from temperature gradients may be
3152 considered secondary stresses if they are self-limiting and do not cause
3153 structural failure. Stresses due to thermal gradients in fuel baskets may not be
3154 self-limiting and should be considered by the applicant because of the possibility
3155 of uneven heat loadings of adjacent assemblies as well as the effects of
3156 asymmetry in the basket structure.
3157

3158 (1) Finite-Element Analyses (HIGH Priority)
3159

3160 Because of the complexity of many structural design considerations and
3161 load conditions, structural design computations are often performed using
3162 finite-element analysis.
3163

3164 The applicant should perform the finite-element analyses using a general-
3165 purpose program that is well benchmarked and widely used for many
3166 types of structural analyses.

3167
3168 To be consistent with the provisions in Section III of the ASME Code, the
3169 analyses should use linear material properties. For materials that do not
3170 serve in a structural capacity (such as shielding materials), inelastic
3171 material properties may be used for cask components that are not stress-
3172 limited and respond inelastically to the load conditions for storage casks.
3173 The SAR should identify the sources used for the inelastic material
3174 properties.

3175
3176 Lead shielding can be modeled either with elastic or inelastic properties.
3177 The elastic modulus and limit used for lead in the elastic analysis should
3178 be determined on the basis of the potential temperature of the material.
3179 An appropriate plasticity model of lead can be used to account for its
3180 inelastic behavior.

3181
3182 Nonstructural components of the confinement cask are generally not
3183 included in finite element models. However, the models should include
3184 any influence these nonstructural components may have on the structural
3185 performance of the cask. Possible influences include the nonstructural
3186 components' inertial weight, restraint to motion of the structural
3187 components, and localized influence on load applications because of
3188 geometrical effects.

3189
3190 Bolted connections can be modeled either discretely or with contact
3191 conditions. To discretely model the bolted connections, the applicant
3192 should use appropriate element types and material properties. With
3193 contact conditions, the interfaces joined by the bolts can be modeled as
3194 tied.

3195
3196 Verify that the applicant has provided information on any computer-based
3197 modeling as described in Appendix 3A to this chapter, and review the
3198 structural analyses submitted by the applicant in accordance with the
3199 Appendix.

3200
3201 (2) Closed-Form Calculations (MEDIUM Priority)

3202
3203 The applicant should perform closed-form calculations for relatively
3204 simple structural load conditions or conditions for which a formula has
3205 been developed. Closed-form calculations are also typically used to
3206 check the results of finite-element analyses. In addition, this type of
3207 calculation can be used for analyses involving principles of conservation
3208 of energy and comparisons of overturning moments.

3209
3210 One source of closed-form equations accepted by the NRC is *Formulas*
3211 *for Stress and Strain* (Roark, 1965). Use of a particular equation or
3212 formulation for the load conditions should be justified. The most
3213 important aspect of the calculations to evaluate is the basis for the
3214 assumptions used in the calculations. In many cases, the calculations

3215 are faulty in that they fail to include portions of the cask, or the load
3216 conditions are idealized inappropriately.

3217
3218 To be consistent with the provisions in Section III of the ASME Code, the
3219 analyses should use linear material properties. Linear analysis should be
3220 the basis for all closed-form calculations.

3221
3222 (3) Structural Analysis for Specific Cask Components

3223
3224 The following paragraphs present a few specific examples of structural
3225 analysis for some of the confinement cask components of a cask storage
3226 system.

3227
3228 (a) Fuel Basket (HIGH Priority)

3229
3230 The fuel basket design should be reviewed to assess the
3231 applicant's analysis of the combined effects of weight, thermal
3232 stresses, and cask-drop impact forces that could arise during
3233 spent fuel transfer and storage operations. The weight supported
3234 by the basket should be the maximum or design weight of the
3235 SNF to be stored. In addition, the applicant should evaluate all
3236 credible potential orientations of the cask and basket during cask
3237 transfer and handling drops while transferring the spent fuel into
3238 storage. End or side drops typically produce the greatest
3239 structural demand on various basket components. In an end drop,
3240 the basket is supported by the bottom of the confinement cask
3241 cavity upon impact. In the side drop, the basket structure and
3242 points of contact with the confinement cask must support the
3243 mass of the basket and loaded fuel.

3244
3245 In previous DSS evaluations, the NRC has accepted two
3246 approaches for analyses regarding the structural capability of the
3247 basket to acceptably survive a cask drop during transfer and
3248 storage. The first approach uses dynamic analyses in a two-step
3249 process. In Step 1, the applicant performs a dynamic analysis of
3250 the cask body impacting a target surface and assesses the
3251 performance of the cask body, including determining the time-
3252 history response from the cask drop impact. In Step 2, this time-
3253 history response can be translated into a forcing function that can
3254 be applied to the supporting contact points of an appropriate
3255 model of the fuel basket.

3256
3257 The second approach uses a quasi-static analysis of the basket
3258 subjected to the equivalent acceleration inertial load derived from
3259 the cask-drop impact analysis. In this analysis, the applicant
3260 should apply the equivalent acceleration inertial load using an
3261 appropriate model of the basket with the location(s) most
3262 vulnerable to the impact. Support provided by the inside surface
3263 of the cask cavity should be represented by the appropriate
3264 boundary conditions on the outside edge of the basket. In
3265 addition, the applicant should conservatively select the equivalent

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acceleration inertial load 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 applicant should also evaluate the buckling capacity of the cask basket materials. Acceptable guidance for this evaluation is provided in Section III of the ASME B&PV Code and NUREG/CR-6322, "Buckling Analysis of Spent Fuel Basket," (Lee and Bumpas, 1995). For this evaluation, the applicant should select the appropriate end conditions used in the buckling capacity equations on the basis of sensitivity studies. These studies can bound the range of conditions that are typically either fixed for a welded connection or free if there is no rigid connection.

(b) Closure Lid Bolts of Confinement Boundary (MEDIUM Priority)

The design analysis for the closure-lid bolts should be reviewed to ensure that it properly includes the combined effects of weight, internal pressure(s), thermal stress, O-ring compression force, cask impact forces, and bolt pre-load. Typically, applicants specify the pre-load and bolt torque for the closure bolts on the basis of 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 direct tensile force on the bolts as well as 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 should then be compared with the tensile bolt force computed from the pre-load and operating temperature load alone. The larger of the two calculated tensile forces should control the design. The maximum design bolt force should then be obtained by combining the larger direct tensile bolt force with the additional prying force. The weight is derived from the maximum or design weight of the closure lids and any cask components supported by the lids. Acceptable analytical methods for closure bolts are given in NUREG/CR-6007, "Stress Analysis of Closure Bolts for Shipping Casks" (Mok and Fischer, 1993).

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.

(c) Trunnions (LOW Priority)

The design of the trunnions, their connections to the cask body, and the cask body in the local area around the trunnions should be reviewed. The design basis for of the trunnions can be either non-redundant or redundant. In either case, the design should

3317 meet the requirements of ANSI N14.6 for critical loads and the
3318 requirements of NUREG-0612, "Control of Heavy Loads at Power
3319 Plants."

3320
3321 Non-redundant lifting systems should be designed for not less
3322 than 6 times the material yield strength and 10 times the material
3323 ultimate strength given the design lift weight of the loaded cask.
3324 Redundant lifting systems should be designed for not less than
3325 3 times the material yield strength and 5 times the material
3326 ultimate strength given the design loaded lift weight of the cask.
3327 Acceptance testing requirements for trunnions are discussed in
3328 Chapter 10, "Acceptance Tests and Maintenance Program
3329 Evaluation," of this SRP.

3330
3331 For a typical trunnion design, the maximum stress occurs at the
3332 base of the trunnion as a combination of bending and shear
3333 stresses. A conservative technique for computing the bending
3334 stress is to assume that the lifting force is applied at the
3335 cantilevered end of the trunnion, and that the stress is fully
3336 developed at the base of the trunnion. If other assumptions are
3337 used, the applicant should provide adequate justification. In
3338 addition, the applicant should evaluate the stresses and forces in
3339 the trunnion connections with the cask body and in the cask body
3340 near the trunnions.

3341
3342 iii. Structural Evaluation

3343
3344 (1) Structural Capability (LOW Priority)

3345
3346 The applicant's structural analyses should be reviewed to assess the
3347 information regarding margins of safety or compliance with ASME Code
3348 stress limits, overturning margins, and other criteria appropriate for the
3349 division of the ASME Code being used. The comparisons of capability
3350 versus demand for the various applicable loading conditions should be
3351 presented in the same terms used in the design code (e.g., type of
3352 stress). In addition, margins of safety should be included on the basis of
3353 comparisons between capacity and demand for each of structural
3354 component analyzed. The minimum margin of safety for any structural
3355 section of a component should be included for the different load
3356 conditions.

3357
3358 (2) Fabrication and Construction (MEDIUM Priority)

3359
3360 The NRC has accepted fabrication of metallic confinement casks in
3361 accordance with Section III, Division 1 of the ASME B&PV Code. If the
3362 fabrication, construction, or assembly deviate in any way from the
3363 subsection of this standard used for design, the SAR must explicitly state
3364 the applicant's justification for the deviation, and the justification must be
3365 acceptable to the NRC.
3366

3367 If the design of the confinement cask is proposed to be governed by
3368 ASME, Section III, Division 2, similar to a metallic-lined concrete pressure
3369 vessel NRC would expect the fabrication/construction of such a cask to
3370 also be governed by the Division 2 requirements. Any deviations from the
3371 Code requirements should be addressed as noted for Division I above for
3372 metallic containment.

3373
3374 If the design of the confinement cask is proposed to be governed by
3375 ASME, Section III, Division 3, the applicant will have to provide
3376 supplemental details to the Code provisions since Subsection WC does
3377 not provide guidance to address all construction details for classic
3378 containments.

3379
3380 **3.5.2 Other System Components and Structures Important to Safety**

3381
3382 3.5.2.1 Scope

3383
3384 This portion of the DSS structural review provides guidance by addressing procedures for
3385 evaluating all structures that are important to safety (as defined in 10 CFR Part 72.3), whether
3386 steel, concrete or other material not addressed as the confinement cask and internals
3387 (Subsection 3.5.1). Structures may include items such as gamma and neutron shielding,
3388 overpack material, any respective encasement foundations, structural supports, ventilation
3389 passages, weather enclosures, earth retention structures, and protective structures. This
3390 evaluation should include drawings, plans, sections, and technical specifications for these
3391 SSCs.

3392
3393 3.5.2.2 Structural Design Criteria and Design Features

3394
3395 i. Design Criteria (MEDIUM Priority)

3396
3397 (1) General Structural Requirements

3398
3399 Structural requirements are driven by the functional roles of the system
3400 components and the need to maintain safety. Safety requirements are
3401 expressed in the referenced rules, standards, and codes and as criteria
3402 specific to the component. The basic safety requirements are that the
3403 structural and functional design must preclude the following:

- 3404
3405 • Unacceptable risk of criticality.
3406
3407 • Unacceptable release of radioactive materials to the environment.
3408
3409 • Unacceptable radiation dose to the public or workers.
3410
3411 • Significant impairment of ready retrievability of stored nuclear
3412 materials during normal and off-normal conditions.

3413
3414 The applicant should consider the potential for liquefaction and other soil
3415 instabilities attributable to vibrating ground motion, for any structure or
3416 system component such as a cask system support pad.

3417

3418 Reinforced concrete pads that support confinement casks in storage do
3419 not constitute “pavements.” As such, they should be designed and
3420 constructed as foundations under an applicable code such as, ACI 349,
3421 ACI 318, or IBC. Such pads typically are not classified as important to
3422 safety; however, in some cases they may be.

3423
3424 Steel embedments in reinforced concrete structures must satisfy the
3425 requirements of the design code applicable to the reinforced concrete
3426 structure. Similarly, structural steel must satisfy the requirements of the
3427 applicable steel design code (e.g., ASME B&PV Code, AISC, or other
3428 identified code).

3429
3430 (2) Applicable Codes and Standards

3431
3432 The codes and standards identified in the SAR should be reviewed as
3433 well as their proposed applications. This subsection addresses the codes
3434 and standards that the NRC has accepted for structures important to
3435 safety categorized by application that are not confinement casks or the
3436 steel internals.

3437
3438 The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and
3439 standards cited therein) as the basic reference for the structures
3440 important to safety that are not designed in accordance with the Section
3441 III, Division 1 or Division 2 of the ASME B&PV Code. However, both the
3442 lifting equipment design and the devices for lifting system components
3443 that are important to safety must comply with ANSI Standard N14.6. The
3444 NRC accepts the load combinations shown in Table 3-3 for structures not
3445 designed under either Section III of the ASME B&PV Code Section III,
3446 Division 1 or 2 (ACI 359). See Table 3-2 for loads and their descriptions.

3447
3448 The reviewer should review the suitability of the applicant’s identification
3449 of codes and standards that are to be met by the structural design and
3450 construction of other components subject to NRC approval. The principal
3451 codes and standards include the following references that may apply to
3452 steel structures and components as well as concrete portions of the cask
3453 system:

3454
3455 • AISC, “Specification for Structural Steel Buildings – Allowable
3456 Stress Design and Plastic Design.” The NRC has not yet received
3457 any applications that propose a steel design on the basis of the
3458 AISC’s “Load and Resistance Factor Design (LRFD) Specification
3459 for Structural Steel Buildings.” If such a design was received, the
3460 NRC would evaluate the proposal for compliance with the load
3461 combinations summarized in Table 3-3 and for consistent
3462 application of the LRFD design methodology.

3463
3464 • To date, the NRC has not required applicants to design or build
3465 structural steel components of a cask system important to safety
3466 in compliance with ANSI/ANS N690, “Nuclear Facilities — Steel
3467 Safety-Related Structures for Design Fabrication and Erection.”
3468

- 3469 • AWS D1.1, "Structural Welding Code Steel."
- 3470
- 3471 • ASCE 7, "Minimum Design Loads for Buildings and Other
- 3472 Structures."
- 3473
- 3474 • ACI 349, Appendix D, for anchoring to concrete or Section 10.14
- 3475 for composite compression sections, as applicable, when
- 3476 constructed of structural steel embedded in reinforced concrete.
- 3477 Where requirements do not conflict, the steel must also comply
- 3478 with the requirements of the codes stated above. In addition, ACI
- 3479 349 defines constraints for obtaining ductile response to extreme
- 3480 loads by ensuring that the strength of steel embedments controls
- 3481 the design; these constraints must not be subverted by over-
- 3482 design of the steel.
- 3483
- 3484 • For reinforced concrete the NRC has not accepted the use of a
- 3485 set of criteria selected from multiple standards and codes, except
- 3486 when the selected criteria meet the most limiting requirements of
- 3487 each code. However, in recognizing a graded approach to quality
- 3488 assurance, the NRC has approved the use of ACI 349 for design
- 3489 and material selection for reinforced concrete structures important
- 3490 to safety (not confinement). The NRC has allowed the optional
- 3491 use of ACI 318 as an alternative standard for construction as
- 3492 described below.
- 3493
- 3494 • In both cases, however, the design, material selection and
- 3495 specification, and construction must also meet any additional or
- 3496 more stringent requirements given in ANSI/ANS-57.9.
- 3497
- 3498 The following paragraphs identify the portions of ACI 349 that
- 3499 apply to design (including material selection) and must be met by
- 3500 applicants who choose to use ACI 318 for construction. (The
- 3501 paragraph references are as in ACI 349-06.). Unlisted and
- 3502 excepted sections address construction requirements for which
- 3503 the NRC accepts substitution of ACI 318.
- 3504
- 3505 Chapter 1 "General Requirements," Sections 1.1 and 1.5
- 3506 (except references to construction), and Sections
- 3507 1.2 and 1.4.
- 3508 Chapter 2 "Definitions."
- 3509 Chapter 3 "Materials" (except Sections 3.1, 3.2.3, 3.3.3,
- 3510 3.5.3.1.1, 3.6.1.0, and 3.7).
- 3511 Chapter 4 "Durability Requirements"
- 3512 Chapter 6 "Form Work, Embedded Pipes, and Construction
- 3513 Joints," Sections 6.3.13, 6.3.14, and 6.3.15.
- 3514 Chapter 7 "Details of Reinforcement."
- 3515 Chapter 8 "Analysis and Design General Considerations."
- 3516 Chapter 9 "Strength and Serviceability Requirements."
- 3517 Chapter 10 "Flexure and Axial Load."
- 3518 Chapter 11 "Shear and Torsion."
- 3519 Chapter 12 "Development and Splices of Reinforcement."

3520	Chapter 13	“Two-way Slab Systems.”
3521	Chapter 14	“Walls.”
3522	Chapter 15	“Footings.”
3523	Chapter 16	“Precast Concrete.”
3524	Chapter 17	“Composite Concrete Flexural Members.”
3525	Chapter 18	“Prestressed Concrete.”
3526	Chapter 19	“Shells.”
3527	Appendix A	“Strut-and-Tie Models.”
3528	Appendix D	“Anchoring to Concrete.”
3529	Appendix E	“Thermal Considerations.”
3530	Appendix F	“Special Provisions for Impulsive and Impactive Effects” (except that the load combinations included herein, must be used.
3531		
3532		
3533		

3534 For fluid systems used with a cask system that may be connected
3535 to a penetration of the confinement barrier outside an enclosing
3536 structure licensed under 10 CFR Part 50 (e.g., the fuel pool
3537 building), the NRC accepts construction consistent with
3538 requirements comparable to those used for Quality Group C, as
3539 shown in RG 1.26, “Quality Group Classifications and Standards
3540 for Water-, Steam-, and Radioactive Waste-Containing
3541 Components of Nuclear Power Plants,” Rev. 4 and
3542 NUREG-0800,” Section 3.2.2, “Standard Review Plan for Nuclear
3543 Power Plants.” In this context, “construction” includes materials,
3544 design, fabrication, examination, testing, inspection, and
3545 certification required in the manufacture and installation of
3546 components. Quality Group D may, under some circumstances
3547 be justified.
3548

3549 Quality Group C requires construction of piping, pumps, valves,
3550 atmospheric storage tanks, and 0-15 psig storage tanks in
3551 conformance with Section III of ASME B&PV Code 1, Class 3
3552 (Subsection ND). In addition, Quality Group C requires that
3553 supports for these components meet the requirements of
3554 Subsection NF.
3555

3556 By contrast, Quality Group D requires compliance with the
3557 following codes, as a minimum:
3558

3559 Piping: ANSI/ASME B31.1, “Power Piping.”

3560 Pumps: Manufacturer’s Standards.

3561 Valves: ANSI/ASME B31.1 and ANSI B16.34, “Valves.”

3562 Atmospheric Storage Tanks:
3563 American Water Works Association (AWWA),
3564 “Standard for Steel Tanks — Standpipes,
3565 Reservoirs, and Elevated Tanks for Water Storage”
3566 (AWWA D100) or ANSI/ASME B96.1, “Specification
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for Welded Aluminum-Alloy Field-Erected Storage Tanks.”

0–15 psig Storage Tanks:
American Petroleum Institute’s (API)
“Recommended Rules for Design and Construction
of Large, Welded, Low-Pressure Storage Tanks”
(API 620).

The NRC accepts the “Boundaries of Jurisdiction” applicable to Section III, Subsections NB-1130 and NC-1130, of ASME B&PV Code. These boundaries apply to attachments to penetrations of the confinement barrier outside an enclosure licensed under 10 CFR Part 50. Specifically, these boundaries define whether the attachments must be designed, fabricated, and installed in accordance with Section III, Subsection NB or NC, of ASME B&PV Code.

Note that codes, other than those discussed herein (e.g., the “Electric, Life Safety, and Lightning Protection Codes” promulgated by the National Fire Protection Association [NFPA]), may apply to the design and construction of the cask system. It is acceptable to include such codes in the design by inclusion in the SAR. Where designs of structures subject to approval are also covered by such other codes, the review should include evaluation of compliance with those codes.

The NRC has not yet received any applications for licensing or approval of a cask system that included masonry important to safety. Masonry is not considered suitable for confinement, but it may be acceptable for enclosures and physical or radiation-shielding applications.

ii. Structural Design Features (MEDIUM Priority)

The design description in the SAR documentation should be reviewed to ensure that it defines the functional performance required of the structures. The design description of the non-confinement safety-related structures of the cask system should provide a clear understanding to be reached by the reviewer of the significance of the safety-related features to the required performance.

The SAR documentation should also be reviewed regarding the physical design of the structures important to safety. This should include the following as a minimum. As appropriate to the specific structure the following information should be provided.

- Dimensioning of all structural elements.
- Locations, sizes, configuration, spacing, welding, fasteners etc. of the safety-related non-confinement structures should be provided.

- 3621 • Locations and specifications for controls, that will be necessary in
3622 fabrication and construction.
- 3623
- 3624 • Structural materials with defining standards or specifications summarized
3625 or references to Chapter 8, "Materials Evaluation" of this SRP herein
3626 should be reviewed.
- 3627
- 3628 • Information on the physical design of attachments, embedments, and
3629 other structural elements should be provided.
- 3630

3631 Auxiliary cask system equipment important to safety has often been specially
3632 designed. In particular, the structural design features that provide for safety
3633 should be supported by design or operational analysis. This analysis should
3634 demonstrate that the equipment will meet the basic safety criteria, regardless of
3635 problems that may occur in mechanical, electrical, human operator, or other
3636 operations.

3637

3638 The NRC has accepted and approved cask system designs that depend on the
3639 operation of new mechanical systems for system use. NRC approval does not
3640 certify that the mechanical systems will operate as projected but rather that
3641 proper functioning is necessary to successfully complete a specified operation.
3642 Such approval reflects a finding by the NRC staff that, regardless of the system's
3643 success (or lack thereof) in mechanical operation, the basic safety criteria will be
3644 met, as stated above.

3645

3646 The proposed system design should be reviewed against planned normal and
3647 off-normal, operations and accidents. The reviewer should determine whether
3648 the structural design of the equipment provides for continuing satisfaction of the
3649 basic safety criteria. The reviewer should consider that the equipment could fail
3650 to operate at any time (i.e., during operations at the physical limits of speed or
3651 range, or during a credible, off-normal, or accident-level event).

3652 3.5.2.3 Structural Analysis

3653 Subsections 3.5.1.4 (i) and (ii) provide guidance regarding structural analysis for the
3654 confinement cask and metallic internals of cask systems. These subsections provide
3655 supplemental guidance primarily related to steel and concrete structures, other than the
3656 confinement cask and its contents and integral components that are important to safety. The
3657 appropriateness, completeness, and correctness of the applicant's proposed implementation of
3658 these load conditions and combinations for the metallic and reinforced concrete structures
3659 should be reviewed.

3660 i. Load Conditions (MEDIUM Priority)

3661

3662 The load definitions and combinations shown in Tables 3-2 and 3-3 have been
3663 accepted by the NRC for analysis of steel and reinforced concrete ISFSI
3664 structures that are important to safety. These load combinations are included in
3665 or derived from ANSI/ANS 57.9 and ACI 349.

3666

3667 Structures that are important to safety should have sufficient capability for every
3668 section to withstand the worst-case loads under normal and off-normal
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conditions. Such capability ensures that these structures will not experience permanent deformation or degradation of the capability to withstand any future loadings.

The NRC accepts the load combinations in Table 3-3 that implement and supplement those of ANSI/ANS-57.9.

(1) Normal Conditions

The SAR documentation should be reviewed to ensure adequate inclusion of the following conditions that may be of particular concern for concrete structures important to safety if the loading condition is appropriate:

- Live and dynamic loads associated with transfer of the confinement cask to and from its storage position and in its storage location for its service lifetime.
- 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 the normal range of operations and ranges of ambient temperature.
- Thermal gradients that may result from impingement of precipitation on highly heated concrete.

(2) Off-Normal Conditions

The SAR should be reviewed to ensure adequate inclusion of the following off-normal operations and events:

- Live and dynamic loads associated with equipment or instrument malfunctions, or accidental misuse during transfer of the confinement cask to and from its storage position.
- Situations in which a confinement cask is jammed or moved at an excessive speed into contact with a reinforced concrete structure.
- The impact of reinforced concrete structures by a suspended transfer, confinement, or storage cask.
- Off-normal ambient temperature conditions (although they may be less severe than accident conditions, these may be of concern because of different sets of factors in the off-normal and accident load combinations, and because concrete temperature limits for off-normal conditions are the same as for normal conditions. Note

3721 that greatly elevated concrete temperatures are allowed for
3722 accident conditions in accordance with ACI 349, Section A.4).

3723
3724 (3) Accident Conditions and Natural Phenomena Events
3725

3726 The SAR should be reviewed for adequate inclusion of the following
3727 conditions associated with accident and conditions that may be of special
3728 concern for reinforced concrete structures:

- 3729 • Loads associated with accidental drops or other impacts during
3730 transfer of the confinement cask to and from its storage position.
- 3731 • Events that produce extreme thermal gradients in the concrete.
- 3732 • Contact caused by earthquake between the confinement cask and
3733 the reinforced concrete structures.
- 3734 • Drop of a closure into position or onto the structure.

3735 The ACI codes are intended to ensure ductile response beyond initial
3736 yield of structural components. ACI 349 also imposes conditions on
3737 design (beyond those of ACI 318) that effectively increase ductility. In
3738 particular, the reviewer should review the proposed reinforced concrete
3739 design to ensure that it provides code levels of ductility by satisfying the
3740 pertinent ACI 349 provisions. Seismic loads are considered to be
3741 "impulsive" and, therefore, are subject to the additional design constraints
3742 of Appendix C to ACI 349. Other accident conditions or natural
3743 phenomenon events may also produce impulsive or impactive loadings
3744 requiring the additional requirements of Appendix F to ACI 349.

3745 Reviewers should check the steel reinforcement schedules and drawings
3746 to ensure that any reinforcing steel quantities, sizes, and locations are
3747 consistent with the design analysis.

3748 In particular, consider the following aspects of the design:

- 3749 • Upper limit (60 ksi, 4219 kgf/cm²) on the specified yield strength of
3750 reinforcement and lower limit (30 ksi, 211 kgf/cm²) on concrete
3751 specified compressive strength (f'c).
- 3752 • Limit on the amount (cross-section area) of compressive
3753 reinforcement in flexural members.
- 3754 • Requirements on continuation and development lengths of tensile
3755 reinforcement.
- 3756 • Specifications for confinement and lateral reinforcement in
3757 compression members, in other compressive steel, and at
3758 connections of framing members.

- 3771 • Aspects of the design that ensure flexure controls (and limits) the
3772 response.
- 3773
- 3774 • Requirements for shear reinforcement.
- 3775
- 3776 • Limitations on the amount of tensile steel in the flexural members
3777 relative to that which would produce a balanced strain condition.
- 3778
- 3779 • Projected maximum responses to design-basis loads within the
3780 permissible ductility ratios for the controlling structural action.
- 3781
- 3782 • Embedments designed for ductile failure and to fail in the steel
3783 before pullout from the concrete.
- 3784

3785 In addition, the construction specifications or descriptions (to the
3786 extent included in the SAR documentation) should be reviewed to
3787 ensure that substitution of materials, use of larger sizes, or
3788 placement of larger quantities of steel will be precluded, and that
3789 provisions for splicing or development of reinforcing steel will not
3790 reduce ductility of the members.

3791
3792 ii. Structural Analysis Methods (HIGH Priority)

3793
3794 The applicant should select and use analytical methods that are appropriate for
3795 the proposed type of materials and construction. In certain instances, however,
3796 the applicant may have to adapt existing analytical methods, codes, and models
3797 for highly specialized cask system equipment designs. Such instances require
3798 special review attention. In particular, the reviewer should ensure that the
3799 adapted approach is fully documented, supported, and acceptable. In addition,
3800 the reviewer should consider the potential for safety-related risk associated with
3801 a possible error in the design of special cask system equipment. The degree of
3802 risk indicates the suitability and acceptability of the adapted approach.
3803 Subsection 3.5.1.4.ii provides acceptable analytical methods of analysis that can
3804 be utilized. Appendix 3A addresses the application of computational modeling
3805 software.

3806
3807 iii. Structural Evaluation (LOW Priority)

3808
3809 In evaluating the variety of cask system equipment and structures that may be
3810 important to safety, the reviewer should ensure compliance with the basic safety
3811 criteria in Subsection 3.5.2.2 (i)(1) and that the specified parameters for
3812 acceptability such as stress, strain or deflection are within the permitted values
3813 identified in Subsection 3.5.2.2.i.(2).

3814
3815 The NRC accepts strength design as presented in the current revision of ACI 349
3816 for reinforced concrete structures important to safety that are not within the scope
3817 of ACI 359. If the applicant uses another design approach, the review conducted
3818 within the scope of the DSS SAR evaluation should include in-depth comparison
3819 of that approach with the provisions of ACI 349.

3820

3821 The NRC accepts the use of guidance in NUREG-0800 for analysis of natural
3822 phenomena, as related to the conditions that apply to the design of cask
3823 systems. However, the load combinations shown in Table 3-3 and the design
3824 and construction requirements of the codes cited above take precedence. The
3825 NRC accepts the American Society of Civil Engineers' "Seismic Analysis of
3826 Safety Related Nuclear Structures" (ASCE 4) and ASCE 7 as the standards for
3827 seismic analysis. In addition, the NRC accepts tornado missile impact analysis in
3828 accordance with Kennedy's *Review of Procedures for the Analysis and Design of*
3829 *Concrete Structures to Resist Missile Impact Effects*.

3830
3831 (1) Structural Capability (LOW Priority)

3832
3833 Section 3.5.1.4.iii (1) addresses the assessment of the structures
3834 capability with respect to the ASME Code stress limits which are
3835 appropriate for metallic structures under Division 1 and for concrete
3836 structures under Division 2.

3837
3838 For other safety related structural concrete, strength (or "ultimate
3839 strength") design is the approach usually used in reinforced concrete
3840 design. Strength design is the only design approach that has been
3841 accepted for reinforced concrete structures that are part of cask systems
3842 not within the scope of ACI 359, and it is the approach used in the current
3843 revisions of ACI 349. This design code was tested and developed on the
3844 basis of extensive empirical experience with concrete construction. The
3845 current strength design approach, as presented in this code, includes
3846 empirically derived requirements and constraints. Determination that a
3847 reinforced concrete structure designed by another approach satisfies
3848 ACI 349 typically requires clause-by-clause review of the code for
3849 compliance. Allowable stress design was formerly used as the basis for
3850 ACI codes related to reinforced concrete design. However, those codes
3851 do not reflect additional experience gained through observations of
3852 structural performance and experimental testing that has since been
3853 included in the current approach to strength design.

3854
3855 With respect to structural steel or other metallic structures important to
3856 safety, but not to the confinement structure or internals, the structural
3857 capability of the design may be based on the ASME Code with the use of
3858 the appropriate subsections as identified in Section 3.5.2.2 (i)(2) herein,
3859 or the AISC specifications also identified. Allowable stress, plastic
3860 design, and load and resistance factor methods of design are acceptable
3861 for use when there is justification for the method used provided in the
3862 application.

3863
3864 (2) Fabrication and Construction (MEDIUM Priority)

3865
3866 For structures and structural components analyzed and designed based
3867 on ASME B&PV Code requirements of Section III, Division 1 or
3868 Division 2, the fabrication and construction provisions of these documents
3869 should form the basis for the production and installation of the structures
3870 and components of the cask storage system.

3871

3872 NRC accepts construction in accordance with ACI 349 or ACI 318.
3873 Selection and validation of the proper concrete mix to meet design
3874 requirements are considered a construction function. By contrast,
3875 specification of cement type, aggregates, and special requirements for
3876 durability and elevated temperatures is considered a design or material
3877 selection function and is, therefore, governed by ACI 349 (and/or ACI
3878 359, if applicable).

3879
3880 The following sections of ACI 318 (chapters, appendix, and
3881 paragraphing per ACI-318-02) have been accepted by the NRC
3882 for construction of ISFSI reinforced concrete structures that are
3883 not within the scope of ACI 359:

3884
3885 Chapter 1 "General Requirements," Sections 1.1.1, 1.1.2,
3886 1.1.3, and 1.1.5 (except references to design and
3887 material properties), and Section 1.3.
3888 Chapter 2 "Definitions" (use ACI 349, Chapter 2).
3889 Chapter 3 "Materials," Sections 3.1 and 3.8 (except A-616,
3890 A-617, A-767, A-775, A-884, and A-934).
3891 Chapter 4 "Durability Requirements."
3892 Chapter 5 "Concrete Quality, Mixing, and Placing."
3893 Chapter 6 "Form Work, Embedded Pipes, and Construction
3894 Joints" (except references to design and material
3895 properties, which are governed by ACI 349).
3896

3897 **3.5.3 Other Structural Components Subject to NRC Approval (MEDIUM Priority)**

3898
3899 3.5.3.1 Scope

3900
3901 The cask system description provided in the SAR may include a variety of components that are
3902 not important to safety such as transporters, ram systems, vacuum drying systems, drain and fill
3903 quick disconnects, support pads and other concrete structures not important to safety. These
3904 components should be reviewed to ensure proper functioning to the extent that the structures
3905 represent required elements of the total cask system. In particular, the reviewer should
3906 evaluate all structures that are proposed for approval in a cask system design acceptable to the
3907 NRC. This evaluation should ensure that the SAR provides sufficient information to confirm the
3908 proper functioning of the components and the overall system. For each system element that is
3909 not important to safety, the reviewer should address the potential response to accidents and
3910 natural phenomenon events to ensure that the given element will not jeopardize the safety
3911 provided by other system elements.

3912
3913 3.5.3.2 Structural Design Criteria and Design Features

3914
3915 i. Design Criteria

3916
3917 (1) General Structural Requirements
3918

3919 Structures subject to approval but not important to safety should be
3920 reviewed on the basis of determining whether the structures can properly
3921 perform their intended function(s). In addition, the NRC review should
3922 ensure that the response of the structures to credible off-normal and
3923 accident conditions will not create secondary hazards for cask system
3924 components or the stored nuclear materials.

3925
3926 (2) Applicable Codes and Standards

3927
3928 The reviewer should review the suitability of the applicant's identification
3929 of codes and standards to be met by the structural design and
3930 construction of other components subject to NRC approval. The principal
3931 codes and standards include the following references although any of the
3932 previously identified codes in Sections 3.5.1.2.ii(2) and 3.5.2.2.i(2) may
3933 be used.

- 3934
- 3935 • ASCE 7.
- 3936
- 3937 • International Building Code (IBC).
- 3938
- 3939 • AISC, "Specification for Structural Steel Buildings—Allowable
3940 Stress Design and Plastic Design."
- 3941
- 3942 • AISC, "Code of Standard Practice for Steel Buildings and
3943 Bridges."
- 3944
- 3945 • ASME B&PV Code, Section VIII.
- 3946
- 3947 • ACI 318.
- 3948

3949 ii. Structural Design Features

3950
3951 The reviewer should examine the adequacy of the applicant's descriptions of
3952 cask system components that are not important to safety but are subject to NRC
3953 approval. These descriptions should adequately identify the intended function(s)
3954 of each component.

3955
3956 Although the components evaluated in this portion of the DSS review are not
3957 directly important to safety, a credible possibility may exist that the structural
3958 response or failure of these components may cause a secondary risk to other
3959 components that *are* important to safety or to the subject nuclear material. For
3960 example, under tornado or seismic event conditions, the components may impact
3961 other components that are important to safety. When such a possibility exists,
3962 the applicant must provide more extensive structural information and greater
3963 assurance of acceptable fabrication and construction.

3964

3965 3.5.3.3 Materials Related to Structural Evaluation

3966
3967 The identification of structural materials should be reviewed in coordination with the materials
3968 discipline in Chapter 8 to the extent appropriate to determine if they are adequate for their
3969 intended function(s). The reviewer should determine the required level of review and extent of
3970 information in relation to the possibility and consequences of secondary effects on components
3971 that are important to safety. Materials should be as permitted or specified in the applicable
3972 code(s).

3973
3974 3.5.3.4 Structural Analysis

3975
3976 i. Load Conditions

3977
3978 The load definitions and combinations shown in Tables 3-2 and 3-3 have been
3979 accepted by the NRC for analysis of steel and reinforced concrete ISFSI
3980 structures that are important to safety. These load combinations may also be
3981 used for structures not important to safety.

3982
3983 In addition, for structures not important to safety, the NRC accepts the use of
3984 load combinations given in the IBC as well as ACI 349, ANSI/ANS 57.9, and
3985 ASCE 7.

3986
3987 The NRC also accepts the load descriptions, combinations, and analytical
3988 approaches given in the ASME B&PV Code, Section VIII, for pressure systems,
3989 vessels, and casks that do not form elements of the confinement cask.

3990
3991 ii. Structural Analysis Methods

3992
3993 The reviewer should evaluate the applicant’s selection and use of structural
3994 analysis methods, codes, and models and ensure that these are consistent with
3995 and appropriate for the design code applicable to the component (as discussed
3996 above).

3997
3998 iii. Structural Evaluation

3999
4000 The reviewer may determine that an NRC structural evaluation of certain other
4001 components is not necessary for approval of the cask system. Similarly, the
4002 NRC may determine that approval of the cask system does not need to include
4003 specific components that are not important to safety, even though the applicant
4004 seeks approval of those components as part of the application.

4005
4006 The SER should identify the system components that are excluded from the
4007 approval, stating the rationale for exclusion of each. As a corollary, the SER
4008 should also identify the components that are included, stating any limitations on
4009 the scope of the NRC review (e.g., “reviewed for functionality only”).

4010
4011 **3.6 Evaluation Findings**

4012
4013 The structural evaluation must provide reasonable assurance that the cask system will allow
4014 safe storage of SNF. This finding should be reached on the basis of a review that considered
4015 the regulation, appropriate RG, applicable codes and standards, and accepted engineering

4016 practices. Acceptance of the structural design of a storage cask system therefore implies that
4017 the design meets the relevant requirements of the following regulations:

4018
4019 F3.1 The SAR adequately describes all SSCs that are important to safety, providing
4020 drawings and text in sufficient detail to allow evaluation of their structural
4021 effectiveness.

4022
4023 F3.2 The applicant has met the requirements of 10 CFR Part 72.236(b). The SSCs
4024 important to safety are designed to accommodate the combined loads of normal
4025 or off-normal operating conditions and accidents or natural phenomena events
4026 with an adequate margin of safety. Stresses at various locations of the cask for
4027 various design loads are determined by analysis. Total stresses for the
4028 combined loads of normal, off-normal, accident, and natural phenomena events
4029 are acceptable and are found to be within limits of applicable codes, standards,
4030 and specifications.

4031
4032 F3.3 The applicant has met the requirements of 10 CFR Part 72.236(c), for
4033 maintaining subcritical conditions. The structural design and fabrication of the
4034 DSS includes structural margins of safety for those SSCs important to nuclear
4035 criticality safety. The applicant has demonstrated adequate structural safety for
4036 the handling, packaging, transfer, and storage under normal, off-normal, and
4037 accident conditions.

4038
4039 F3.4 The applicant has met the requirements of 10 CFR 72.236(l), "Specific
4040 Requirements for Spent Fuel Storage Cask Approval." The design analysis and
4041 submitted bases for evaluation acceptably demonstrate that the cask and other
4042 systems important to safety will reasonably maintain confinement of radioactive
4043 material under normal, off-normal, and credible accident conditions.

4044
4045 F3.5 The applicant has met the requirements of 10 CFR 72.236 with regard to
4046 inclusion of the following provisions in the structural design:

4047
4048 - Design, Fabrication, Erection, and Testing to Acceptable Quality
4049 Standards.

4050
4051 - Adequate Structural Protection Against Environmental Conditions
4052 and Natural Phenomena, Fires, and Explosions.

4053
4054 - Appropriate Inspection, Maintenance, and Testing.

4055
4056 - Adequate Accessibility in Emergencies.

4057
4058 - A Confinement Barrier that Acceptably Protects the Cladding
4059 During Storage.

4060
4061 - Structures that are Compatible with Appropriate Monitoring
4062 Systems.

4063
4064 - Structural Designs that are Compatible with Ready Retrievalability
4065 of SNF.
4066

4067 F3.6 The Applicant has met the specific requirements of 10 CFR 72.236(g) and (h) as
4068 they apply to the structural design for spent fuel storage cask approval. The cask
4069 system structural design acceptably provides for the following required
4070 provisions:

- 4071
- 4072 - Storage of the Spent Fuel for a Minimum of 20 Years.
- 4073
- 4074 - Compatibility with Wet or Dry Loading and Unloading Facilities.
- 4075

4076 The reviewer should provide a summary statement similar to the following:
4077

4078 “The staff concludes that the structural properties of the structures, systems, and
4079 components of the [cask designation] are in compliance with 10 CFR Part 72, and that
4080 the applicable design and acceptance criteria have been satisfied. The evaluation of the
4081 structural properties provides reasonable assurance that the [cask designation] will allow
4082 safe storage of SNF for a licensed (certified) life of ____ years. This finding is reached
4083 on the basis of a review that considered the regulation itself, appropriate regulatory
4084 guides, applicable codes and standards, and accepted engineering practices.”
4085

4086 **3.7 Designations and Descriptions of Loads**

4087
4088 Definitions of terms used in the following table are as accepted by the NRC. Many definitions
4089 are expanded with their intended applications more fully described and implemented than in the
4090 referenced sources.

4091
4092 Tables 3-2 and 3-3 do not apply to the analysis of confinement casks and other components
4093 designed in accordance with Section III of the ASME B&PV Code.

4094
4095 Capacities (“S” and “U” terms) and demands (factored or unfactored loads may be loads, forces,
4096 moments, or stresses caused by such loads. Usage must be consistent among the terms used
4097 in the load combination. Units of force, rather than mass, are to be used for loads.

4098
4099 Definitions of terms used in the load combination expressions for reinforced concrete and steel
4100 are derived from ANSI 57.9, ACI 349, AISC specifications, or another source. Where used in an
4101 expression related to steel analysis, definitions derived from ACI 349 are not limited in
4102 application to reinforced concrete analyses.

4103
4104 The load combinations defined on the basis of allowable stress apply to total stresses (that is,
4105 combined primary and secondary stresses). The load and stress factors do not change if
4106 secondary stresses are included.

Table 3-2 Loads and Their Descriptions

Symbol	Capacity or Load Term	Capacity or Load (or Demand) Description
S	Steel ASD strength	Strength of a steel section, member, or connection computed in accordance with the "allowable stress method" of the AISC "Specification for Structural Steel Buildings."
S_v	Steel ASD shear strength	Shear strength of a section, member, or connection computed in accordance with the "allowable stress method" of the AISC "Specification for Structural Steel Buildings."
U_s	Steel plastic strength	Strength (capacity) of a steel section, member, or connection computed in accordance with the "plastic strength method" of the AISC "Specification for Structural Steel Buildings."
U_c	reinforced concrete available strength	Minimum available strength (capacity) of reinforced concrete section, member, or embedment to meet the load combination, calculated in accordance with the requirements and assumptions of ACI 349 and, after application of the strength reduction factor, ϕ , as defined and prescribed at §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, D (instead of by subtraction from minimum available strength) reinforced concrete footing and foundation sections whose demand loads are dominated by the maximum soil reaction may be designed and evaluated using U_f .
U_f	Strength of foundation sections	Minimum available strength of reinforced concrete footing and foundation sections whose demand loads are dominated by the maximum soil reaction, and after the strength reduction factor, ϕ , as defined and prescribed at §9.3, "Design Strength," of ACI 349 is applied. Structural elements interface with columns, walls, grade beams, or footings and foundations should be evaluated by using load factors and load combinations for U_c . These interface elements include anchor bolts and other embedments, dowels, lugs, keys, and reinforcing extended into the footing or foundation.
U_g	Soil reaction or pile capacity	Minimum available soil reaction or pile capacity is determined by foundation analysis (expressed in a SAR for approval of a cask system as a required minimum for the cask system design). U_g is derived using the same load factors and load combinations as shown for determination of U_c .
O/S	Overturning/ sliding resistance	Required minimum available resistance capacity of structural unit against both overturning or sliding. Capacities for resistance of overturning and sliding are checked against the factored load combination separately, although the minimum margins of safety may occur concurrently. O/S is not determined by strength capacities of structural elements. Stress or strength demands resulting from an overturning or sliding situation are evaluated in load combinations involving S, S_v , U_s , U_c , and U_f .

Table 3-2 Loads and Their Descriptions

Symbol	Capacity or Load Term	Capacity or Load (or Demand) Description
	All loads used in combination	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 (factor) 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. Each load that may not always be present in the load combinations is to be varied from 0 to 100 percent to simulate the most adverse loading conditions (to the extent of proving that the lowest margins of safety have been determined).
D	Dead load	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 by conservative estimates. Loads resulting from 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. D includes the weight of soil vertically over a footing or foundation for the purposes of determining U_g , U_f , and O/S. Regardless of the load combination factor applied, D is to be varied by +5 percent if that produces the most adverse loading condition.
L	Live loads	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 associated with handling equipment and routine snow, rain, ice, and wind loads, and normal and off-normal impacts of equipment. Loads attributable to piping and equipment reactions are included. Depending on the case being analyzed, may include normal or off-normal events not separately identified, as may be caused by handling (not including drop), equipment or instrument malfunction, negligence, and other man-made or natural causes. Live loads attributable to casks with stored fuel need only be varied by credible increments of loading of an individual cask. Live loads attributable to multiple casks should be varied for the presence and positioning of one or more cask(s), as necessary and varied to determine the lowest margins of safety.

Table 3-2 Loads and Their Descriptions

Symbol	Capacity or Load Term	Capacity or Load (or Demand) Description
L	Live load for precast structures before final integration in-place	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. The NRC is only concerned with analysis of loading of reinforced concrete structures before use for cask system 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 obvious before installation, analysis of capacity versus pre-completion demands is not required.
DB	"Design-basis" (accident-level) loads	<p>Design-basis loads are controlling bounds for the following external event estimates:</p> <ul style="list-style-type: none"> (1) Extreme credible natural events to be used for deriving design bases that consider historical data or rated parameters, physical data, or analysis of upper limits of the physical processes involved. (2) Extreme credible external man-induced events used for deriving design bases on the basis of analysis of human activity in the region taking into account the site characteristics and associated risks. <p>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. Capacity to resist design basis loads can be assumed to be that of a structure that has not been degraded by previous design basis loads unless prior significant degradation in structural capacity may credibly occur and remain undetected.</p>
T	Thermal loads	Thermal loads, including loads associated with "normal" condition temperatures, temperature distributions, and thermal gradients within the structure; expansions and contractions of components; and restraints to expansions and contractions with the exception of 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 cask system. Thermal loads shall be determined for the most severe of both steady-state and accident 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, alternating casks in place).

Table 3-2 Loads and Their Descriptions

Symbol	Capacity or Load Term	Capacity or Load (or Demand) Description
T _a	Accident- level thermal loads	Thermal loads produced directly or as a result of <i>off-normal or design-basis</i> 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 the same 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	Loads attributable to 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-induced occurrences; or from severe natural phenomena not separately defined. Loads attributable to direct and secondary effects may be assumed to be nonconcurrent unless they might be additive. The capacity for resistance to the demand resulting from secondary effects would be that residual capacity following any degradation caused by the direct effect.
H	Lateral soil pressure	Loads caused by lateral soil pressure as would exist in normal, off-normal, or design-basis conditions corresponding to the load combination in which used. H includes lateral pressure resulting from ground water, the weight of the earth, and loads external to the structure transmitted to the structure by lateral earth pressure (not including earthquake loads, which are included in E, see below). H does not include soil reaction associated with attempted lateral movement of the structure or structural element in contact with the earth.
G	Loads attributable to soil reaction	Used only in load combinations for footing and foundation structural sections for which demand is limited by the soil reactions. G represents loads attributable to the maximum soil reaction (horizontal (passive pressure limit) and vertical (soil or pile bearing limit) that would exist in normal, off-normal, or design-basis conditions corresponding to the load combination used. G is a function of U _g (i.e., G = f (U _g)).
W	Wind loads	Wind loads produced by normal and off-normal maximum winds. Pressure resulting from 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.

Table 3-2 Loads and Their Descriptions

Symbol	Capacity or Load Term	Capacity or Load (or Demand) Description
W_t	Tornado loads	Loads attributable to 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 resulting from 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	Loads attributable to the direct and secondary effects of the design earthquake 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.

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3.7.1 Load Combinations for Steel and Reinforced Concrete Non-Confinement Structures

The reinforced concrete structure load combinations apply to reinforced concrete structures important to safety that are not within the scope of ACI 359 (ASME B&PV Code, Section III, Division 2). The load combinations apply to steel structures important to safety that are not within the scope of the ASME B&PV Code, Section III, Division 1. The NRC accepts, but does not require use of these load combinations for steel and reinforced concrete structures that are not important to safety. The NRC accepts steel analyses that reflect allowable stress design or plastic strength design. Steel load combinations may be determined on the basis of the set of load combination expressions involving either “S” or “U_s.”

Table 3-3 Load Combinations for Steel and Reinforced Concrete Non-Confinement Structures

Load Combination	Acceptance Criteria
<i>Reinforced Concrete Structures — Normal Events and Conditions</i>	
$U_c > 1.4 D + 1.7 L$	Capacity/demand >1.00 for all sections.
$U_c > 1.4 D + 1.7 (L + H)$	Capacity/demand >1.00 for all sections.
<i>Reinforced Concrete Structures — Off-Normal Events and Conditions</i>	
$U_c > 1.05 D + 1.275 (L + H + T)$	Capacity/demand >1.00 for all sections.
$U_c > 1.05 D + 1.275 (L + H + T + W)$	Capacity/demand >1.00 for all sections.
<i>Reinforced Concrete Structures — Accidents and Conditions</i>	
$U_c > D + L + H + T + (E \text{ or } F)$	Capacity/demand >1.00 for all sections.

Table 3-3 Load Combinations for Steel and Reinforced Concrete Non-Confinement Structures

Load Combination	Acceptance Criteria
$U_c > D + L + H + T + A$	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.
$U_c > D + L + H + T_a$	Capacity/demand >1.00 for all sections.
$U_c > D + L + H + T + W_t$	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.
<i>Reinforced Concrete Footings/Foundations — Normal Events and Conditions</i>	
$U_f > D + (L + G)$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
$U_f > D + (L + H + G)$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
<i>Reinforced Concrete Footings/Foundations — Off-Normal Events and Conditions</i>	
$U_f > D + (L + H + T + G)$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
$U_f > D + (L + H + T + W + G)$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
<i>Reinforced Concrete Footings/Foundations — Accident-Level Events and Conditions</i>	
$U_f > D + L + H + T + E + G$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
$U_f > D + L + H + T + A + G$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
$U_f > D + L + H + T_a + G$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
$U_f > D + L + H + T + W_t + G$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
$U_f > D + L + H + T + F + G$	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
<i>Steel Structures Allowable Stress Design — Normal Events and Conditions</i>	
$(S \text{ and } S_v) > D + L$	Factored strength/demand >1.00 for all sections.
$(S \text{ and } S_v) > D + L + H$	Factored strength /demand >1.00 for all sections.

Table 3-3 Load Combinations for Steel and Reinforced Concrete Non-Confinement Structures

Load Combination	Acceptance Criteria
<i>Steel Structures Allowable Stress Design — Off-Normal Events and Conditions</i>	
1.3 $(S \text{ and } S_v) > D + L + H + W$	Factored strength /demand >1.00 for all sections.
1.5 $S > D + L + H + T + W$	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.4 $S_v > D + L + H + T + W$	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
<i>Steel Structures Allowable Stress Design — Accidents and Conditions</i>	
1.6 $S > D + L + H + T + (E \text{ or } W_t \text{ or } F)$	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.4 $S_v > D + L + H + T + (E \text{ or } W_t \text{ or } F)$	Factored strength (allowable stress design)/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.7 $S > D + L + H + T + A$	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.4 $S_v > D + L + H + T + A$	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.7 $S > D + L + H + T_a$	Factored strength/demand >1.00 for all sections.
1.4 $S_v > D + L + H + T_a$	Factored strength/demand >1.00 for all sections.
<i>Steel Structures Plastic Strength Design — Normal Events and Conditions</i>	
$U_s > 1.7 (D + L)$	Plastic capacity/demand >1.00 for all sections.
$U_s > 1.7 (D + L + H)$	Plastic capacity/demand >1.00 for all sections.
<i>Steel Structures Plastic Strength Design — Off-Normal Events and Conditions</i>	
$U_s > 1.3 (D + L + H + W)$	Plastic capacity/demand >1.00 for all sections.
$U_s > 1.3 (D + L + H + T + W)$	Plastic capacity/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.

Table 3-3 Load Combinations for Steel and Reinforced Concrete Non-Confinement Structures

Load Combination	Acceptance Criteria
<i>Steel Structures Plastic Strength Design — Accidents and Conditions</i>	
$U_s > 1.1 (D + L + H + T + (E \text{ or } W_t \text{ or } F))$	Plastic capacity/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile. 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.
$U_s > 1.1 (D + L + H + T + A)$	Plastic 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. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
$U_s > 1.1 (D + L + H + T_a)$	Plastic capacity/demand >1.00 for all sections.
<i>Overturning and Sliding — Normal and Off-Normal Events and Conditions</i>	
$O/S \geq 1.5 (D + H)$	Capacity/demand ≥ 1.00 for structure to be satisfied for both overturning and sliding.
<i>Overturning and Sliding — Accidents and Conditions</i>	
$O/S \geq 1.1 (D + H + E)$	Capacity/demand ≥ 1.00 for structure to be satisfied for both overturning and sliding.
$O/S \geq 1.1 (D + H + W_t)$	Capacity/demand ≥ 1.00 for structure to be satisfied for both overturning and sliding.

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APPENDIX 3A - COMPUTATIONAL MODELING SOFTWARE

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Technical Review Guidance:

Computational Modeling Software (CMS) Application

The staff does not endorse the use of any specific type or code vendor of CMS. Any appropriate CMS application could be used for analyses of cask or package components; however, for any CMS to demonstrate that a particular cask design satisfies regulatory requirements, adequate validation of that CMS must be demonstrated by the applicant. Descriptions of CMS validations can be contained within a given application or incorporated by reference.

The reviewer should verify that the following information is provided in the SAR or related documentation (such as proprietary calculation packages or benchmark reports):

- (1) details of the methodology used to assemble the computational models and the theoretical basis of the program used;
- (2) a description of benchmarking against other codes or validation of the CMS against applicable published data or other technically qualified and relevant data that is appropriately documented;
- (3) standardized verification problems analyzed using the CMS, including comparison of theoretically predicted results with the results of the CMS; and
- (4) release version and applicable platforms.

Once the information described above has been docketed, it need not be submitted with each subsequent application, but can be referred to in subsequent SARs or related documents. If an applicant changes their analysis methodology or changes the type or vendor of the CMS used, the applicant should submit either a revision of previously submitted information or include a clear explanation of the methodology changes, and their effects on the analysis in question, in subsequent SAR submittals.

Modeling Techniques and Practices

Modeling techniques and practices used by applicants may need to be verified to demonstrate adequacy of the model.

- The reviewer should verify that the CMS and the options used by the applicant are appropriate for adequately capturing the behavior of a cask, package, or any components.

Relevant input and results files or an equivalent detailed model description and output should be submitted with the original application.

- Analysis input files should be submitted in an electronic format that would most easily allow the solution to be executed by the staff, should the staff desire to do so. In-depth review of CMS models is most easily done with input files that contain individual commands used to develop the model and apply the various

4173 boundary conditions; therefore, a text input file format (versus database format)
4174 is preferred.

- 4175 • Input files should be annotated in a way that clearly demonstrates the process
4176 behind building and solving models developed using CMS. A well annotated
4177 input file will expedite staff review and preclude the need for further clarification
4178 questions by the staff.
- 4179 • Appropriate electronic media should be used for submitting case and support
4180 files. It should be noted that electronic media should be delivered to the
4181 appropriate SFST staff directly, if possible, as electronic media sent to the NRC
4182 Document Control Desk may be damaged during security screening.

4185 Computer Model Development

4186 The reviewer should verify that the computer model used for the analysis is adequately
4187 described, either in the SAR or in other documentation, is geometrically representative of the
4188 cask design being analyzed, has addressed how material and manufacturing uncertainties
4189 might affect the analysis, has appropriate boundary conditions, and has no significant analysis
4190 errors.

- 4191 • The reviewer should verify that the model description includes an adequate basis
4192 for the selection of parameters and/or components used in the analysis model
4193 (e.g., why was a particular element type applied in the analysis model?)
- 4194 • The reviewer should verify that models sufficiently represent cask or package
4195 geometry and that adequate justification is provided for simplifications used.
4196 Models created with CMS are often simplified to reduce computer processing
4197 time. Models can often omit geometric details or use homogenized or smeared
4198 material properties to represent complex geometry or material combinations and
4199 still retain analytic accuracy.
- 4200 • The reviewer should verify that the applicant has discussed how manufacturing
4201 and/or assembly tolerances and contact resistances will affect the analyses that
4202 have been conducted, if at all, in both the structural and thermal disciplines. The
4203 reviewer should also verify that the applicant has described how tolerances
4204 and/or contact resistances are accounted for, if applicable, in the cask or
4205 package analysis models that are submitted for review.
- 4206 • The reviewer should verify that the applicant has provided a general discussion
4207 of how error, warning, or advisory messages generated by the software affect the
4208 analysis result (if applicable). When processing a computer model developed
4209 using CMS, the software will frequently provide error, warning, or advisory
4210 messages indicating a possible problem with the model that may or may not be
4211 sufficient to terminate processing. If the error/warning function has been
4212 disabled during processing, an explanation of why this is appropriate should be
4213 provided.
- 4214 • The reviewer should verify that, within the specific disciplines, the dimensions
4215 and physical units used in the models developed are clearly labeled and mutually
4216 consistent. The fundamental units of time, mass, and length should be clearly
4217

4224 identified. All other physical units derived must be consistent with the basic units
4225 adopted. For example, if the unit of length is the millimeter (mm), time in
4226 milliseconds (ms), and mass in gram (g), then, the mechanical force will have
4227 units of Newton (N), energy in milliJoule (mJ), and stress in megapascal (MPa).
4228 Verify that the input parameters are expressed in the units as assigned. If an
4229 applicant chooses not to adopt this uniformity of units, the appropriate conversion
4230 must be applied prior to processing input into CMS. Similar assurances must be
4231 provided for the output for the analysis solution.

4232 Computer Model Validation

- 4233 • The reviewer should verify that model validation done with applicable
4234 experiments or testing is properly documented and appropriate references are
4235 provided.
- 4236 • The reviewer should ensure that if the applicant takes credit for modeling
4237 conservatism, those conservatisms have been demonstrated through validation
4238 of the model or analysis methodology. For example, accounting for certain
4239 conditions that occur during the hypothetical accident condition (HAC) fire, such
4240 as combustion of materials, the turbulent flow of hot gasses in the pool fire
4241 environment, and material anomalies that may manifest themselves in a fire can
4242 be done with specialized CMS codes (specifically, coupled CFD-FEA codes such
4243 as Sandia National Lab's CAFÉ code), high performance computer hardware and
4244 extended compute times. Each of these conditions can be treated in a
4245 conservative fashion using standard CMS; however, validation of the CMS
4246 against actual data (such as open pool fire test data or material combustion
4247 data), to demonstrate the applicability of the CMS under the HAC fire, for a
4248 configuration similar to that which is being modeled, would be necessary.

4252 Justification of Bounding Conditions/Scenario for Model Analysis

4253 The applicant must determine the most damaging orientation and worst-case conditions for a
4254 given design and document how the analytic model was configured for the scenario.

4255 The reviewer should verify that the applicant provided sufficient justification for selecting the
4256 most damaging orientation and worst-case conditions.

4257 Description of Boundary Conditions and Assumptions

- 4258 • The reviewer should verify, as necessary, that boundary conditions and
4259 assumptions are addressed in the textual description included in the SAR or
4260 other documents (e.g., emissivity values, absorptivity values, convective
4261 coefficients, radiation view factors, symmetry planes, and rigid surfaces). This
4262 information should be presented in either tabular form or in a complete textual
4263 manner. Justifications and bases for such items should also be included in the
4264 textual description.
- 4265 • Values or quantities indicating performance enhancements, i.e., increasing
4266 material conductivity values to mimic internal convection or substantially reduced
4267 design load factors (DLFs) reflecting an unusually high degree of impact

4274 damping, should be accompanied with justifications and should be closely
4275 reviewed and independently verified, if needed, by staff.

4276
4277 Documentation of Material Properties

4278
4279 As needed, the reviewer should assess that:

- 4280
- 4281 (1) units for material properties are consistent throughout the individual SAR
4282 chapters.
 - 4283
 - 4284 (2) material properties for all applicable temperature ranges are included.
 - 4285
 - 4286 (3) references to materials used by the CMS application and specific material
4287 properties based on geometry (e.g., conductivity in the X, Y and Z directions), are
4288 listed in the SAR or related documents.
 - 4289

4290 Description of Model Assembly

- 4291
- 4292 • The reviewer should verify that the types of elements used in the model are listed
4293 in the SAR, preferably in tabular format, along with the corresponding materials
4294 or components in which they are used in the analysis model. (i.e., the reviewer
4295 should quickly be able to discern what elements and materials are associated
4296 with specific components of the analysis model.)
 - 4297
 - 4298 • The reviewer should verify that a sufficient explanation of the logic behind the
4299 creation of each specific computer model is provided, for effective confirmatory
4300 calculations to be performed.
 - 4301
 - 4302 • The reviewer should verify that the applicant has provided annotated input files
4303 (as appendices to the SAR or in related documents), that clearly outline the
4304 various steps in building the computer models submitted. If input files are not
4305 provided or do not adequately describe model assembly, the applicant should
4306 provide an adequate explanation of how computer models were assembled using
4307 the CMS in the appropriate SAR chapters or related documents.
 - 4308

4309 Loads and Time Steps

- 4310
- 4311 • The reviewer should verify that loads, load combinations, and, if used by the
4312 analytical code, the load steps utilized in the computer model are clearly
4313 explained by the applicant. The staff should evaluate all loads, how they are
4314 placed on the computer models, load combinations, and if used, the time steps
4315 applied in the analysis.
 - 4316
 - 4317 • The reviewer should verify that the time steps specified for the solution of the
4318 analysis are sufficiently small to accurately capture the behavior of the structures,
4319 systems, or components being modeled.
 - 4320
 - 4321 • The reviewer should verify that incremental time steps (or sub-steps) are
4322 adequately converged. Information of convergence may be obtained from the
4323 output generated by the execution of the analysis solution.
 - 4324

4325 Sensitivity Studies

4326

4327 The discussion of sensitivity studies should be included in the general Computer Model
4328 Development discussion, as noted above, with relevant references to examples included in the
4329 SAR or related documents.

4330

4331 • The reviewer should verify that the applicant has completed sensitivity studies for
4332 relevant CMS modeling parameters. This includes mesh type and density, load
4333 step size, interfacing gaps or contact friction, material models and model
4334 parameters selection, and property interpolation, if applicable. For example, a
4335 mesh sensitivity study should be conducted not only for mesh density but also for
4336 mesh density/refinement in areas of thermal or structural concern or where
4337 performance of the material is crucial, such as seal areas, lid bolts, etc.

4338

4339 • The reviewer should verify that the results of applicable sensitivity studies are
4340 clearly described in the SAR or related documentation and can be independently
4341 verified, if necessary.

4342

4343 • The reviewer should verify that the applicant's documentation includes at least a
4344 brief discussion of the different models used in their mesh sensitivity studies.

4345

4346 Results of the Analysis

4347

4348 • The reviewer should verify that the SAR, or related document(s), include all
4349 relevant results (tabular and computer plots) for applicable load cases and load
4350 combinations evaluated for design code compliance, and that all governing
4351 results (stresses/deformation) are clearly identified in the tables and on plots.

4352

4353 • The reviewer should verify that results are consistent throughout the SAR, and
4354 that the correct results are used in calculations of other cask or package
4355 performance parameters (e.g., calculated temperatures used in the internal
4356 pressure calculation should be verified).

4357

4 THERMAL EVALUATION

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4.1 Review Objective

The thermal review ensures that the cask and fuel material temperatures of the dry storage system (DSS) will remain within the allowable values or criteria for normal, off-normal, and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation that could lead to gross rupture. Also confirmed is the use by the applicant of acceptable analytical and/or testing methods in the Safety Analysis Report (SAR) when evaluating the DSS thermal design.

4.2 Areas of Review

As defined in Section 4.5, "Review Procedures," a comprehensive thermal evaluation should encompass the following areas of review:

Decay Heat Removal System

Material and Design Limits

Thermal Loads and Environmental Conditions

Analytical Methods, Models, and Calculations

- Configuration
- Material Properties
- Boundary Conditions
- Computer Codes
- Temperature Calculations
- Pressure Analysis
- Confirmatory Analysis

4.3 Regulatory Requirements

This section presents a summary matrix of the portions of the U.S. Code of Federal Regulations (CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Greater Than Class C Waste," Title 10, "Energy" (10 CFR Part 72) that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should be familiar with the regulatory language in these sections. Table 4-1 matches the relevant regulatory requirements associated with this chapter to the areas of review.

Table 4-1 Relationship of Regulations and Areas of Review		
Area of Review	10 CFR Part 72 Regulations	
	72.122 (h)(1), (l)	72.236 (b), (f), (g), (h)
Decay Heat Removal Systems	•	•
Material and Design Limits		•
Thermal Loads and Environmental Conditions	•	•
Analytical Methods, Models, and Calculations	•	•

4401

4.4 Acceptance Criteria

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4403

4404

4.4.1 Decay Heat Removal System

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The applicant must provide a detailed description of the proposed cask heat removal system and its passive cooling characteristics. All major components are to be clearly identified and their contribution to heat-removal from the fuel thoroughly explained. The mechanism of heat removal (i.e., conduction, convection, radiation) for each component should also be discussed.

4410

4411

4412

Evidence must be provided by the applicant that the decay heat removal system will operate reliably under normal, loading, off-normal, and accident-level conditions.

4413

4414

All instrumentation used to monitor cask thermal performance should also be described.

4415

4416

4.4.2 Material and Design Limits

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4418

4419

4420

Cask components and fuel materials should be maintained between their minimum and maximum temperature limits for normal, loading, off-normal, and accident-level conditions to enable all components to perform their intended safety function.

4421

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To guarantee cladding integrity of zirconium-based alloys, the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations, including cask drying and backfilling. A higher temperature limit may ONLY be used for low burnup spent nuclear fuel (SNF) (less than 45 GWd/MTU), as long as the applicant can demonstrate that the best estimate cladding hoop stress is equal to or less than 90 MPa (13.1 ksi) for the temperature limit that is proposed. During loading operations, repeated thermal cycling should be limited to less than 10 cycles, with cladding temperature variations more than 65°C (149°F). For off-normal and accident conditions, the maximum zirconium based cladding temperature should not exceed 570°C (1058°F).

4431

4432

4433

To guarantee stainless steel cladding integrity, the maximum calculated fuel cladding temperature should not exceed 570°C (1058°F) for off-normal and accident conditions and the

4434 maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal
4435 conditions of storage and short-term loading operations, including cask drying and backfilling.

4436
4437 The applicant must clearly identify the operational temperature limits for all important-to-safety
4438 component materials under normal, loading, unloading, off-normal and accident-level
4439 conditions. The applicant shall provide reliable basis for all the temperature limits.

4440
4441 The maximum internal pressure of the fuel container should remain within its design pressures
4442 for normal, off-normal, and accident-level conditions assuming rupture of 1 percent, 10 percent,
4443 and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include
4444 release of 100 percent of the initial fill gas and 30 percent of the fission product gases
4445 generated within the fuel rods during operation.

4446
4447 The applicant must clearly identify the design pressure limits for the fuel container under normal,
4448 off-normal and accident-level conditions.

4449

4450 **4.4.3 Thermal Loads and Environmental Conditions**

4451
4452 Identification and justification of the design basis thermal load must be made by the applicant as
4453 well as the insulation and ambient temperature assumptions used as boundary conditions for
4454 the normal, loading, off-normal, and accident scenarios.

4455

4456 **4.4.4 Analytical Methods, Models, and Calculations**

4457
4458 The applicant shall present a thermal analysis that clearly demonstrates the storage system's
4459 ability to manage design heat loads and have the various materials and components remain
4460 within temperature limits. The analysis shall be conducted for normal, loading,
4461 draindown/reflood, off-normal, and accident-level conditions. Resulting temperature profile and
4462 internal pressure information are necessary to support the structural analysis (Chapter 3) and
4463 the confinement analysis (Chapter 5) of the SAR.

4464
4465 The applicant shall specify the analytical methods used in the thermal evaluations including any
4466 computational modeling software, (i.e., heat transfer or computational fluid dynamics computer
4467 analysis codes) and shall discuss the basis for the parameters and options selected for the
4468 analysis. All models should be clearly described. Material thermal properties for all cask
4469 components shall be provided and justified. The applicant must address, quantify, and report
4470 the degree of conservatism associated with the proposed models and the resulting safety
4471 margins.

4472
4473 The computer codes used in the thermal evaluation should be well-verified and validated. The
4474 applicant must provide acceptable basis (e.g., benchmark efforts, published results) for the
4475 accuracy of the chosen computer code(s) and justification for its use in the proposed evaluation.
4476 A discussion of the resulting level of convergence and conservatism achieved as a function of
4477 the modeling options (e.g., meshing, time-differencing) must be provided by the applicant.

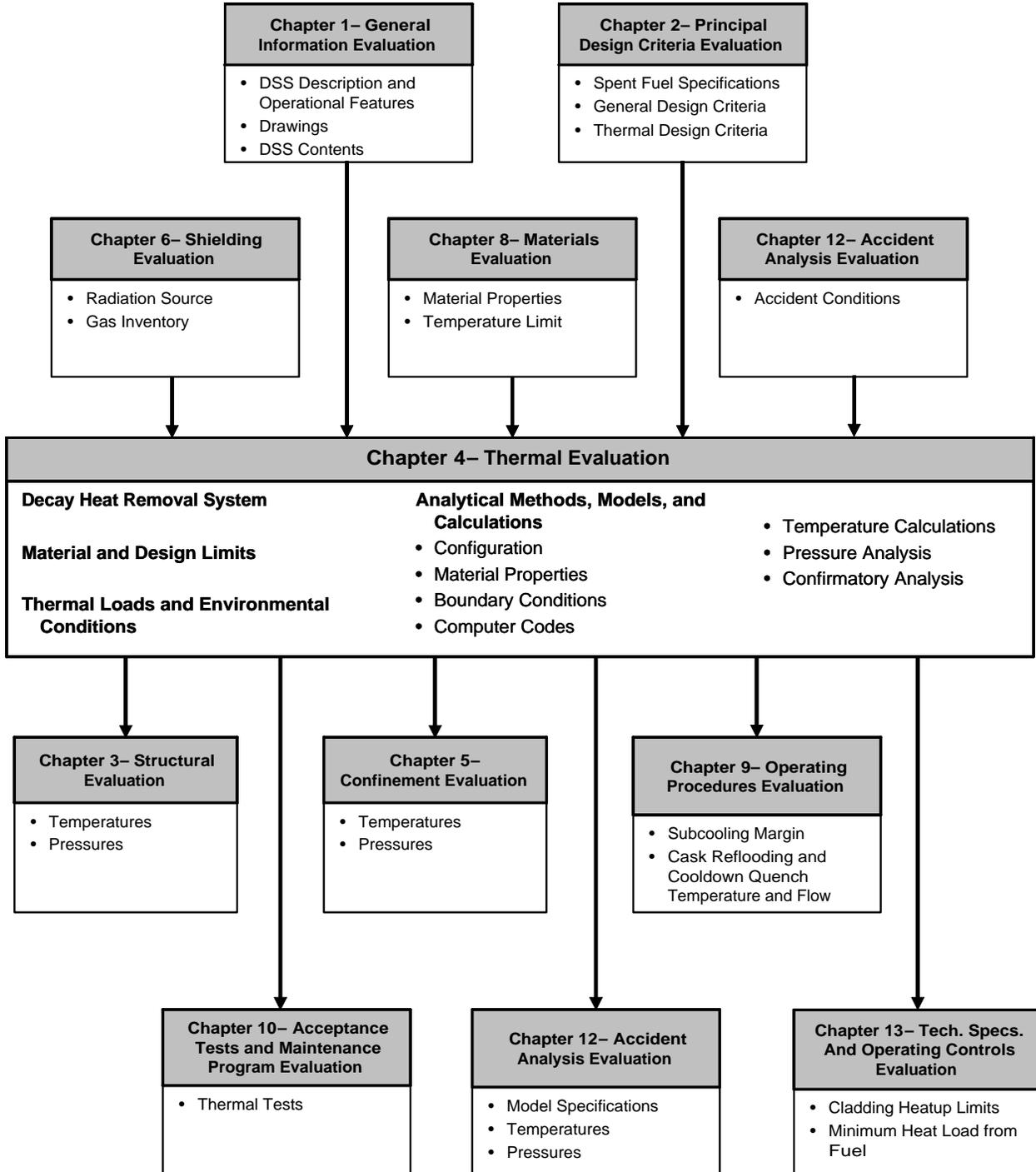
4478
4479 To facilitate confirmatory analyses, electronic copies of the most significant input and output
4480 files should be provided. Further guidance on the review of analytical methods, models, and
4481 calculations provided to the staff for review is provided in Appendix 3A, "Computational
4482 Modeling Software."

4483

4484 **4.5 Review Procedures**

4485

4486 Figure 4-1 presents an overview of the evaluation process and can be used as a guide to assist
 4487 in coordinating with other review disciplines.



4488

4489

Figure 4-1 Overview of the Thermal Evaluation

4490 Design features and acceptance criteria, initially presented in SAR Chapter 1, "General
4491 Information," and Chapter 2, "Principal Design Criteria," should be reviewed for additional insight
4492 about the thermal models that are being presented. Reviewers should examine the
4493 appropriateness of the proposed heat loads and environmental conditions. Modeling details
4494 such as simulation options, simplifications, and accuracy of results should be assessed. The
4495 DSS is to be analyzed under normal, loading, off-normal, and accident scenarios. If necessary,
4496 the resulting temperature distributions and internal pressures calculated in the SAR should be
4497 confirmed in order to verify compliance with design criteria and regulatory requirements.

4498
4499 One of the most important results of the DSS thermal evaluation is confirmation that the fuel
4500 cladding temperature will remain below a specified limit to prevent unacceptable degradation
4501 during storage.

4502
4503 Thermal performance of the cask under accident conditions is also evaluated in accordance
4504 with Chapter 12, "Accident Analyses Evaluation," of this SRP, as appropriate, in the overall
4505 accident analyses presented in the SAR.

4506
4507 In conducting a comprehensive thermal evaluation, reviewers should perform the established
4508 review procedures, as applicable, for each of the following areas of review.

4509
4510 **4.5.1 Decay Heat Removal System (HIGH Priority)**

4511
4512 The reviewer should examine the description of the DSS presented in SAR Chapter 1, "General
4513 Information Evaluation" as supplemented by the additional information provided in SAR Chapter
4514 4, "Thermal Evaluation." These two sources of information should be consistent and
4515 supplementary. In addition to the material compositions, the dimensions of the cask
4516 components and SNF assemblies are to be clearly indicated. All drawings, figures, and tables
4517 should be sufficiently detailed to support in-depth staff evaluation.

4518
4519 The applicant's analysis should include the description of the significant thermal design features
4520 and operating characteristics of all pertinent DSS components and subsystems. Design
4521 features typically include the cask body, thermal fins, shielding materials, fuel baskets, heat
4522 transfer disks, containment seals, drain and vent ports, and external pressure relief devices for
4523 the case of transfer casks, among others. The reviewer should verify that the thermal design
4524 features will adequately perform their intended safety functions during normal, loading, off-
4525 normal, and accident-level conditions. All thermal design features should be passive.
4526 Applicants have requested temporary external forced cooling of cask systems during loading
4527 operations or as a Technical Specification action statement during transfer operations. Such
4528 requests need to be examined by the staff to ensure that they meet the original intent of the
4529 regulations; that cask systems remain passively cooled during normal operations.

4530
4531 Any instrumentation used to monitor cask thermal performance should also be described by the
4532 applicant in sufficient detail to support in-depth staff evaluation. The monitoring instrumentation
4533 components should have a safety classification (presented in SAR Chapter 2, "Principle Design
4534 Criteria Evaluation") commensurate with their function and should be fully justified. Applicable
4535 operating controls and criteria, such as temperature criteria and surveillance requirements,
4536 should be clearly indicated in SAR Chapter 13, "Technical Specifications and Operational
4537 Controls and Limits" discussed in the Safety Evaluation Report (SER), and included in the
4538 Certificate of Compliance (CoC), as appropriate.

4539

4540 **4.5.2 Material and Design Limits (Priority - as indicated)**

4541
4542 (MEDIUM Priority) One of the most important results of the thermal evaluation is the
4543 confirmation that the fuel cladding temperature is sufficiently low to prevent cladding damage or
4544 potential failure during storage. Section 4.4.2, "Material and Design Limits," of this SRP
4545 identifies the criteria for cladding temperature limits. The application must clearly agree with
4546 these criteria.

4547
4548 (MEDIUM Priority) During licensing reviews, the thermal reviewer should ensure that either of
4549 the following criteria are used: (1) the maximum calculated temperatures for normal conditions
4550 of storage and for fuel loading operations do not exceed 400°C (752°F), or (2) the maximum
4551 calculated temperatures for normal conditions of storage do not exceed 400°C (752°F) and that
4552 the materials reviewer has verified that the best estimate cladding hoop stress is less than 90
4553 MPa (13.1 ksi) for the maximum allowable temperature specified by the applicant for short-term
4554 fuel loading. If the applicants use the latter approach, the thermal reviewer will verify that the
4555 materials reviewer has verified that the cladding hoop stresses are less than 90 MPa (13.1 ksi)
4556 for each fuel assembly type (e.g., 14x14, 17x17, 9x9, etc.) proposed for storage. Cladding
4557 oxide thickness used to compute hoop stress should be evaluated by the materials reviewer.
4558 Since the hoop stress is dependent on the rod internal pressure, cladding geometry, and the
4559 temperature of the gases inside the rod, the staff will verify that the applicant has calculated the
4560 best estimate hoop stress corresponding to the rod internal pressure of the highest burnup fuel
4561 assemblies of the specific type of assembly.

4562
4563 (MEDIUM Priority) To limit the amount of SNF that could be released from the cladding under
4564 off-normal conditions or accidents, the maximum calculated cladding temperatures should be
4565 maintained below 570°C (1058°F).

4566
4567 (MEDIUM - bolted closure/LOW - welded closure) The reviewer should verify that temperature
4568 restrictions (upper and lower allowable limits) on all components important to safety (e.g.,
4569 confinement, shielding, subcriticality, heat removal) during normal, loading, off-normal, and
4570 accident scenarios are clearly identified in the application and that the predicted thermal
4571 behavior of the entire DSS is indeed within the specified limits. The thermal reviewer should
4572 confirm with the assigned materials reviewer the acceptability of all proposed temperature limits.

4573
4574 (LOW Priority) The maximum internal pressure of the fuel container should remain within its
4575 design limits for normal, off-normal, and accident-level conditions assuming rupture of 1
4576 percent, 10 percent, and 100 percent of the fuel rods, respectively. The thermal reviewer
4577 should confirm with the assigned structural reviewer the acceptability of the proposed design
4578 pressure limits.

4579
4580 (HIGH Priority) Any operating scenario (loading or unloading) that results on a time-dependent
4581 limiting condition (e.g., number of hours allowed for vacuum drying before fuel cladding
4582 temperature reaches its allowable limit) should also be addressed in Chapter 13, "Technical
4583 Specifications and Operating Controls and Limits Evaluation," of the SRP and should be
4584 included as a limiting condition for operation (e.g., technical specification) in the CoC, as
4585 appropriate.

4586

4587 **4.5.3 Thermal Loads and Environmental Conditions (Priority - as indicated)**

4588

4589 (LOW Priority) The reviewer should examine the specification for the design-basis fuel decay
4590 heat presented in SAR Chapter 2, "Principle Design Criteria Evaluation" and ensure that this
4591 decay heat is consistent with the specified fuel types, burnups, enrichments and cooling times, if
4592 included. Some applications, however, may provide a bounding decay heat load (kW/assembly)
4593 without specifying details about the SNF (design, enrichment, cooling time).

4594

4595 (LOW Priority) The axial distribution for the decay heat sources should also be discussed by the
4596 applicant with clear justification for a bounding approach. The reviewer should expect a
4597 somewhat flat-at-the center axial distribution with a peak-to-average value in the range of 1.1 to
4598 1.2, tapering towards both ends.

4599

4600 (MEDIUM Priority) In general, the NRC staff accepts insolation values presented in 10 CFR Part
4601 71 for 10 CFR Part 72 applications. Because of the large thermal inertia of a storage cask, the
4602 insolation values listed in 10 CFR Part 71.71 may be averaged over a 24-hour day assuming
4603 steady-state conditions.

4604

4605 (MEDIUM Priority) The reviewer should verify that the ambient temperatures used for normal
4606 and off-normal condition evaluations do indeed bound the available historical temperature data
4607 for any suggested storage site (current or future). The National Oceanic Atmospheric
4608 Administration (NOAA) National Climatic Data Center provides temperature statistics for many
4609 American cities and regions. (<http://www.ncdc.noaa.gov/oa/ncdc.html>).

4610

4611 (MEDIUM Priority) Loading and unloading evaluations should be established on the basis of the
4612 SNF pool's technical specification maximum temperature limit (typically 46°C (115°F)).

4613

4614 **4.5.4 Analytical Methods, Models, and Calculations (MEDIUM Priority)**

4615

4616 For cask system components in which material properties and performance vary with
4617 temperature, the reviewer should examine the assumptions used in determining temperature
4618 maxima, minima, gradients, and differences for the cask system, as well as review the
4619 assumptions used to determine fuel cladding temperatures. The assumed temperature
4620 changes over time should result in the bounding conditions for the structural analysis. The
4621 calculated temperatures in the various cask system components should be compared to the
4622 limiting temperature criteria for the appropriate materials. Ferritic materials are subject to failure
4623 by brittle fracture at low temperatures. The reviewer should verify the assumed low
4624 temperatures for cask system handling operations for consistency with material properties.
4625 Ambient temperature restrictions may be appropriate for cask handling operations. Any limiting
4626 conditions regarding ambient temperatures should be addressed in SAR Chapter 13, as well as
4627 SER Chapter 13, "Technical Specifications and Operating Controls and Limits Evaluation," and
4628 should be included as a limiting condition for operation (e.g., technical specification) in the CoC,
4629 as appropriate.

4630

4631 Analysis for accident-level ("design-basis") temperatures should not be considered to envelop
4632 the analysis of normal or off-normal temperatures. The acceptance criteria for normal and off-
4633 normal temperature demands for structural capacity will differ. Therefore, all three conditions
4634 should be analyzed. In addition, the duration over which accident temperature conditions may
4635 exist should be evaluated.

4636

4637 4.5.4.1 Configuration (HIGH Priority)

4638
4639 The reviewer should verify that any model used in the thermal evaluation is clearly described.
4640 Separate models and submodels may be used for the evaluation of different conditions (normal
4641 storage, loading, off-normal situations, and accidents). Coordination with the structural review is
4642 necessary to evaluate any damage that may result from accidents or natural phenomena
4643 events. All models should be shown as conservative.

4644
4645 Examination by the reviewer of the sketches or figures of all models ensures their proper use in
4646 the thermal calculations and verifies that the dimensions and materials are consistent with those
4647 in the drawings of the actual cask, as presented in SAR Chapter 1, "General Information
4648 Evaluation". If possible, the reviewer should examine the computer input files to verify
4649 consistency with the model sketches and engineering drawings. Differences between the actual
4650 cask configuration and the model should be identified, and the model should be shown to be
4651 conservative.

4652
4653 Particular attention during the review should be paid to gaps between cask components.
4654 Tolerances should be considered so that the thermal resistance of each gap is treated
4655 conservatively. Gases (e.g., air, helium) assumed to be present in the gap shall be described
4656 and justified. If a specific gas other than air in the cask cavity or gaps between cask
4657 components is relied upon for heat removal, the reviewer should verify that the applicant shows
4658 that the gas is retained *and* that the gas is not diluted by other gases having lower thermal
4659 conductivities during the entire storage period. For cask components that are important to heat
4660 removal, manufacturing techniques for joining components, surface roughness, contact
4661 pressures, and gap conductance values should be adequately described and justified.

4662
4663 The reviewer should verify that decay heat generated in the SNF is limited to the active fuel
4664 region of the assemblies. The model should specifically account for the peaking in the central
4665 region or provide another conservative approach. Heat from any other stored component (e.g.,
4666 control rods), if applicable, should also be distributed appropriately. In addition, the positions of
4667 heat sources relative to other cask components should be identified.

4668
4669 The application should address the thermal interaction among casks in an array by using a view
4670 factor less than unity. Generally, this will result in an operating control and limit in SAR Chapter
4671 13 that imposes a minimum spacing between storage casks.

4672
4673 Coordination with the structural reviewer is necessary to ensure that the applicant has analyzed
4674 situations that may produce the worst-case cask loads. The greatest gradients and loadings
4675 caused by thermal expansion may occur with casks in alternative storage or in temporary
4676 handling positions.

4677
4678 The heat transfer processes used in the analysis should be examined. Conduction and
4679 radiation are typically defined as the primary heat transfer mechanisms within the cask itself.
4680 Convection by natural circulation should be limited to that between the external surface of the
4681 cask and the ambient environment. In narrow regions of any orientation, little or no convective
4682 heat transfer will occur, and only conduction through the gas filled void spaces is assumed.
4683 Larger gas volume regions can experience a significant level of convective heat transfer. The
4684 staff suggests that the applicant demonstrate the existence of convection in the larger gas
4685 regions and quantify the contribution of convection heat transfer to the overall removal of heat
4686 from the package. Traditionally, the staff has maintained that natural convection in horizontal
4687 basket designs should be validated through robust CFD calculations or physical experiments.

4688
4689 4.5.4.1.1 General Guidance on Computational Fluid Dynamics Analyses (HIGH Priority)
4690

4691 Since the computational resources necessary to fully resolve flow between individual fuel pins in
4692 a cask model with numerous fuel assemblies would be enormous, one acceptable approach
4693 would be to treat fuel assemblies as a porous media for applications seeking to credit heat
4694 removal from fuel via internal convection. The reviewer should verify that any CFD approach
4695 utilizes realistic or bounding flow friction factors in the porous media representation of the fuel,
4696 and that friction factors are obtained for each of the limiting fuel assembly types sought as
4697 approved contents for the cask.

4698
4699 An acceptable approach to calculate the friction factors would be to perform a computational
4700 fluid dynamics (CFD) analysis for each type of fuel assembly for the expected operating
4701 conditions (pressure and average gas temperature). From the detailed CFD analysis of a single
4702 fuel assembly, wall shear stresses should be obtained separately for bare fuel rods and for fuel
4703 rods and associated grid straps. The friction factor shall be calculated based on the wall shear
4704 stress method.

4705
4706 The reviewer should evaluate the method used to obtain the friction factors and ensure that the
4707 obtained values are realistic or bounding for the intended fuel assembly types. Also, since the
4708 friction factor is generally very sensitive to the geometric information (dimensions) and fuel
4709 assembly configuration, the reviewer should verify this information by reviewing the fuel
4710 assembly design drawings provided by the applicant.

4711
4712 For ventilated spent fuel storage systems (a canister containing the fuel within an outer
4713 overpack), the mesh spacing (computational cell size) and density between an overpack liner
4714 and canister outer shell wall play an important role when selecting a turbulence model for the air
4715 flow through this annular gap.

4716
4717 The near-wall modeling significantly impacts the fidelity of numerical solutions, inasmuch as
4718 walls are the main source of flow mean vorticity and turbulence. After all, it is in the near-wall
4719 region that the solution variables have large gradients, and the transport of momentum and
4720 other scalar variables occurs more vigorously. Therefore accurate representation of the flow in
4721 the near-wall region determines a successful prediction of wall-bounded turbulent flows. When
4722 dealing with wall effects on the flow usually two modeling options are available to the analyst.
4723 The first one is the use of the semi-empirical formulas called "standard wall functions" which are
4724 used to bridge the viscosity-affected region between the wall and the fully-turbulent core region.
4725 Generally a uniform mesh would be used when these wall functions are invoked. The use of
4726 wall functions obviates the need to modify the turbulence models to account for the presence of
4727 the wall. This modeling approach is usually applicable to flows with high Reynolds number. In
4728 the second approach, the viscosity-affected region is resolved with a mesh all the way to the
4729 wall, including the viscous sublayer. This type of approach is referred to as "near wall
4730 modeling" approach. The dimensionless distance between the wall and the cell center near the
4731 wall (y^+) for the mesh used for this case should generally be around 1. Guidance on how to
4732 apply any of these modeling approaches should be provided in the CFD program
4733 documentation used in the application. Any modeling approach taken should be fully justified
4734 and validated.

4735
4736 To properly characterize the flow (internal, external, annular, etc.), Reynolds number estimates
4737 shall be made using velocities from initial runs for the cooling air in the annulus and helium fill
4738 inside the canister. Reynolds number above 3000 based on the channel hydraulic diameter are

4739 above the critical Reynolds number of 2300 for internal flows, characterizing the flow in the
4740 transitional range between the laminar and turbulent zone. Since these are buoyancy driven
4741 flows, both the Grashof (Gr) number based on the hydraulic diameter of the channel and the
4742 modified Grashof number defined as Graetz number ($Gz = Gr * W/H$) where W and H are the
4743 width and height of the air channel) should also be calculated to properly characterize the
4744 annular flow. On the other hand, buoyancy driven helium flow, cooling the inside of the canister,
4745 generally would be laminar based on both the Grashof and the Reynolds numbers due to higher
4746 kinematic viscosities, and low achieved velocities within the canister.

4747
4748 Actual SNF properties and uncertainties (e.g., friction factors, crud and oxide buildup,
4749 eccentricities, non-uniform axial and radial decay heat profiles) should also be addressed.
4750 Applicants must avoid using an effective thermal conductivity for the cover gas (e.g., helium) in
4751 lieu of a specific convection model.

4752
4753 If applicable, the applicant should evaluate the added heat from components stored with the
4754 SNF assemblies (e.g., control rods, fuel channels, etc.). This would ultimately affect the
4755 maximum predicted cladding temperature.

4756
4757 4.5.4.1.2 General Guidance on Application of Effective Conductivity Models (MEDIUM
4758 Priority)

4759
4760 In addition to a CFD method utilizing a porous media, fuel assemblies may be modeled as a
4761 homogenous region using an effective thermal conductivity (this is a typical approach when
4762 utilizing a finite element analysis approach). The manner in which effective conductivity is
4763 determined for each fuel assembly should be examined by the reviewer. Guidance on effective
4764 thermal conductivity of the fuel is presented in Section 4.5.4.2, "Material Properties."

4765
4766 Use of effective thermal conductivity coefficients for regions within the confinement cask other
4767 than the fuel (e.g., gaps) may overestimate heat transfer. If effective thermal conductivity is
4768 used in this manner, the reviewer should verify that the same values have been determined
4769 from test data that are representative of similar geometry, materials, temperatures, and heat
4770 fluxes used in current application. The reviewer should pay particular attention to the effective
4771 thermal conductivity of neutron shield regions, such as those embedded within thermal fins.
4772 Voids or gaps typically exist as a result of either tolerances or shrinkage, and should be
4773 considered in calculating effective thermal conductivity. Also, the applicant should pay
4774 particular attention to the values assumed for surface emissivities and view factors, as well as
4775 the manner used to account for radiation heat transfer in determining the effective thermal
4776 conductivities.

4777
4778 4.5.4.2 Material Properties (MEDIUM Priority)

4779
4780 The reviewer should coordinate with the materials discipline to verify that the material
4781 compositions and thermal properties are provided for all components used in the calculational
4782 model that the thermal properties used in the safety analysis are appropriate, and that potential
4783 degradation of materials over their service life has been evaluated. Temperature and
4784 anisotropic dependencies of thermal properties should be considered. If regional thermal
4785 properties are determined from a combination of individual materials, the manner in which these
4786 effective properties are calculated should be fully described and justified.

4787
4788 If the thermal model is axisymmetric or three-dimensional, the longitudinal thermal conductivity
4789 should generally be limited to the conductivity of the cladding (weighted by its fractional area)

4790 within the fuel assembly. Gaps between fuel pellets and cracks in the pellets themselves can
4791 result in a considerable uncertainty regarding the contribution of the fuel to longitudinal heat
4792 transfer. High-burnup effects should also be considered in determining the fuel region effective
4793 thermal conductivity.

4794
4795 4.5.4.3 Boundary Conditions (Priority - as indicated)
4796

4797 (MEDIUM Priority) The reviewer should verify that the applicant identifies boundary conditions
4798 for normal, loading, off-normal, and accident conditions. The required boundary conditions
4799 include the decay heat rate from each fuel assembly and the external conditions on the cask
4800 surface. The peak power factor for a fuel assembly should be specified and the peak linear
4801 power (“peaking factor”) of a fuel assembly should be stated for a given active fuel length.

4802
4803 (MEDIUM Priority) The boundary conditions on the cask surface depend on the environment
4804 surrounding the cask. Consequently, the temperature of the environment should be specified
4805 for all simulated conditions, as should the incident and absorbed insolation. The mechanisms
4806 and models for dissipating the absorbed insolation and decay heat from the surface of the cask
4807 to the environment should also be identified and described. The mechanisms for transferring
4808 heat from the cask surface usually consist of natural (free) convection and thermal radiation. A
4809 heat balance on the surface of the cask should be conducted and the results presented in the
4810 applicant’s SAR.

4811
4812 (LOW Priority) The initial temperature distribution of the storage cask system before a fire
4813 accident should be established on the basis of the hottest temperature distribution during
4814 normal or off-normal storage conditions. The duration and flame temperature of the fire should
4815 be specified, as should gas velocities and flame emissivity. The flame and cask surface
4816 emissivities specified in 10 CFR 71.73(c)(4) for a hypothetical accident test of transportation
4817 packages are satisfactory for use with regard to a fire accident involving a storage cask.

4818
4819 (LOW Priority) The applicant should identify and describe the mechanisms and models for
4820 coupling the fire energy to the cask surface. These mechanisms include forced convection in
4821 relation to the flame velocity (5 to 15 m/s, or about 16 to 49 ft/s) as well as thermal radiation. In
4822 addition, justification of the convection coefficients during the fire should be provided. Natural
4823 convection coefficients are not appropriate; as such coefficients imply downward gas flow
4824 adjacent to relatively cool cask walls. In general, for the fire condition, buoyant, upward flow,
4825 driven by hot gasses, will dominate. The orientation of the cask should also be considered.

4826
4827 (LOW Priority) Following the fire, the cask is subject to insolation and content decay heat while
4828 being cooled by natural convection and thermal radiation to the environment. The applicant
4829 should identify the post-fire conditions of the cask, including any changes in surface conditions
4830 and/or geometry that may affect radiation and convection heat losses. Identification and
4831 description of the models used for the analysis of the post-fire processes should also be
4832 provided by the applicant.

4833
4834 4.5.4.4 Computer Codes (HIGH Priority)
4835

4836 The reviewer should verify that the applicant has provided information on any computer-based
4837 modeling as described in Appendix A to Chapter 3.0, “Structural Evaluation,” and review the
4838 thermal analysis submitted by the applicant in accordance with the Appendix.

4839

4840 4.5.4.5 Temperature Calculations (Priority – as indicated)

4841
4842 (MEDIUM - bolted closure/LOW - welded closure) The application should include a table that
4843 lists the maximum and minimum temperatures of all components important to safety under
4844 normal, loading, off-normal, and accident-level conditions. This table should specify the
4845 operating temperature range for each component. The reviewer should verify that temperatures
4846 have been calculated for key components and that they do not exceed the allowable range for
4847 each. Justification shall be provided in the application for any material important to safety that
4848 exceeds acceptable temperature ranges. If compliance with minimum temperature criteria
4849 relies on a specific minimum heat load from the fuel, such heat load shall be quantified and
4850 included as an operating control and a technical specification criterion in SAR Chapter 13.

4851
4852 (MEDIUM Priority) The reviewer should pay particular attention to the maximum temperature of
4853 the cladding. These temperature limits are discussed in Section 4.4.2, "Material and Design
4854 Limits," with review guidance presented in Section 4.5.2, "Material and Design Limits."

4855
4856 (MEDIUM Priority) Some storage systems rely upon natural circulation of air through internal
4857 passages to remove heat from the stored confinement canister. For storage systems with
4858 internal air flow passages, blockage of inlet and/or outlet flow is an accident situation that should
4859 be evaluated. Total blockage of all inlets and outlets may result in fuel heatup, which has been
4860 assumed to approach adiabatic conditions. To ensure that blockages do not go undetected for
4861 significant periods, the NRC has required objective evidence that inlet and outlet flows are not
4862 obstructed. Consequently, for these types of storage systems, the NRC has accepted periodic
4863 visual inspection of the vents coupled with temperature measurements to verify proper thermal
4864 performance and detect flow blockages. The inspections should take place within an interval
4865 that will allow sufficient time for corrective actions to be taken before the accident temperature is
4866 reached. The inspection interval should be more frequent than the time interval required for the
4867 fuel to heatup to the established accident temperature criteria, assuming a total blockage of all
4868 inlets and outlets.

4869
4870 (MEDIUM Priority) The review of the heatup calculations should specifically address any
4871 assumptions regarding limiting components and quasi-steady state responses. The initial
4872 ambient temperature for the heatup calculations should bound the maximum "normal condition"
4873 temperature. The resulting heatup time history should be included in the SAR documentation,
4874 and should support the proposed inspection and monitoring intervals. This information is also
4875 useful in developing contingency operation procedures, since it indicates the available time in
4876 which to take corrective actions before the fuel accident temperature criteria may be exceeded.

4877
4878 (HIGH Priority) Some storage systems may use a transfer cask to move the loaded confinement
4879 canister from the fuel handling building to the independent spent fuel storage installation (ISFSI)
4880 site. When the canister is within the transfer cask, the rate of cooling is typically less than for
4881 normal operation. Therefore, fuel cladding temperatures are expected to be higher than for
4882 normal storage conditions.

4883
4884 (HIGH Priority) The reviewer should examine the temperature distribution calculations for the
4885 canister inside the transfer cask and verify that heat transfer through gap regions has been
4886 treated in a conservative manner, and that material properties and dimensions of the transfer
4887 cask are consistent with the design data defined in the SAR documentation. The initial ambient
4888 temperature should be the maximum "normal condition" temperature. Cask preparation for
4889 storage or unloading operations may include situations in which the canister is evacuated while
4890 it is in the transfer cask. If the fuel cladding temperature calculation is based on heatup over a

4891 limited time period for cask drying operations, the reviewer should verify that limiting conditions
4892 for the operations have been imposed in the technical specifications. Such limiting conditions
4893 should ensure that the temperature will remain acceptable during the operations, and that
4894 normal cooling will begin before the temperature criterion is exceeded.

4895
4896 (HIGH Priority) During wet fuel transfer operations, the liquid in the fuel canister should not be
4897 permitted to boil. This practice avoids uncontrolled pressures on the canister and the connected
4898 dewatering, purging, and recharging system(s), unacceptable discharge of liquids which may be
4899 providing radiation shielding, and a potentially unacceptable reduction in the safety margin. The
4900 reviewer should ensure that to prevent any of the above conditions, an adequate subcooling
4901 margin is identified in both the SAR and corresponding operating procedures to prevent boiling.
4902 This margin may be cask-specific, depending on the design of the fuel basket and key
4903 assumptions used in the criticality analysis. The reviewer should ensure that the applicant
4904 reviews the heatup and time-to-boil calculations and assesses whether any technical
4905 specification or limiting conditions for operation are needed. Heatup calculations should be
4906 established on the basis of the SNF pool's technical specification maximum temperature limit
4907 (typically 46°C (115°F)).

4908
4909 (HIGH Priority) For unloading operations, the thermal reviewer should ensure that the applicant
4910 evaluates temperature and pressure calculations supporting procedural steps presented in SAR
4911 Chapter 9, "Operating Procedures Evaluation," for cask cooldown and reflooding of the cask
4912 internals. To ensure that the cask does not overpressurize and that the fuel assemblies are not
4913 subjected to excess thermal stresses, the applicant's analysis should specify and justify the
4914 appropriate temperature and flow rate of the quench fluid, assuming maximum fuel cladding
4915 temperatures in the unloading configuration. This analysis should also be referenced in Chapter
4916 12, "Accident Analyses Evaluation," of the SAR as having been considered in the development
4917 of thermal models for the unloading procedures, and be included, as appropriate, in the
4918 technical specifications. The thermal reviewer should provide thermal profiles to the materials
4919 reviewer so that latter can determine if the applicant has adequately addressed the issue of fuel
4920 rod response to a reflood incident in Chapter 8, "Materials Evaluation".

4921
4922 (LOW Priority) The most extreme thermal conditions may result from credible ambient
4923 temperatures, temperature-time histories, an adjacent fire, or any off-normal or design-basis
4924 event (DBE) resulting in blockage of ventilation passages. The worst-case structural loads may
4925 occur at temperatures lower than those of design-basis accidents (DBAs) or natural phenomena
4926 since load combination expressions effectively require greater safety factors for normal and off-
4927 normal analyses than for any DBE. Typically, fire has been the worst-case accident thermal
4928 condition for storage systems without internal air flow passages.

4929
4930 (LOW Priority) The burning of fuel and other combustibles associated with vehicles involved in
4931 transfer operations should, at a minimum, be presumed to be a DBE with the cask in the most
4932 exposed situation during transfer or loading into storage. Fire parameters included in 10 CFR
4933 71.73 have been accepted for characterizing the heat transfer during the in-storage fire.
4934 However, a bounding analysis that limits the fuel source thus limits the length of the fire (e.g., by
4935 limiting the source to the fuel in the transporter) has also been accepted.

4936
4937 (LOW Priority) Some structures, systems, and components (SSC) may experience the most
4938 severe conditions if exposure to high temperatures is followed by dousing with water (such as
4939 rain or fire suppression activities). A small amount of exterior concrete spalling may result from
4940 a fire, the application of fire suppression water, rain on heated surfaces or other high-
4941 temperature condition. The damage from these events is readily detectable and appropriate

4942 recovery or corrective measures may be presumed. Therefore, the loss of such a small amount
4943 of shielding material is not expected to cause a storage system to exceed the regulatory
4944 requirements in 10 CFR 72.106 and need not be estimated or evaluated in the SAR. The NRC
4945 accepts that concrete temperatures may exceed the temperature criteria of American Concrete
4946 Institute (ACI) 349 for accidents if the temperatures result from a fire. In that case, corrective
4947 action may be required for continued safe storage.

4948
4949 (LOW Priority) The methods that are acceptable for analyzing and reviewing the consequences
4950 of a fire depend upon the duration of the fire and the margin between the predicted
4951 temperatures and the actual thermal limits of the components. A fire of sufficient duration, or
4952 one in which material temperatures are close to the criteria of their acceptable operational
4953 range, will require a detailed model of the cask and its contents. Cask system components
4954 (e.g., the neutron shield) may be assumed to be intact at the start of the fire.

4955
4956 (LOW Priority) If a cask tipover is a credible accident, the reviewer should verify that the
4957 applicant has evaluated the effect on cask and fuel temperatures in the new configuration. An
4958 analysis may be warranted when a significant portion of heat removal capability is attributed to
4959 internal convection if a change in orientation of that cask may have a significant effect.

4960 4.5.4.6 Pressure Analysis (LOW Priority)

4961
4962 Pressure calculations should be performed using the ideal gas law (i.e., $PV = nRT$ where P is
4963 pressure, V is volume, n is the number of moles of a gas, R is a constant for a given gas, and T
4964 is the absolute temperature) and summing the partial pressures of each of the gas constituents
4965 in the cask cavity. The application should identify the method and all assumptions used in the
4966 pressure analysis, including the determination of the fission gas inventory.

4967
4968 It is necessary to consider the temperature distribution of all components within the cask cavity
4969 and the cavity walls in calculating the gas pressure in the cavity. For the fire accident analysis,
4970 the application should identify the maximum gas temperature reached during the post-fire
4971 accident phase, explain the method used to determine the average gas temperature, and
4972 specify the time in the accident at which the peak gas temperature is attained.

4973
4974 This pressure also depends on the free volume in the cask cavity, the amount (moles) of cover
4975 gas (helium) in the cavity, and the amount of gases released from ruptured fuel pins. The free
4976 volume calculation should be reviewed to determine if all components internal to the cask cavity
4977 (e.g., fuel assemblies, basket, structural supports, spacer disks, reactor control components)
4978 have been properly considered.

4979
4980 The NRC accepts that normal conditions occur with less than 1 percent of the fuel rods failed,
4981 off-normal conditions occur with up to 10 percent of the fuel rods ruptured, and 100 percent of
4982 the fuel rods will have ruptured following a DBE. The NRC also accepts that a minimum of
4983 100 percent of the fill gas and 30 percent of the significant radioactive gases (e.g., ^3H , Kr, and
4984 Xe) within a ruptured fuel rod is available for release into the cask cavity.

4985
4986 Under the conditions where any of the cask component temperatures are close (within 5
4987 percent) to their limiting values during an accident or the Maximum Normal Operating Pressure
4988 (MNOP) is within 10 percent of its design basis pressure, or any other special conditions, the
4989 applicant should consider, by analysis, the potential impact of the fission gas in the canister to
4990 the cask component temperature limits and the cask internal pressurization.

4991
4992

4993 The reviewer should coordinate with the structural reviewer to verify that the containment
4994 pressure of the cask is within its design limits for normal and accident conditions.

4995
4996 4.5.4.7 Confirmatory Analysis (HIGH Priority)
4997

4998 Reviewers may need to perform a confirmatory analysis of the thermal performance of the cask
4999 SSCs identified as important to safety. Confirmatory analyses are recommended where
5000 margins between the calculated temperatures and prescribed component temperature limits are
5001 small, where particularly complex thermal analyses are submitted by applicants, or where the
5002 applicant is submitting a new thermal methodology or analysis approach.
5003

5004 The application should be reviewed to ensure that the applicant made the correct assumptions
5005 and provided the correct input, and that the output is consistent with established physical
5006 (thermal) behavior. These results should specifically include steady-state temperature
5007 distributions, local heat balances, temperatures reached and temperature distributions within
5008 any reinforced concrete SSCs, and cask cavity pressures for the bounding ambient
5009 temperatures.
5010

5011 To provide the most reliable confirmation, confirmatory analysis should, to the degree possible,
5012 use a different thermal analysis method than that used by the applicant. The code used for the
5013 confirmatory analysis may be the same as or different from that used by the applicant.
5014 Regardless, a review of the applicant's analytical approach and analysis models should be
5015 considered part of the overall confirmatory analysis. Similar confirmation of accident
5016 temperatures (e.g., during a fire) should be performed, as applicable to the SAR analysis.
5017

5018 If a full confirmatory analysis is not deemed necessary, the minimum confirmatory review should
5019 include verifying that key design parameters have been appropriately determined and correctly
5020 expressed as input into the computer program(s) used for the thermal analysis. Key parameters
5021 include proper dimensions, material properties (including surface emissivities and view factors
5022 for radiation), and definition of heat sources. A heat balance at the outer surface of the cask
5023 should be performed to verify that the heat from the SNF and insolation, balance that removed
5024 by convection and radiation. Correlations for the heat transfer coefficient should then be
5025 assessed to confirm that they are appropriate for the existing storage conditions. The
5026 temperature of the cask's inner surface should be estimated by calculating the temperature
5027 distribution across the cask body with simple heat balance approximations. Finally, the
5028 difference between the cask's inner surface temperature and the maximum cladding
5029 temperature should be compared with that of similar casks and baskets reviewed in previous
5030 SARs.
5031

5032 As discussed above, a more detailed confirmatory analysis may be required, and could include
5033 a model of a portion of the cask or basket to ensure that the SAR results are realistic and
5034 conservative. A more extensive confirmatory analysis may involve the full geometry of the cask,
5035 with relevant component details, to determine temperature distributions in the cask system.
5036

5037 Additional guidance on review of analytical models and conduct of confirmatory analyses can be
5038 found in Appendix 3A, "Computational Modeling Software."
5039

5040 As an alternative to a confirmatory analysis, the applicant may be required to perform design-
5041 verification testing of an as-built cask system to confirm the thermal analyses presented in the
5042 SAR. Such testing may include verifying gap conductance values assumed in modeling thermal
5043 resistance. The test conditions, configuration, and type and location of instrumentation used, if
5044

5044 any, should be sufficiently described in SAR Chapter 10, "Acceptance Criteria and
5045 Maintenance."

5046

5047 **4.6 Evaluation Findings**

5048

5049 The reviewer should review the 10 CFR Part 72 acceptance criteria and provide a summary
5050 statement for each. These statements should be similar to the following model:

5051

F4.1 Structures, systems, and components (SSCs) important to safety are described
5052 in sufficient detail in Chapters _____ of the SAR to enable an evaluation of their
5053 thermal effectiveness. Cask SSCs important to safety remain within their
5054 operating temperature ranges.

5055

F4.2 The [cask designation] is designed with a heat-removal capability having
5056 verifiability and reliability consistent with its importance to safety. The cask is
5057 designed to provide adequate heat removal capacity without active cooling
5058 systems.

5059

F4.3 The spent fuel cladding is protected against degradation leading to gross
5060 ruptures by maintaining the cladding temperature for _____ -year cooled fuel
5061 below _____ °C (____ °F) in an [applicable gas] environment. Protection of the
5062 cladding against degradation is expected to allow ready retrieval of spent fuel for
5063 further processing or disposal.

5064

5065 The reviewer should provide a summary statement similar to the following:

5066

5067 "The staff concludes that the thermal design of the [cask designation] is in compliance
5068 with 10 CFR Part 72, and that the applicable design and acceptance criteria have been
5069 satisfied. The evaluation of the thermal design provides reasonable assurance that the
5070 [cask designation] will allow safe storage of spent fuel for a licensed (certified) life of
5071 years. This finding is reached on the basis of a review that considered the regulation
5072 itself, appropriate regulatory guides, applicable codes and standards, and accepted
5073 engineering practices."
5074
5075
5076

5 CONFINEMENT EVALUATION

5.1 Review Objective

In this portion of the dry storage system (DSS) review, the U.S. Nuclear Regulatory Commission (NRC) evaluates the confinement features and capabilities of the proposed cask system. In conducting this evaluation, the NRC staff seeks to ensure that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

5.2 Areas of Review

This chapter of the DSS Standard Review Plan (SRP) provides guidance for use in evaluating the design and analysis of the proposed cask confinement system for normal, off-normal, and accident conditions. This evaluation includes a more detailed assessment of the confinement-related design features and criteria initially presented in Chapters 1, "General Information Evaluation" and 2, "Principle Design Criteria Evaluation" of the applicant's Safety Analysis Report (SAR), as well as the proposed confinement monitoring capability, if applicable. In addition, the NRC staff assesses the potential releases of radionuclides associated with spent fuel by independently estimating their potential leakage to the environment and the subsequent impact on a hypothetical individual located at or beyond the controlled area boundary.

As prescribed in U.S. Code of Federal Regulations (CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, "Energy" (10 CFR Part 72), the regulatory requirements 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 leak tightness of the confinement). Thus, an overall assessment of the compliance of the proposed DSS with these regulatory limits is deferred to Chapter 11, "Radiation Protection Evaluation," of this SRP. In addition, the performance of the cask confinement system under accident-level conditions, as evaluated in this chapter, may also be addressed in the overall accident analyses as discussed in Chapter 12, "Accident Analyses Evaluation," of this SRP.

As described in SRP Section 5.5, "Review Procedures," a comprehensive confinement evaluation should encompass the following areas of review:

Confinement Design Characteristics

Design Criteria

Design Features

Confinement Monitoring Capability

Nuclides with Potential for Release

Confinement Analyses

Normal Conditions

Off-Normal Conditions (Anticipated Occurrences)

Design Basis Accident Conditions (Including Natural Phenomenon Events)

Supplemental Information

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5.3 Regulatory Requirements

This section presents a summary matrix of the portions of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should read the exact referenced regulatory language. Table 5-1 matches the relevant regulatory requirements associated with this chapter to the areas of review.

Table 5-1 Relationship of Regulations and Areas of Review			
Areas of Review	10 CFR Part 72 Regulations		
	72.104 (a)	72.122(a), (b)(1), (h)(1), (4), (i)	72.236 (d), (e), (i), (j), (l)
Confinement Design Characteristics		●	●
Confinement Monitoring Capability		●	
Nuclides with Potential for Release	●		●
Confinement Analyses	●	●	●

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5.4 Acceptance Criteria

In general, the DSS confinement evaluation seeks to ensure that the proposed design fulfills the following acceptance criteria that the NRC staff considers to be minimally acceptable to meet the confinement requirements of 10 CFR Part 72.

5.4.1 Confinement Design Characteristics

The design should provide redundant sealing of the confinement boundary (10 CFR 72.236(e)). Typically, this means that field closures of the confinement boundary should either have two seal welds or two metallic O-ring seals.

The confinement design should be consistent with the regulatory requirements as well as the applicant's "General Design Criteria" reviewed in Chapter 2, "Principal Design Criteria Evaluation," of this SRP. The NRC staff has previously accepted construction of the primary confinement barrier in conformance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsections NB or NC. This code defines the standards for all aspects of construction including materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components. In such instances, the staff has relied upon Section III to define the minimum acceptable margin of safety. Therefore, the applicant must fully document and completely justify any deviations from the specifications of Section III. In some cases, after careful and deliberate consideration, the staff has made exceptions to this requirement. In addition, the ASME has published in 2005 Division

5161 3 to Section III which is written specifically for Containments for the Transportation and Storage
5162 of Spent Nuclear Fuel and is considered to be the governing code for this component, but has
5163 not yet been reviewed and endorsed by the NRC.
5164

5165 The design must provide a nonreactive environment to protect fuel assemblies against fuel
5166 cladding degradation, which might otherwise lead to gross rupture (PNL, 1987). Measures for
5167 providing a nonreactive environment within the confinement cask typically include drying and
5168 backfilling with a nonreactive cover gas (such as helium). Experimental data have not
5169 demonstrated an acceptably low oxidation rate for UO₂ spent fuel over the 20-year licensing
5170 period to permit safe storage in an air atmosphere during dry storage. Therefore, to reduce the
5171 potential for fuel oxidation and subsequent cladding failure, an inert atmosphere (e.g., helium
5172 cover gas) has been used for storing UO₂ spent fuel in a dry environment. See Chapter 9,
5173 "Operating Procedures Evaluation," of this SRP for more detailed information on the cover gas
5174 filling process. Note that other fuel types, such as graphite fuels for the high-temperature gas-
5175 cooled reactors (HTGRs), may not exhibit the same oxidation reactions as UO₂ fuels and,
5176 therefore, may not require an inert atmosphere. Applicants proposing to use atmospheres other
5177 than inert gas should discuss how the fuel and cladding will be protected from oxidation.
5178

5179 **5.4.2 Confinement Monitoring Capability**

5180
5181 The reviewer should ensure the application describes the proposed monitoring capability and/or
5182 surveillance plans for mechanical closure seals. In instances involving welded closures, the
5183 staff has previously accepted that no closure monitoring system is required. This practice is
5184 consistent with the fact that other welded joints in the confinement system are not monitored.
5185 However, the lack of a closure monitoring system has typically been coupled with a periodic
5186 surveillance program that would enable the licensee to take timely and appropriate corrective
5187 actions to maintain safe storage conditions if closure degradation occurred.
5188

5189 To show compliance with the requirement for continuous monitoring, 10 CFR Part 72.122(h)(4),
5190 cask vendors have proposed, and the staff has accepted, routine surveillance programs and
5191 active instrumentation to meet the continuous monitoring requirements.
5192

5193 **5.4.3 Nuclides with Potential for Release**

5194
5195 The applicant must estimate the maximum credible quantity of radionuclides with potential for
5196 release to the environment. The radionuclides potentially available for release to the
5197 environment are based on the radiological source term evaluation presented in Chapter 6,
5198 "Shielding Evaluation," of this SRP.
5199

5200 **5.4.4 Confinement Analyses**

5201
5202 The application should specify the maximum allowed leakage rates for the total primary
5203 confinement boundary and redundant seals. Applicants frequently display this information in
5204 tabular form including the leakage rate of each seal. The maximum allowed leakage rate is the
5205 "as tested" leak rate measured by the leak test performed on the cask field closure. Generally,
5206 as discussed below, the allowable leakage rate must be evaluated for its radiological
5207 consequences and its effect on maintaining an inert atmosphere within the cask. However, the
5208 analyses discussed below are unnecessary¹ for storage casks including its closure lid that are

¹ For casks that are demonstrated to be leak tight, the review procedures discussed in Sections 5.5.3 and 5.5.4 are not applicable.

5209 designed and tested to be “leak tight” as defined in the American National Standards Institute
5210 (ANSI), Institute for Nuclear Materials Management’s “American National Standard for Leakage
5211 Tests on Packages for Shipment of Radioactive Materials” (ANSI N14.5-1997).
5212

5213 • The analysis of potential releases should be consistent with the methods
5214 described in ANSI N14.5-1997 (ANSI, 1997).
5215

5216 • During normal operations and anticipated occurrences, dose calculations based
5217 on the allowable leakage rate must demonstrate that the annual dose equivalent
5218 to any real individual who is located beyond the controlled area does not exceed
5219 the limits given in 10 CFR 72.104(a).
5220

5221 • For any design-basis accident, dose calculations based on the allowable leakage
5222 rate must demonstrate that an individual at the boundary or beyond the nearest
5223 boundary of the controlled area does not receive a dose that exceeds the limits
5224 given in 10 CFR 72.106(b)-(discussed further in Chapter 12, “Accident Analyses
5225 Evaluation”)
5226

5227 • The analysis of potential releases must demonstrate that an inert atmosphere will
5228 be maintained within the cask during the storage lifetime.
5229

5230 • For casks that employ a pressurized inert gas to facilitate internal natural
5231 convection heat transfer, the analysis of potential releases must demonstrate that
5232 the pressurized atmosphere will be maintained within the cask during the storage
5233 lifetime.
5234

5235 **5.4.5 Supplemental Information**
5236

5237 The reviewer should ensure all supportive information or documentation that justifies
5238 assumptions or analytical procedures is provided in the application.
5239

5240 **5.5 Review Procedures**
5241

5242 Figure 5-1 presents an overview of the evaluation process for coordination with other review
5243 disciplines.
5244

5245 **5.5.1 Confinement Design Characteristics (MEDIUM Priority)**
5246

5247 **5.5.1.1 Design Criteria**
5248

5249 The reviewer should examine the principal design criteria presented in SAR Chapter 2 as well
5250 as any additional detail provided in SAR Chapter 5, “Confinement.”
5251

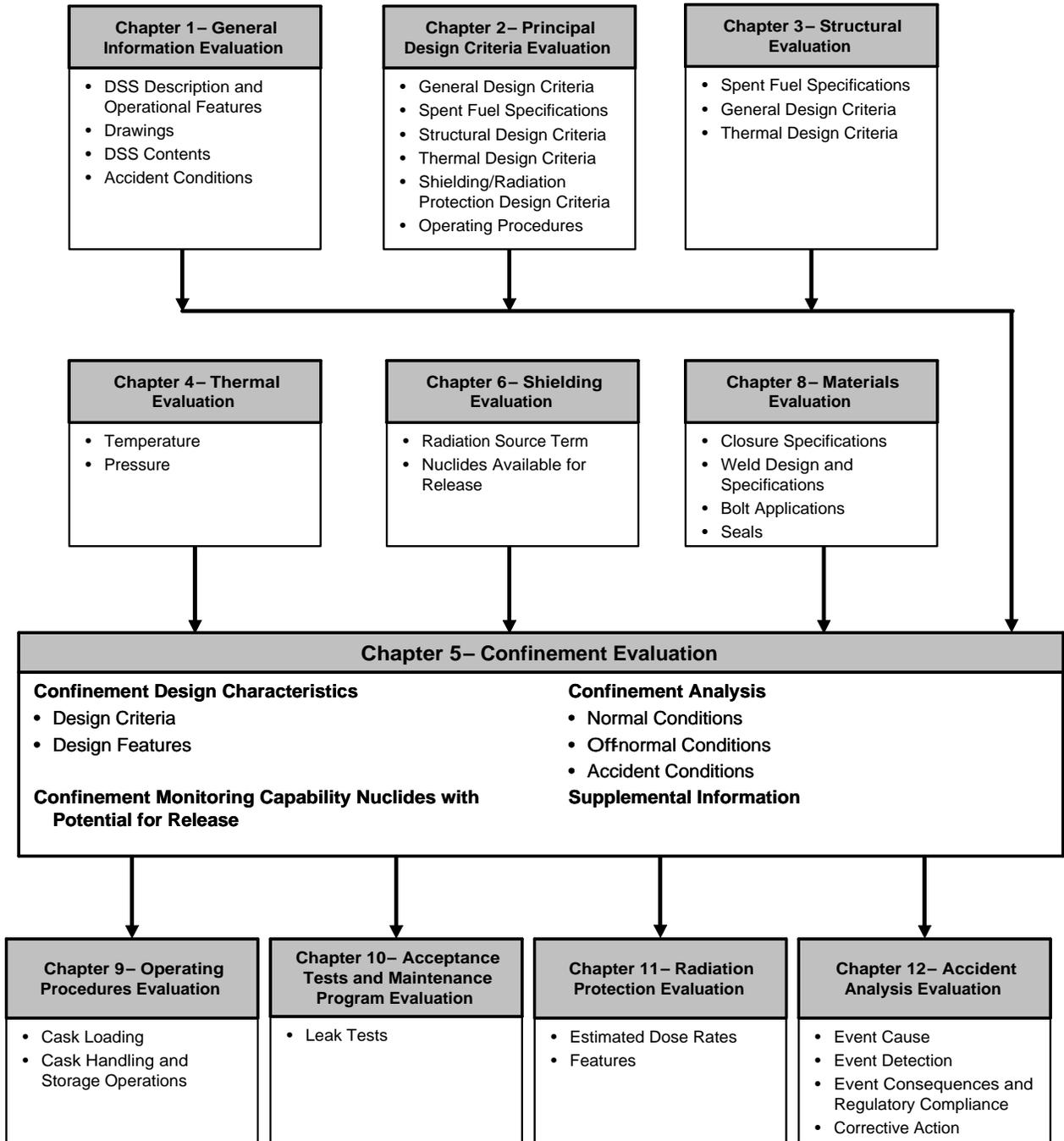


Figure 5-1 Overview of the Confinement Evaluation

5255 5.5.1.2 Design Features
5256

5257 The reviewer should examine the general description of the cask presented in SAR Chapter 1,
5258 "General Description," as well as any additional information provided in SAR Chapter 5,
5259 "Confinement Evaluation". All drawings, figures, and tables describing confinement features
5260 should be sufficiently detailed to stand alone.

5261
5262 The reviewer should verify that the applicant has clearly identified the confinement boundaries.
5263 This identification should include the confinement vessel, its penetrations, valves, seals, welds,
5264 and closure devices, and corresponding information concerning the redundant sealing.
5265

5266 The reviewer should verify that the design and procedures provide for drying and evacuation of
5267 the cask interior as part of the loading operations. Also, the reviewer should verify that the
5268 confinement design is acceptable for the pressures that may be experienced during normal, off-
5269 normal and accident conditions.
5270

5271 The reviewer should verify that, on completion of cask loading, the gas fill of the cask interior is
5272 at a pressure level that is expected to maintain a nonreactive environment and heat transfer
5273 capabilities for at least the 20-year storage life of the cask interior under both normal and off-
5274 normal conditions and events. This verification can include pressure testing, seal monitoring,
5275 and maintenance for casks with seals that are not welded if these are included in Chapter 13,
5276 "Technical Specifications and Operating Controls and Limits Evaluation," of this SRP as
5277 conditions of use. Acceptance tests for pressure testing are described in Section 10.5.1.1,
5278 "Structural/Pressure Tests," of this SRP. The NRC has previously accepted specification of an
5279 overpressure of approximately 14 kPa (~2 psig) and cask leak testing as conditions of use for
5280 satisfying this requirement. However, this general rule is not applicable to those designs that
5281 employ a pressurized content (i.e., to several atmospheres) to facilitate natural circulation
5282 cooling within the canister
5283

5284 The reviewer should coordinate with the structural and materials disciplines respectively
5285 reviewing Chapter 3, "Structural Evaluation," and Chapter 8, "Materials Evaluation," of this SRP
5286 to ensure that the applicant has provided proper specifications for all welds and, if applicable,
5287 that the bolt torque for closure devices is adequate and properly specified. If applicable, the
5288 reviewer should verify the capability of the seal to maintain long-term closure. Because of the
5289 performance requirements over the 20-year license period, the reviewer should evaluate the
5290 potential for seal deterioration associated with bolted closures. The NRC staff has previously
5291 accepted only metallic seals for the primary confinement. This review should be coordinated
5292 with the thermal discipline to ensure that the operational temperature range for the seals
5293 (specified by the manufacturer) will not be exceeded.
5294

5295 The staff has concluded that welded canisters can be used as a confinement system provided
5296 that the following design/qualification guidance is met:
5297

- 5298 • The canister is constructed from austenitic stainless steel.
- 5299
- 5300 • The canister closure welds meet the guidance of Section 8.5.2.3, "Weld Design
5301 and Specifications," of this SRP.
- 5302
- 5303 • The canister maintains its confinement integrity during normal conditions,
5304 anticipated occurrences, and credible accidents and natural phenomena as
5305 required in 10 CFR Part 72.

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- The canister shell has been helium leak tested prior its loading as required by 10 CFR 72.236(i). This test demonstrates that the canister is free of defects that could lead to a leakage rate greater than the design basis leakage rate which could result in doses at the control area boundary in excess of the regulatory limits.
 - Records documenting the fabrication and closure welding of canisters shall comply with the provisions of 10 CFR Part 72.174, "Quality Assurance Records" and SRP Section 8.5.2.3. Records storage should comply with ANSI N45.2.9, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants."
 - Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with a NRC-approved quality assurance program as required in 10 CFR Part 72, Subpart G, "Quality Assurance."

5324 The qualification standards discussed above provide a sufficient alternative to the fabrication, periodic, and pre-shipment leak-testing requirements of ANSI 14.5 for the final closure welds.

5325 **5.5.2 Confinement Monitoring Capability (LOW Priority)**

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5328
5329 The NRC staff has found that casks closed entirely by welding do not require seal monitoring. However, for casks with bolted closures, the staff has found that a seal monitoring system is required to adequately demonstrate that seals can function to limit releases and maintain a helium atmosphere in the cask for the term of the 10 CFR Part 72 general license. A seal monitoring system, combined with periodic surveillance, enables the licensee to determine when to take corrective action to maintain safe storage conditions.

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5336 Although the details of the monitoring system may vary, the general design approach has been to pressurize the region between the redundant seals with a nonreactive gas to a pressure greater than that of the cask cavity and the atmosphere. The monitoring system is leak tested to the same leak rate as the confinement boundary. Installed instrumentation is routinely checked per surveillance requirements. A decrease in pressure between these seals indicates that the nonreactive gas is leaking either into the cask cavity or out to the atmosphere. For normal operations, radioactive material should not be able to leak to the atmosphere; hence, this design allows for detecting a faulty seal without radiological consequence. Note that the volume between the redundant seals should be pressurized using a nonreactive gas, thereby preventing contamination of the interior cover gas.

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5347 The staff has accepted monitoring systems as not important to safety and classified them as Category B under the guidelines of NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety (INEL-95/0551)." Although its function is to monitor confinement seal integrity, the failure of the monitoring system alone does not result in a gross release of radioactive material. Consequently, the monitoring system for bolted closures need not be designed to the same requirements as the confinement boundary (i.e., ASME Section III).

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Dependant on the monitoring system design, there could be a lag time before the monitoring system indicates a postulated degraded seal leakage condition. Degraded seal leakage is

5357 leakage greater than the tested rate that is not identified within a few monitoring system
5358 surveillance cycles. The occurrence of a degraded seal without detection is considered a
5359 “latent” condition and should be presumed to exist concurrently with other off-normal and
5360 design-basis events (see Section 2.5.2.2, “External Conditions,” of this SRP). Note that once
5361 the degraded seal condition is detected, the cask user will initiate corrective actions.
5362

5363 For the “latent” condition, the monitoring system boundary would remain intact and this
5364 condition would be bounded by the off-normal analysis. If the monitoring system would not
5365 maintain integrity under design-basis accident conditions, additional safety analysis may be
5366 necessary. The staff recognizes that the possibility of a degraded seal condition is small and
5367 that the possibility of a degraded seal condition concurrent with a design-basis event that
5368 breaches the monitoring system pressure boundary is very remote. However, these
5369 probabilities have not been quantified. To address this concern, the staff accepts a
5370 demonstration that the dose consequences of this event are within the limits of
5371 10 CFR 72.106(b).
5372

5373 The reviewer should examine the specified pressure of the gas in the monitored region to verify
5374 that it is higher than both the cask cavity and the atmosphere. The reviewer should coordinate
5375 with the structural and thermal reviewers associated with Chapters 3 and 4 of this SRP to verify
5376 the pressure in the cask cavity.
5377

5378 The reviewer should examine the applicant's analysis to verify that the total volume of gas in the
5379 cavity is such that normal seal leakage will not cause all of the gas to escape over the lifetime of
5380 the cask. The proposed maximum leakage rate should be based on the confinement evaluation
5381 described in Sections 5.5.3 and 5.5.4 of this SRP. The maximum allowable leakage rate should
5382 be specified as a minimum acceptance test criterion in SAR Chapter 9, “Acceptance Criteria
5383 and Maintenance Program,” and Chapter 13, “Technical Specifications and Operating Controls
5384 and Limits Review,” even though the actual leakage rate of the seals is expected to be
5385 significantly lower.
5386

5387 For redundant welded closures, the reviewer should ensure that the applicant has provided
5388 adequate justification that the welds have been sufficiently designed, fabricated, tested and
5389 examined to ensure that the weld will behave similarly to the adjacent parent material of the
5390 cask.
5391

5392 The reviewer should verify that any leakage test, monitoring, or surveillance conditions are
5393 appropriately specified in SAR Chapter 10 “Acceptance Tests and Maintenance Program
5394 Evaluation”; Chapter 12, “Accident Analyses”; Chapter 13, “Technical Specifications and
5395 Operational Controls and Limits Evaluation” ; and/or the Certificate of Compliance (CoC).
5396

5397 **5.5.3 Nuclides with Potential for Release (LOW Priority)**

5398

5399 The quantities of radioactive nuclides are often presented in the SAR Chapter 6, “Shielding
5400 Evaluation,” since they are generally determined during the evaluation of gamma and neutron
5401 source terms in the shielding analysis. The reviewer should coordinate with the shielding
5402 discipline to verify that the applicant has adequately developed the source term.
5403

5404 For determination of the radionuclide inventory available for release, the NRC staff has
5405 accepted, as a minimum for the analysis, the activity from the ⁶⁰Co in the crud, the activity from
5406 iodine, fission products that contribute greater than 0.1 percent of design basis fuel activity, and
5407 actinide activity that contributes greater than 0.01 percent of the design basis activity. In some

5408 cases, the applicant may have to consider additional radioactive nuclides, depending upon the
 5409 specific analysis. The total activity of the design basis fuel should be based on the cask design
 5410 loading that yields the bounding radionuclide inventory (considering initial enrichment, burnup,
 5411 and cool time).

5412
 5413 The staff has determined that, as a minimum, the fractions of radioactive materials available for
 5414 release from spent nuclear fuel (SNF), provided in Table 5-2 for pressurized-water reactor
 5415 (PWR) fuel and boiling-water reactor (BWR) fuel for normal, anticipated occurrences (off-
 5416 normal) and accident-level conditions, should be used in the confinement analysis to
 5417 demonstrate compliance with 10 CFR Part 72. These fractions account for radionuclides
 5418 trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that is not
 5419 releasable to the environment under credible normal, off-normal, and accident-level conditions.
 5420 Other release fractions may be used in the analysis provided the applicant properly justifies the
 5421 basis for their usage. For example, the staff has accepted, with adequate justification, reduction
 5422 of the mass fraction of fuel fines that can be released from the cask. Also, for the applicant to
 5423 utilize the release fractions in Table 5-2, the reviewer should ensure that the condition of the fuel
 5424 described in the SAR is bounded by the experimental data presented in NUREG/CR-6487.
 5425 Specifically, this experimental data is based on the release from a single breach of one fuel rod
 5426 and this data should not be used for spent fuel described as damaged.

5427
 5428 The staff has accepted the rod breakage fractions in Section 4.5.4.6, "Pressure Analysis," of this
 5429 SRP for the confinement evaluations. It is important to recognize that confinement boundary
 5430 failure under design basis normal or accident-level conditions is not acceptable. Confinement
 5431 boundary structural integrity during design basis conditions is confirmed by the structural
 5432 analysis. The confinement analyses demonstrate that, at the measured leakage rates and
 5433 assumed nominal meteorological conditions, the requirements of 10 CFR 72.104(a) and
 5434 10 CFR 72.106(b) can be met. Each independent spent fuel storage installation (ISFSI),
 5435 whether it is a site-specific or general license, is also required to have a site-specific
 5436 confinement analysis and dose assessment to demonstrate compliance with these regulations.
 5437

Table 5-2 Fractions of Radioactive Materials Available for Release from Spent Fuel^a

Variable	Fractions Available for Release ^b	
	PWR and BWR Fuel	
	Normal and Off-normal Conditions	Design Basis Accident Conditions
Fraction of Fuel Rods Assumed to Fail	0.01 (normal) 0.10 (off-normal)	1
Fraction of Gases Released Due to a Cladding Breach, f_G^c	0.3	0.3
Fraction of Volatiles Released Due to a Cladding Breach, f_V^c	2×10^{-4}	2×10^{-4}
Mass Fraction of Fuel Released as Fines Due to Cladding Breach, f_F	3×10^{-5}	3×10^{-5}

Table 5-2 Fractions of Radioactive Materials Available for Release from Spent Fuel^a

Variable	Fractions Available for Release ^b	
	PWR and BWR Fuel	
	Normal and Off-normal Conditions	Design Basis Accident Conditions
Fraction of Crud that Spalls Off Cladding, f_c	0.15 ^d	1.0 ^d

- a Values in this table are taken from NUREG/CR-6487.
- b Except for Co-60, only failed fuel rods contribute significantly to the release. Total fraction of radionuclides available for release should be multiplied by the fraction of fuel rods assumed to have failed.
- c In accordance with NUREG/CR-6487, gases species include H-3, I-129, Kr-81, Kr-85, and Xe-127; volatile species include Cs-134, Cs-135, Cs-137, Ru-103, Ru-106, Sr-89, and Sr-90.
- d The source of radioactivity in crud is Co-60 on fuel rods. At the time of discharge from the reactor, the specific activity, S_c , is estimated to be 140 $\mu\text{Ci}/\text{cm}^2$ for PWRs and 1254 $\mu\text{Ci}/\text{cm}^2$ for BWRs. Total Co-60 activity is this estimate times the total surface area of all rods in the cask (Sandoval, et al., 1991). Decay of Co-60 to determine activity at the minimum time before loading is acceptable.

5.5.4 Confinement Analyses (MEDIUM Priority)

The reviewer should examine the applicant’s confinement analysis and the resulting doses for the normal, off-normal, and accident conditions at the controlled area boundary.

The analysis typically includes the following common elements:

- Calculation of the specific activity (Ci/cm^3) for each radioactive isotope in the cask cavity based on rod breakage fractions, release fractions, isotopic inventory, and cavity free volume.
- Using the tested leak rate and conditions during testing as input parameters, calculation of the adjusted maximum seal leakage rates (cm^3/s) under normal, off-normal, and accident conditions (e.g., temperatures and pressures).
- Calculation of isotope specific leak rates ($Q_i\text{-Ci}/\text{s}$) by multiplying the isotope specific activity by the maximum seal leakage rates for normal, off-normal, and accident conditions.
- Determination of doses to the whole body, thyroid, other critical organs, lens of the eye, and skin from inhalation and immersion exposures at the controlled area boundary (considering atmospheric dispersion factors $-\chi/Q$).

The application should specify maximum allowable “as tested” seal leakage rates as a Technical Specification, as discussed in SRP Chapter 13. Guidance on the calculations of the specific activity for each isotope in the cask and the maximum allowable helium seal leakage rates for normal, off-normal, and accident-level conditions can be found in NUREG/CR-6487, “Containment Analysis for Type B Packages Used to Transport Various Contents” (Anderson, 1996), and ANSI N14.5-1997. The minimum distance between the casks and the distance to the controlled area boundary is generally also a design criterion; however, 10 CFR 72.106(b) requires this distance to be at least 100m from the ISFSI.

5479 For the dose calculations, the NRC staff has accepted the use of either an adult breathing rate
5480 (BR) of $2.5 \times 10^{-4} \text{ m}^3/\text{s}$ ($8.8 \times 10^{-3} \text{ ft}^3/\text{s}$), as specified in Regulatory Guide (RG) 1.109, "Calculations
5481 of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of
5482 Evaluating Compliance with 10 CFR Part 50 Appendix I," or a worker breathing rate of 3.3×10^{-4}
5483 m^3/s ($1.2 \times 10^{-2} \text{ ft}^3/\text{s}$), as specified in the U.S. Environmental Protection Agency (EPA) Guidance
5484 Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose
5485 Conversion Factors for Inhalation, Submersion, and Ingestion" (EPA, 1988). The dose
5486 conversion factors (DCF) in EPA Guidance Report No. 11 for the whole body, critical organs,
5487 and thyroid doses from inhalation should be used in the calculation. The bounding DCFs from
5488 EPA Report No. 11 should be used for each isotope unless the applicant justifies an alternate
5489 value. The staff is not accepting weighting or normalization of the dose conversion factors. For
5490 each isotope, the committed effective dose equivalent (CEDE_i - for the internal whole body
5491 dose) or the committed dose equivalent (CDE_i - for the internal organ dose) are calculated as
5492 follows:

5493
5494 CEDE_i or CDE_i (in mrem per year for normal/off-normal or mrem per accident)
5495 = $Q_i * \text{DCF}_i * \chi/Q * \text{BR} * \text{Duration} * \text{conversion factor}$ (The conversion factor, if
5496 required, converts the input units into the desired form [CEDE_i or CDE_i] in mrem
5497 per year for normal/off-normal or mrem per accident).
5498

5499 For the contributions to the whole body, thyroid, critical organs, and skin doses from immersion
5500 (external) exposure, the DCFs in EPA Guidance Report No. 12, "External Exposure to
5501 Radionuclides in Air, Water, and Soil" (EPA, 1993), should be used. Again, the NRC staff is not
5502 accepting weighting or normalization of the dose conversion factors.
5503

5504 The deep dose equivalent (DDE_i - for the external whole body) and the shallow dose equivalent
5505 (SDE_i - for the skin dose) are calculated as follows:

5506
5507 DDE_i or SDE_i (in mrem per year for normal/off-normal or mrem per accident)
5508 = $Q_i * \text{DCF}_i * \chi/Q * \text{Duration} * \text{conversion factor}^2$
5509

5510 The total effective dose equivalent, TEDE = $\sum \text{CEDE}_i + \sum \text{DDE}_i$

5511 For a given organ, the total organ dose equivalent, TODE = $\sum \text{CDE}_i + \sum \text{DDE}_i$

5512 The total skin dose equivalent SDE = $\sum \text{SDE}_i$
5513

5514 Compliance with the lens dose equivalent (LDE) limit is achieved if the sum of the SDE and the
5515 TEDE does not exceed 0.15 Sv (15 rem). This approach is consistent with guidance in the
5516 Publication 26 of International Commission on Radiological Protection (ICRP), "Statement from
5517 the 1980 Meeting of the ICRP" (ICRP, 1980) and as specified in SRP Chapter 11, "Radiation
5518 Protection Evaluation."
5519

5520 In general, the staff evaluates analyses for normal, off-normal, and accident-level conditions.
5521

5522 5.5.4.1 Normal Conditions 5523

5524 For normal conditions, a bounding exposure duration assumes that an individual is present at
5525 the controlled area boundary for one full year (8,760 hours). An alternative exposure duration
5526 may be considered by the staff if the applicant provides justification.
5527

² The conversion factor, if required, converts the input units into the desired form, e.g., mrem/year.

5528 Because any potential release resulting from seal leakage would typically occur over a
5529 substantial period of time, the staff accepts (for applications for certificates) calculation of the
5530 atmospheric dispersion factors (χ/Q) according to RG 1.145, "Atmospheric Dispersion Models
5531 for Potential Accident Consequence Assessments at Nuclear Power Plants," assuming
5532 D-stability diffusion and a wind speed of 5 m/s (16 ft/s).
5533

5534 For the likely case of an ISFSI with multiple casks, the doses need to be assessed for a
5535 hypothetical array of casks during normal conditions according to Section 2.5.3.4,
5536 "Shielding/Confinement/Radiation Protection," of this SRP. Therefore, the staff anticipates that
5537 the resulting doses from a single cask will be a small fraction of the limits prescribed in
5538 10 CFR 72.104(a) to accommodate the array and the external direct dose.
5539

5540 Note: If the region between redundant, confinement boundary, mechanical seals is maintained
5541 at a pressure greater than the cask cavity, the monitoring system boundaries are tested to a
5542 leakage rate equal to the confinement boundary, the pressure is routinely checked, and the
5543 instrumentation is verified to be operable in accordance with a Technical Specification
5544 Surveillance Requirement, the NRC staff has accepted that no discernible leakage is credible.
5545 Therefore, calculations of dose to the whole body, thyroid, and critical organs at the controlled
5546 area boundary from atmospheric releases during normal conditions would not be required.
5547

5548 5.5.4.2 Off-Normal Conditions (Anticipated Occurrences)

5549

5550 For off-normal conditions, the bounding exposure duration and atmospheric dispersion factors
5551 (χ/Q) are the same as those discussed above for normal conditions.
5552

5553 To demonstrate compliance with 10 CFR 72.104(a), the staff accepts whole body, thyroid, and
5554 critical organ dose calculations for releases from a single cask. However, the dose contribution
5555 from cask leakage should also be a fraction of the limits specified in 10 CFR 72.104(a) since the
5556 doses from other radiation sources are added to this contribution.
5557

5558 5.5.4.3 Design-Basis Accident Conditions (Including Natural Phenomenon Events)

5559

5560 For accident-level conditions, the duration of the release is assumed to be 30 days (720 hours).
5561 A bounding exposure duration assumes that an individual is also present at the controlled area
5562 boundary for 30 days. This time period is the same as that used to demonstrate compliance for
5563 reactor facilities licensed per 10 CFR 50 and provides good defense in depth since recovery
5564 actions to limit releases are not expected to exceed 30 days.
5565

5566 For accident-level conditions, the staff has accepted calculation of the atmospheric dispersion
5567 factors (χ/Q) of RG 1.145 or RG 1.25, "Assumptions Used for Evaluating the Potential
5568 Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage
5569 Facility for Boiling and Pressurized Water Reactors," on the basis of F-stability diffusion and a
5570 wind speed of 1 m/s (3.3 ft/s).
5571

5572 To demonstrate compliance with 10 CFR 72.106(b), the staff accepts whole body, thyroid,
5573 critical organ, and skin dose calculations for releases of radionuclides from a single cask.
5574

5575 5.5.5 Supplemental Information

5576

5577 The reviewer should ensure that all supportive information or documentation has been provided
5578 or is readily available. This includes, but is not limited to, justification of assumptions or

5579 analytical procedures, test results, photographs, computer program descriptions, input and
5580 output, and applicable pages from referenced documents. Reviewers should request any
5581 additional information needed to complete the review.
5582

5583 **5.6 Evaluation Findings** 5584

5585 The reviewer should examine the 10 CFR Part 72 acceptance criteria and provide a summary
5586 statement for each. These statements should be similar to the following model:
5587

5588 F5.1 Chapter(s) _____ of the SAR describe(s) confinement structures, systems, and
5589 components (SSCs) important to safety in sufficient detail in to permit evaluation
5590 of their effectiveness.
5591

5592 F5.2 The design of the (cask designation) adequately protects the spent fuel cladding
5593 against degradation that might otherwise lead to gross ruptures. Chapter 4,
5594 "Thermal Evaluation" of the safety evaluation report (SER) discusses the relevant
5595 temperature considerations.
5596

5597 F5.3 The design of the (cask designation) provides redundant sealing of the
5598 confinement system closure joints by _____.
5599

5600 F5.4 The confinement system is monitored with a _____ monitoring system as
5601 discussed above (if applicable). No instrumentation is required to remain
5602 operational under accident conditions.
5603

5604 F5.5 The quantity of radioactive nuclides postulated to be released to the environment
5605 has been assessed as discussed above. In Chapter 11, "Radiation Protection
5606 Evaluation" of the SER, the dose from these releases will be added to the direct
5607 dose to show that the (cask designation) satisfies the regulatory requirements of
5608 10 CFR 72.104(a) and 10 CFR 72.106(b).
5609

5610 F5.6 The cask confinement system has been evaluated (by appropriate tests or by
5611 other means acceptable to the NRC) to demonstrate that it will reasonably
5612 maintain confinement of radioactive material under normal, off-normal, and
5613 credible accident conditions.
5614

5615 A summary statement similar to the following should be made:
5616

5617 "The staff concludes that the design of the confinement system of the (cask designation)
5618 is in compliance with 10 CFR Part 72 and that the applicable design and acceptance
5619 criteria have been satisfied. The evaluation of the confinement system design provides
5620 reasonable assurance that the (cask designation) will allow safe storage of spent fuel.
5621 This finding is reached on the basis of a review that considered the regulation itself,
5622 appropriate regulatory guides, applicable codes and standards, the applicant's analysis
5623 and the staff's confirmatory analysis, and accepted engineering practices."

6 SHIELDING EVALUATION

6.1 Objective

The shielding review is to evaluate if the proposed shielding features provide adequate protection against direct radiation from the dry storage system (DSS) contents. The shielding features should limit the dose to the operating staff and members of the public so that the dose remains within regulatory requirements during normal operating, off-normal, and design-basis accident (DBA) conditions. The review seeks to ensure that the shielding design is sufficient and reasonably capable of meeting the operational dose requirements of 10 CFR 72.104 and 72.106 in accordance with 10 CFR 72.236(d).

6.2 Areas of Review

This chapter of the DSS Standard Review Plan (SRP) provides guidance for use in evaluating the shielding features of the proposed cask system. As defined in Section 6.5, "Review Procedures," a comprehensive shielding evaluation may encompass the following areas of review:

Shielding Design Description

- Design Criteria
- Design Features

Radiation Source Definition

- Gamma Source
- Neutron Source

Shielding Design Specification

- Configuration of Shielding and Source
- Material Properties

Shielding Analyses

- Computer Codes
- Flux-to-Dose-Rate Conversion
- Dose Rates
- Confirmatory Analysis

Supplementary Information

- Shielding model description
- Computer model input and output

As prescribed in 10 CFR Part 72, the regulatory requirements for doses at and beyond the controlled area boundary include both direct radiation and radionuclides in effluents. An overall assessment of the compliance of the proposed DSS with these regulatory limits is contained in Chapter 11, "Radiation Protection Evaluation," of this SRP.

In order to ensure that the shielding design of the DSS meets the regulatory requirements as defined in 10 CFR Part 72, the applicant should also include information in the SAR regarding the technical specifications which are necessary for the DSS system to meet the dose rate limits at the controlled area boundary (See Chapter 13).

5675 In addition, the applicant should demonstrate that the system design, uses, and operating
 5676 procedures follow the ALARA Principle.

5677
 5678 **6.3 Regulatory Requirements**
 5679

5680 10 CFR Part 72 requires that spent fuel storage and handling systems be designed with
 5681 adequate shielding to provide sufficient radiation protection under normal, off-normal, and
 5682 accident-level conditions. The DSS application should describe the design principle and
 5683 functional features of the shielding structures, systems, and components (SSCs) important to
 5684 safety in sufficient detail to allow the U.S. Nuclear Regulatory Commission (NRC) staff to
 5685 thoroughly evaluate their effectiveness. It is the responsibility of the vendor and the facility
 5686 owner to analyze such SSCs with the objective of assessing the impact of direct radiation doses
 5687 and effluent releases to the environment on public health and safety. The reviewers should
 5688 verify the applicant's evaluations through review of the applicant's model, or confirmatory
 5689 analyses or independent modeling analysis. In addition, SSCs important to safety should be
 5690 designed to withstand the effects of both credible accidents and severe natural phenomena
 5691 without impairing their capability to perform their safety functions.

5692
 5693 This section presents a summary matrix of the portions of 10 CFR Part 72 that are relevant to
 5694 the review areas addressed by this chapter. The NRC staff reviewer should read the exact
 5695 referenced regulatory language. The NRC staff reviewer should verify the association of
 5696 regulatory requirements with the areas of review presented in the matrix to ensure that no
 5697 requirements are overlooked as a result of unique design features. Table 6-1 matches the
 5698 regulatory requirements associated with this chapter to the areas of review.
 5699

Table 6-1 Relationship of Regulations and Areas of Review				
Areas of Review	10 CFR Part 72 Regulations			
	72.104(a)	72.106(b)	72.122(b), (c)	72.236(d)
Shielding Design Description			•	•
Radiation Source Definition	•	•	•	•
Shielding Modeling Specifications	•	•	•	•
Shielding Analyses	•	•	•	•

5700
 5701 **6.4 Acceptance Criteria**
 5702

5703 Several technical and licensing factors should be considered during the shielding evaluation of
 5704 the proposed DSS. First, 10 CFR Part 72 states regulatory dose limits in terms of annual site-
 5705 specific doses for normal conditions and total absorbed dose from accident conditions.
 5706 Because the regulations do not specify cask dose rates (such as package dose rates in 10 CFR
 5707 Part 71), site-specific factors will have to be considered at each ISFSI when determining
 5708 compliance with the dose limits in 10 CFR 72.104 and 10 CFR 72.106. These site-specific
 5709 factors include the geometric arrangement of storage cask arrays, topography, distances to

5710 dose receptors, exposure times of dose receptors, actual spent nuclear fuel (SNF) loading
5711 patterns in each storage cask, and dose contributions from other surrounding fuel cycle
5712 facilities. Because all of these potential site-specific factors at various sites cannot be fully
5713 considered in the safety analysis report (SAR) for a DSS design, the regulations in
5714 10 CFR 72.236(d) only require that a demonstration of the shielding design is sufficient to
5715 satisfy 10 CFR 72.104 and 72.106. The general licensee DSS user is required by
5716 10 CFR 72.212 to consider its site-specific factors and ultimately demonstrate compliance with
5717 10 CFR 72.104. Therefore, the acceptance criteria for DSS shielding seeks to define standard
5718 analyses for single casks, and a generic array of casks, to demonstrate a sufficient shielding
5719 design. In addition, the acceptance criteria seeks to establish acceptable dose rate levels
5720 surrounding each DSS and acceptable dose calculation methodologies for further use by
5721 general licensees.

5722
5723 In general, the DSS shielding evaluation should provide reasonable assurance that the
5724 proposed design fulfills the following acceptance criteria:

5725

5726 1. The radiation shielding features of the proposed DSS are sufficient for it to meet
5727 the radiation dose requirements in 10 CFR 72.104(a) and 72.106(b). The
5728 applicant demonstrates this with:

5729

5730 a. A shielding analysis of the surrounding dose rates that contribute to
5731 occupational exposure and off-site doses at large distances (for a single
5732 storage and transfer cask with bounding fuel source terms at various cask
5733 locations), and

5734

5735 b. A shielding analysis of a single cask and a generic array of casks at large
5736 distances.

5737

5738 2. The shielding features of and the radiations emitted by the cask, in conjunction
5739 with its proposed operating procedures presented in Chapter 9, "Operating
5740 Procedures," of the SAR, are consistent with a well-established "as low as is
5741 reasonably achievable" (ALARA) program for activities in and around the storage
5742 site.

5743

5744 3. Radiation shielding and confinement features must be sufficient to meet the
5745 requirements in 10 CFR 72.106. 10 CFR 72.106(b) states: "Any individual
5746 located on or beyond the nearest boundary of the controlled area may not
5747 receive from any design basis accident the more limiting of a total effective dose
5748 equivalent [TEDE] of 0.05 Sv (5 rem), or the sum of the deep dose equivalent [DDE]
5749 and the committed dose equivalent [CDE] to any individual organ or tissue
5750 (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent
5751 [LDE] may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent [SDE]
5752 to skin or any extremity shall not exceed 0.5 Sv (50 rem)."

5753

5754 4. The proposed shielding features should demonstrate that the DSS is capable of
5755 meeting the regulatory requirements prescribed in 10 CFR Part 20.

5756

5757 The following sections provide additional guidance on acceptance criteria for each area of
5758 review for acceptability of SAR informational content and the details and method of evaluation of
5759 the proposed shielding features.

5760

5761 **6.4.1 Shielding Design Description**

5762
5763 6.4.1.1 Design Criteria

5764
5765 10 CFR 72.104 provides dose rate criteria for occupational exposure and for members of the
5766 public. Chapter 2, "Principal Design Criteria," of the SAR should specify the criteria that have
5767 been used as a basis for protection against direct radiation. Design criteria should include the
5768 identification of maximum dose rates and should also be specified for occupancy areas and
5769 correlated with occupancy duration and distance to radiation sources. An estimate of collective
5770 doses (person-rem per year) should be provided for each occupancy area under various
5771 operations (see Chapter 11, "Radiation Protection Evaluation" of this SRP).

5772
5773 The design should consider the ALARA principle. The reviewer should note that it is the
5774 responsibility of the general licensee using the DSS design to develop detailed procedures that
5775 incorporate the ALARA objectives of its site-specific radiation protection program. Further
5776 information on ALARA considerations is contained in the Radiation Protection Chapter.

5777
5778 6.4.1.2 Design Features

5779
5780 The SAR should describe the use of shielding to reduce direct radiation dose rates, and
5781 consider the following:

- 5782 • Self-shielding provided by the radioactive material being stored;
- 5783
- 5784 • Gamma and neutron shielding provided by the structural and nonstructural
- 5785 materials forming the walls and ends of the cask;
- 5786
- 5787 • Neutron capture provided by borated materials incorporated into the cask;
- 5788
- 5789 • Shielding provided by the temporary placement of water into the cask system
- 5790 during loading and unloading procedures; and
- 5791
- 5792 • Shielding provided by temporary placement of equipment and portable shields
- 5793 placed on and around the cask during loading and unloading procedures.
- 5794
- 5795

5796 **6.4.2 Radiation Source Definition**

5797
5798 The SAR should describe each type of contained radiation source used as a basis for shield
5799 design calculations. The physical and chemical form, source geometry, radionuclide content,
5800 and estimated radiation source strength in Becquerels or Curies and the bases for the
5801 estimation should be described in a manner suitable for use as input for shielding calculations.

5802
5803 The energy group structure from the source term calculation should correspond to that of the
5804 cross-section set of the shielding calculation. The computer methodology or database
5805 application used to compute source term strength should be specified.

5806
5807 6.4.2.1 Gamma Sources

5808
5809 A tabulation of radiological characteristics for each gamma-ray source type should be provided,
5810 including isotopic composition and photon yields by x-ray and gamma-ray energy group. The
5811 SAR should specify gamma source terms for both spent fuel and activated materials.

5812
5813 The SAR should describe the extent to which radioactivity may be induced by interactions
5814 involving neutrons originating in the stored materials. The SAR should also provide source term
5815 descriptions for induced radioactivity and the bases (assumptions and analytical methods) used
5816 for their estimation. Alternatively, the SAR may describe the bases for excluding induced
5817 radioactivity source terms.

5818
5819 6.4.2.2 Neutron Sources
5820

5821 The SAR should describe the neutron source terms and tabulate the neutron yield by energy
5822 group and the bases used to determine the source terms.

5823
5824 **6.4.3 Shielding Model Specification**
5825

5826 The application should include information in the SAR relative to materials and arrangements of
5827 all SSCs important to safety.

5828
5829 6.4.3.1 Configuration of Shielding and Source
5830

5831 The SAR should describe the geometric arrangement of shielding and include illustrations that
5832 identify the spatial relationships among sources, shielding, and design dose rate locations. The
5833 SAR should clearly indicate the physical dimensions of sources and shielding materials. The
5834 SAR should also identify penetrations, voids, or irregular geometries that provide potential paths
5835 for gamma or neutron streaming. These potential streaming paths should be clearly identifiable
5836 on submitted drawings. The SAR should describe design features used to minimize streaming
5837 through these penetrations.

5838
5839 The SAR should clearly state any differences between shielding features during normal or off-
5840 normal conditions and accident-level conditions.

5841
5842 6.4.3.2 Material Properties
5843

5844 The shielding reviewer should consult with the materials reviewer to assure that the SAR
5845 adequately describes the composition and geometry of the shielding materials.

5846
5847 **6.4.4 Shielding Analyses**
5848

5849 The SAR should describe the computer codes, version, computational models, data, and
5850 assumptions with their bases used in evaluating shielding effectiveness, and should provide
5851 dose rate estimates for areas of concern. The reviewer should perform confirmatory
5852 calculations, as necessary, to verify the results of the applicant's shielding analyses.

5853
5854 6.4.4.1 Computer Codes
5855

5856 The SAR should identify the computer codes and models used in evaluating shielding for each
5857 significant radiation source identified in Section 6.4.2, "Radiation Source Definition," and
5858 reference the appropriate documentation. For each computer code used, test problem solutions
5859 that demonstrate substantial similarity to solutions from other sources (hand calculations,
5860 published literature results, etc.) should be provided. A summary should be provided in the
5861 SAR that compares the test problem solutions in either graphical or numeric form. These
5862 solutions may be referenced and need not be submitted in the SAR if the references are widely

5863 available or have been previously submitted to the NRC for the same computer code and
5864 version.

5865
5866 The SAR should clearly present the data used as input for computational purposes and identify
5867 any differences between actual material properties or physical dimensions and those used in
5868 the analytical method (e.g., for simplifying the computational process). The applicant should
5869 defend any simplifications and assumptions by showing that the approach used will result in
5870 conservative (bounding) estimates.

5871
5872 The SAR should address calculational error in computer codes for both radiological and thermal
5873 source terms. Because validation data is relatively limited for burnups above 45 GWd/MTU
5874 (i.e., high burnup fuel), the SAR should numerically specify source term uncertainties for high
5875 burnup fuels.

5876
5877 The SAR should determine whether source term values with uncertainties should be applied to
5878 the shielding, thermal, and confinement analyses, instead of nominal calculated values. In this
5879 determination, the SAR may consider: (1) other conservative assumptions and design margins
5880 in the each respective analyses; (2) the maximum fuel assembly heat loads; (3) the maximum
5881 gamma and neutron dose rates; and (4) any measurable temperature or dose rate limitations
5882 proposed in the technical specifications.

5883
5884 A representative computer code input file used in the shielding computation performed for the
5885 DSS should be included in the SAR.

5886
5887 6.4.4.2 Flux-to-Dose-Rate Conversion

5888
5889 The basis for the flux-to-dose-rate conversion in the shielding analysis should be stated in the
5890 SAR, including conversions that are done by a computer code using its own data library. The
5891 SAR should include a table that shows the one to one conversion factor for each energy group
5892 of the cask specific source term spectrum. The NRC accepts flux-to-dose-rate conversion
5893 factors in American National Standards Institute/American Nuclear Society Standard 6.1.1-1977
5894 (ANSI/ANS-6.1.1-1977).

5895
5896 6.4.4.3 Dose Rates

5897
5898 The SAR evaluation of shielding effectiveness should include calculated or estimated dose rates
5899 in representative areas around the DSS. The dose rate calculations should account for such
5900 factors as a minimum distance no less than 100m (328 ft.), contributions from radionuclide
5901 releases, and other significant factors. These criteria are identified and evaluated in the
5902 radiation protection evaluation described in Chapter 11 of this SRP. The criteria below relate
5903 primarily to the completeness of information provided in the SAR.

5904
5905 The SAR should clearly indicate the physical locations on and around the casks for which dose
5906 rate calculations have been performed. These locations should include points on or in the
5907 immediate vicinity of cask surfaces where workers will perform operations during loading,
5908 retrieval, handling operations, and any projected maintenance and surveillance. For storage
5909 casks with internal labyrinthine air flow passages, the SAR should include dose rate estimates
5910 for the air inlets and air outlets using a computer code appropriate for streaming calculations.
5911 The SAR should identify points that have the highest calculated dose rates.

5912

5913 The SAR should include dose rate estimates for all onsite areas at which workers will be
5914 exposed to elevated dose rates. Dose rates within restricted areas should be calculated in
5915 enough detail to estimate doses received by workers performing ISFSI functions and off-site
5916 doses at large distances. This should be demonstrated with a standard dose-versus-distance
5917 curve or table for a single cask and for a generic DSS array.
5918

5919 The SAR should calculate the dose rate from the cask surface for off-normal events and DBA
5920 conditions to ensure compliance with 10 CFR 72.104(a) and 72.106(b), respectively. The
5921 computational model used for these calculations should be consistent with the expected
5922 condition of the cask after the event.
5923

5924 **6.5 Review Procedures**

5925
5926 Figure 6-1 presents an overview of the evaluation process and can be used as a guide to assist
5927 in coordinating with other review disciplines.
5928

5929 **6.5.1 Shielding Design Description**

5930 **6.5.1.1 Design Criteria (MEDIUM Priority)**

5931
5932
5933 Dose rates at the cask surface and in the vicinity of a loaded cask may vary during storage,
5934 transfer, and in-storage activities. While 10 CFR Part 72 establishes dose requirements only for
5935 ISFSI, it does not impose specific dose rate limits to the individual casks. Storage cask dose
5936 rates from 20 to 400 mrem/hour have been accepted in previous Part 72 evaluations.
5937 Acceptable dose rates depend on a number of factors such as the geometry of the storage
5938 array, the time workers will routinely spend in the storage array for activities like monitoring or
5939 maintenance, the proximity to other areas frequently occupied by workers, and the proximity to
5940 the controlled area boundary or other public access areas. The dose requirements are based
5941 on 10 CFR 20.1201 for the total expected exposure to workers during anticipated DSS
5942 operations, and 10 CFR 72.104 for members of the public who are located beyond the
5943 controlled area (i.e., assumed to be at the closest boundary but, in accordance with 10CFR
5944 72.106(b), at least 100m from the storage cask).

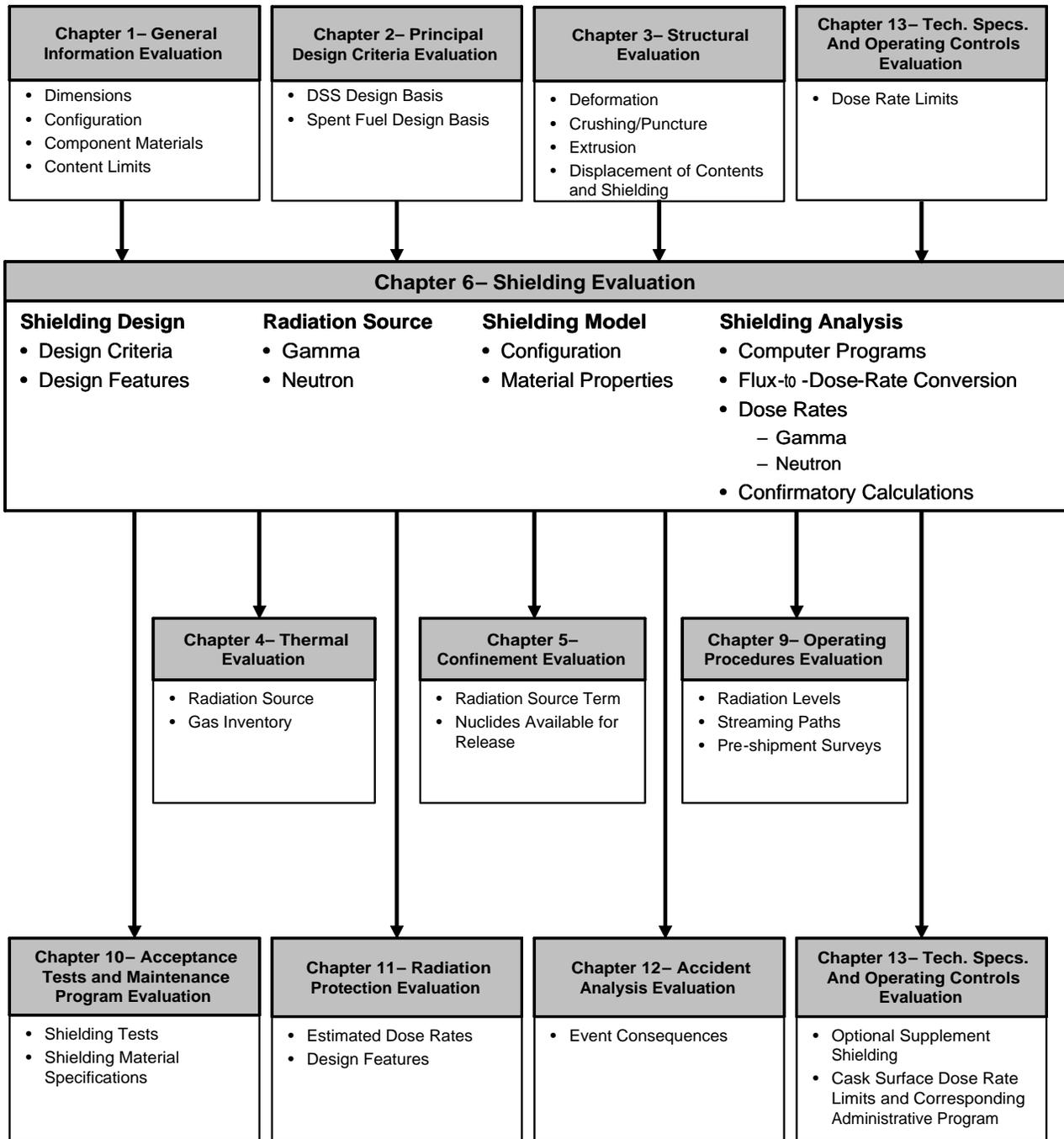


Figure 6-1 Overview of the Shielding Evaluation

5945
5946
5947

5948 The reviewer should coordinate with the SRP Chapter 2, "Principal Design Criteria Evaluation,"
5949 as well as review any additional shielding-related criteria. The reviewer should also refer to
5950 SRP Chapter 9, "Operating Procedures Evaluation," to consider any expected operating
5951 procedures that would require close proximity to the cask such as cask equipment that should
5952 be monitored or serviced frequently. However, the evaluated dose rates at the side of the same
5953 cask should be reviewed to ensure that the ALARA principles are either engineered into the
5954 design or evoked by specific operating procedures in Chapter 9, "Operating Procedures
5955 Evaluation" of the SAR.

5956
5957 6.5.1.2 Design Features (HIGH Priority)
5958

5959 The reviewer should be familiar with the general description of the DSS presented in Chapter 1,
5960 "General Description," of the SAR, as well as any additional information provided in Chapter 5,
5961 "Shielding Evaluation," of the SAR. All drawings, figures, and tables describing shielding
5962 features should be sufficiently detailed to allow the staff to perform an in-depth evaluation.
5963

5964 **6.5.2 Radiation Source Definition (HIGH Priority)**
5965

5966 Burnup, cooling time, initial uranium loading, and initial enrichment are parameters that affect
5967 the total source term of SNF. The reviewer should examine the description of the design-basis
5968 fuel in Chapter 2, "Principle Design Criteria" of the SAR to verify that the applicant had
5969 calculated the bounding source term. The review confirms that the applicant examined all fuel
5970 designs and burnup conditions for which the cask system is to be certified, to ensure that the
5971 bounding fuel type and values are used. Particular attention should be devoted to the combined
5972 effects of gamma and neutron source terms as a function of fuel burnup, cooling times, and
5973 enrichment. In many cases, there is no single specific enrichment-burnup combination and
5974 cooling time that bounds all potential cask loadings (see the analysis presented in NUREG/CR-
5975 6716). Variations in fuel assembly type play a secondary role for pressurized-water reactor
5976 (PWR) fuel. For boiling-water reactor (BWR) fuel, void fractions and channel sizes may affect
5977 the strengths of neutron and gamma sources. For a cask that contains spent fuel assemblies
5978 with irradiated burnable poison rod assemblies (BPRAs), a potential large effect is from
5979 activated component hardware (mainly activated cobalt in steel). Again, NUREG/CR-6716
5980 demonstrates for BPRAs designs containing stainless steel, the impact on the gamma dose rate
5981 can be large.
5982

5983 The design-basis radiation source term should be based on a saturation value for activation of
5984 cobalt impurities or on cobalt activation from a specified maximum burnup and minimum cool
5985 time. The reviewer should consider other activation products, as appropriate. These values
5986 should be bounded by those listed in the Technical Specifications.
5987

5988 6.5.2.1 Initial Enrichment
5989

5990 The specifications in Chapter 2, "Principle Design Criteria" of the SAR should indicate the
5991 maximum fuel enrichment used in the criticality analysis. For shielding evaluations, however,
5992 the neutron source term increases considerably with lower initial enrichment for a given burnup.
5993 As present in Section 3.4.1.2 of NUREG/CR-6716, as the initial enrichment decreases, the fuel
5994 is exposed to a larger neutron fluence to achieve the same burnup. The larger neutron fluence
5995 generates a larger actinide content resulting in a larger neutron source term and larger neutron
5996 and secondary gamma ray dose rate contribution illustrated in Figure 6-2 (reproduced from
5997 NUREG/CR-6716). Consequently, the SAR should specify the minimum initial enrichment as
5998 an operating control and limit for cask use, or justify the use of a neutron source term, in the

5999 shielding analysis, that specifically bounds the neutron sources for fuel assemblies to be placed
6000 in the cask. Because average initial enrichments typically increase with increasing burnup
6001 within the spent fuel population, the latter option may be used if the applicant uses low
6002 enrichments that bound the historical enrichments for fuels at the proposed burnups. However,
6003 the applicant and the staff should not attempt to establish specific source terms as operating
6004 controls and limits for cask use.

6005
6006 6.5.2.2 Computer Codes for Radiation Source Definition

6007
6008 The reviewer should verify that the applicant determines the source terms using a computer
6009 code, such as ORIGEN-S (e.g., as a SAS2 sequence of Oak Ridge National Laboratory's
6010 [ORNL] "SCALE" computer code package) that is well benchmarked and recognized and widely
6011 used by the industry. If a vendor proprietary code is used, the reviewer should check the code
6012 validation and verification records and procedures, preferably with sample testing problems.

6013
6014 The reviewer should ensure that appropriate descriptive information, including validation and
6015 verification status, and reference documentation has been provided. The reviewer also should
6016 determine if the computer code is suitable for determining the source terms and it has been
6017 correctly used. Area Of Applicability (AOA) is an important aspect. The reviewer should pay
6018 particular attention to AOA to verify if the application fall into the parameter ranges that the code
6019 is validated. The reviewer should determine whether the computer code is appropriately applied
6020 and the SAR includes verification that the chosen cross-section library is appropriate for the fuel
6021 specifications being considered. Many libraries are not appropriate for a burnup exceeding
6022 45,000 MWd/MTU because validation data is limited at high burnups.

6023
6024 The reviewer should verify that the applicant has adequately addressed calculational error and
6025 uncertainties of the computer codes used to determine source terms for the thermal, shielding,
6026 and confinement analyses. The reviewer should consider: (1) other conservative assumptions
6027 and design margins in the analyses; (2) the maximum fuel assembly heat loads; (3) the
6028 maximum gamma and neutron dose rates (including relative contributions to total); and (4) any
6029 measurable temperature or dose rate limits proposed for the technical specifications.

6030
6031 When reviewing the source term calculations, the reviewer should also consider the factor that
6032 nuclide importance changes in high burnup fuels as a function of burnup and validation data.
6033 The data for benchmarking the calculations and computer codes is limited at high burnups.
6034 Additional data and information on high burnup source term issues are provided in several
6035 NRC-sponsored studies (DeHart, 1996; Hermann, 1998; NUREG/CR-6700, NUREG/CR-6701,
6036 NUREG/CR-6798.

6037
6038 6.5.2.3 Gamma Source

6039
6040 The reviewer should verify that the applicant specified gamma source terms as a function of
6041 energy for both the spent fuel and activated hardware. If the energy group structure from the
6042 source term calculation differs from that of the cross-section set of the shielding calculation, the
6043 applicant may need to regroup the photons. Regrouping can be accomplished by using the
6044 nuclide activities from the source term calculation as input to a simple decay computer code
6045 with a variable group structure. Some applicants will convert from one structure to another
6046 using simple interpolation. In general, only gammas with energies from approximately 0.8 to 2.5
6047 MeV will contribute significantly to the dose rate through typical types of shielding; thus,
6048 regrouping outside this range is of a lesser importance. The reviewer should determine whether

6049 the source terms are specified per assembly, per total assemblies, or per metric ton, and ensure
6050 that the total source is correctly used in the shielding evaluation.

6051

6052 Determining source terms for fuel assembly hardware is generally not as straightforward as for
6053 the SNF due to cobalt contained in the fuel assembly hardware. The potential impact on the
6054 gamma dose rate could be very large during the cooling times in which ^{60}Co is the dominant
6055 gamma ray source (up to about 50 years) (NUREG/CR-6716). In particular, steel clad fuel
6056 potentially increases the cask dose rate by more than an order of magnitude over that from
6057 conventional Zircaloy clad fuel. The stainless steel in the BPRAs was assumed to have a
6058 nominal cobalt impurity level of 800 ppm, a value associated with older assembly designs. As
6059 presented in NUREG/CR-6716, the largest potential effect from assemblies residing in a cask
6060 that contains irradiated BPRAs is from activated component hardware (mainly activated cobalt
6061 in steel). For BPRAs designs containing stainless steel, the impact on the gamma dose rate can
6062 be large. The effort devoted to reviewing this calculation should be based on the contribution of
6063 these terms to the dose rates presented in the shielding evaluation. Also, it should be noted
6064 whether or not the cask is intended to contain special hardware, such as control assemblies or
6065 shrouds, and ensured that source terms from these components are included, if applicable. The
6066 reviewer should confer with the Chapter 2, "Principle Design Criteria Evaluation" review team to
6067 make this determination.

6068

6069 Depending on the cask design, neutron interactions may result in the production of high energy
6070 gammas near the cask surface. If this source term was not treated by the shielding analysis
6071 computer code, the reviewer should verify that it has been determined by other appropriate
6072 means.

6073

6074 As part of the source term determination, the reviewer should verify that the applicant calculates
6075 the quantities of certain nuclides (e.g., ^{85}Kr , ^3H , and ^{129}I) for use in analyzing doses from the
6076 release of radioactive material during postulated accidents in later sections of the SAR. These
6077 calculations are typically presented in Chapter 5, "Confinement," of the SAR with the shielding
6078 reviewer, in coordination with the confinement reviewer, verifying the information.

6079

6080 6.5.2.4 Neutron Source

6081

6082 The reviewer should verify that the neutron source term is expressed as a function of energy.
6083 The neutron source will generally result from both spontaneous fission and alpha-n reactions in
6084 the fuel. Depending on the method used to determine these source terms, the applicant may
6085 need to independently determine in the SAR, the energy group structure. This analysis is often
6086 accomplished by selecting the nuclide with the largest contribution to spontaneous fission (e.g.,
6087 ^{244}Cm) and using that spectrum for all neutrons, since the contribution from alpha-neutron
6088 reactions is generally small. For SNF with cooling times less than 5 years, the analysis should
6089 address the spectra of ^{242}Cm and ^{252}Cf .

6090

6091 The specification of a minimum initial enrichment may be a basis for defining the allowed
6092 contents. The reviewer should verify that the assumed minimum enrichments bounds all
6093 assemblies proposed for the casks in the application. Specific limits are needed for inclusion in
6094 the Certificate of Compliance (CoC). Lower enriched fuel, irradiated to the same burnup as
6095 higher enriched fuel, produces a higher neutron source. Consequently, the reviewer should
6096 verify that Chapter 13, "Technical Specifications and Operational Controls and Limits
6097 Evaluation," of the SAR specifies the minimum initial enrichment as an operating control and
6098 limit for cask use. Alternately, the applicant should specifically justify the use of a neutron
6099 source term, in the shielding analysis, that specifically bounds the neutron sources for fuel

6100 assemblies to be placed in the cask. An applicant may demonstrate that the assumed
6101 enrichment(s) bound the proposed fuel population except for possible outliers in the SNF
6102 population. This is acceptable if the SAR specifically requires each user to verify minimum
6103 enrichment with the Final SAR values, and if there specific dose rate limits in the technical
6104 specifications. The applicant and the staff should not attempt to establish specific source terms
6105 as the operating controls and limits for cask use.

6106
6107 **6.5.2.5 Other Parameters Affecting the Source Term**

6108
6109 The reviewer should ensure the SAR contains specific information concerning reactor
6110 operations that affects the source term. Several NRC technical reports (specifically,
6111 NUREG/CR-6716, but also NUREG/CR-6700, NUREG/CR-6701, and NUREG/CR-6798)
6112 discuss the potential affects of other parameters not typically included as a shielding technical
6113 specification (e.g., moderator soluble boron concentrations, maximum poison loading, minimum
6114 moderator density (for BWR fuels), and maximum specific power). For example, the net impact
6115 of moderator density on cask shielding is expected to be low for PWR fuels. However, the
6116 reviewer should be aware that the axial variation in moderator density in BWR cores can have a
6117 measurable effect on the axial dose rate profile of a BWR spent fuel assembly. The dose rate
6118 may increase near the top of the assemblies where the moderator density was the lowest. This
6119 is particularly important for neutron sources because reduced moderator density will harden
6120 neutron spectrum and hence induce more actinide production.

6121
6122 **6.5.3 Shielding Model Specification (HIGH Priority)**

6123
6124 The reviewer should verify that the applicant adequately describes the models that were used in
6125 the shielding evaluation for storage under normal, off-normal, and accident-level conditions. For
6126 example, if the cask has an external neutron shield, it should be determined whether the cask
6127 would be damaged by a tipover accident or degraded in a fire. Applicants should assume liquid,
6128 polyesters, or other resin neutron shields are not present after an accident, unless justification is
6129 made that they remain intact. The reviewer should confirm this analysis with the structural and
6130 thermal evaluation reviews of Chapter 3, "Structural Evaluation," and Chapter 4, "Thermal
6131 Evaluation," of the SAR, as appropriate. The reviewer should also confirm that the shielding
6132 assumptions made in dose rate calculations, for both occupational workers and the public, are
6133 consistent with the design criteria and design drawings.

6134
6135 **6.5.3.1 Configuration of the Shielding and Source**

6136
6137 The reviewer should examine the sketches or figures that indicate how the shielding design of
6138 the canister, storage overpack, and transfer cask is modeled. The reviewer should verify that
6139 the model dimensions and materials are consistent with those specified in the cask drawings
6140 presented in Chapter 1, "General Information Evaluation" of the SAR. Voids, streaming paths,
6141 and irregular geometries should be accounted for or otherwise treated in a conservative
6142 manner. In addition, the reviewer should verify that the applicant clearly states the differences,
6143 if any, between normal, off-normal, and accident-level conditions.

6144
6145 The reviewer should verify that the applicant properly modeled the source term locations for
6146 both spent fuel and structural support regions (i.e., fuel assembly hardware). In some cases,
6147 the fuel and basket materials may be homogenized within the fuel region to facilitate the
6148 shielding calculations. The reviewer should watch for cases when homogenization may not be
6149 appropriate. For example, homogenization should not be used in neutron dose calculations
6150 when significant neutron multiplication can result from moderated neutrons (i.e., when

6151 significant amounts of moderating materials are present such as when the cask is flooded).
6152 Similarly, homogenization should not be used in configurations where significant radiation
6153 streaming can occur between the basket components.

6154
6155 If the applicant has requested storage of damaged fuel assemblies, ensure that the applicant
6156 has adequately described the proposed damage assemblies. If the fuel assemblies are
6157 damaged to the extent that reconfiguration of the fuel into a geometry different from intact fuel
6158 assemblies can occur, ensure that the applicant provides appropriate close assessments for
6159 normal, off-normal and accident conditions.

6160
6161 SNF typically has a cosine shape burnup profile along its axial length. If axial peaking appears
6162 to be significant, the reviewer should verify that the applicant has appropriately accounted for
6163 the condition. Typically, fuel gamma source terms vary proportionally with axial burnup. Fuel
6164 neutron source terms vary exponentially by a power of 4.0 to 4.2 (NUREG/CR-6802,
6165 "Recommendations for Shielding Evaluations for Transport & Storage Packages") with axial
6166 burnup (NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup
6167 Credit Analyses"). In addition, the structural support regions (e.g., top and bottom end pieces
6168 and plenum) of the assembly should be correctly positioned relative to the SNF. These support
6169 regions may be individually homogenized with the basket materials when particle streaming
6170 through the gaps between basket components is not an issue. Generally, however, at least
6171 three source regions (i.e., fuel and top/bottom assembly hardware) are necessary. Some
6172 canisters may also employ fuel spacers to center the SNF inside the canister.

6173
6174 The reviewer should verify that the SAR shows or adequately describes the locations selected
6175 for the various dose calculations. The reviewer should ensure that these dose points are
6176 representative of all locations relevant to radiation protection issues. The reviewer should pay
6177 particular attention to dose rates from streaming paths to which occupational workers would be
6178 exposed (e.g., at vent/drain port covers, lid bolts, air vents, etc.). The shielding end points
6179 should be noted as well (such as lead in the cask wall in relation to the assembly hardware and
6180 use of fuel spacers to center the fuel). See Section 6.5.4.3 for additional information regarding
6181 the selection of locations for dose calculations.

6182 6183 6.5.3.2 Material Properties

6184
6185 The reviewer should verify that the SAR provides information concerning compositions and
6186 densities for all materials used in the calculation model. For nonstandard materials, such as
6187 neutron shields, Chapter 10 of the SAR, "Acceptance Tests and Maintenance Program
6188 Evakuation," should also reference the source of the data and indicate validation criteria. Many
6189 shielding computer codes allow the densities to be input directly in g/cm^3 . If input is required in
6190 atoms/barn-cm the reviewer should pay particular attention to the conversion.

6191
6192 The shielding reviewer should ensure that the elemental composition and density of shielding
6193 materials are conservatively adjusted in the shielding analyses to account for any degradation
6194 from aging, high temperature, accumulated radiation exposure, and manufacturing tolerances.
6195 The shielding reviewer should coordinate with the materials reviewer to obtain reasonable
6196 assurance that any degradation that may occur will not impact the safe performance of the
6197 shielding materials for the term proposed in the CoC application.

6198 6199 6.5.4 Shielding Analyses

6200

6201 6.5.4.1 Computer Codes (MEDIUM Priority)

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The reviewer should evaluate the computer codes or programs used for the shielding analysis. There are several recognized computer codes widely used for shielding analysis. These include computer codes that use Monte Carlo, deterministic transport, and 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. It is also important for the reviewer to assess whether the number of dimensions of the computer code being applied for the shielding analysis is appropriate for the dose rates being calculated. Typically, NRC staff does not accept the use of one-dimensional codes for calculations other than shielding designs with simple cylindrical geometries. At the least, a two-dimensional calculation is generally necessary. One-dimensional computer codes provide little information about off-axis locations and streaming paths that may be significant to determining occupational exposure. Even a two-dimensional calculation may not be adequate for determining any streaming paths if the modeled configuration is not properly established. These considerations in applying a particular computer code also apply to the computation of dose rates at the end of storage confinement casks. In some cases, the applicant will use the flux output from a deep-penetration shielding code as input to a large distance, skyshine code. The reviewer should verify that the use and interface of these codes are appropriate.

The reviewer should be aware that often adjoint calculations are performed by the applicant with computer codes to determine the neutron and gamma importance functions (units of mrem/hr/particle/s-cm). Multiplying the importance functions by a neutron and gamma source term-per-unit length yields dose rates on the surface of the cask. Using the neutron and gamma importance functions, the applicant could determine the minimum cooling time required to meet both a decay heat limit and any technical specification at the maximum dose rate limit on the side of the cask.

A valuable primer on shielding computer codes and analysis techniques has been published by ORNL (Broadhead, 1995).

The computer codes given below have been previously applied for DSS source and shielding analysis in applications reviewed by the NRC. However, their previous use does not constitute generic NRC approval and, as presented above, the reviewer is cautioned that these computer codes can produce errors when used incorrectly. Specifically, care should be taken to ensure any streaming paths in the cask are appropriately determined with multi-dimensional computer codes under normal, off-normal, and accident-level conditions. The reviewer should also determine that the SAR has specified design control measures that will ensure the quality of computer codes used for shield analysis.

The source of the computer codes given below vary from government sources, such as the Radiation Safety Information Computational Center³ (RSICC) and other U.S. Department of Energy (DOE) national laboratories, to commercial shielding computer codes. It is also important for the reviewer to be aware that due to proliferation and security concerns, access to specific U.S. government-sponsored computer code packages may be restricted and special permission may be required when granting their use to the applicant. The applicant should use the latest released computer code version that is valid for the particular computational platform used to perform the analysis. Computer codes are periodically updated to be compatible with

³ Radiation Safety Information Computational Center, Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, Tennessee, 37831-6362 and on the Internet at <<http://www-rsicc.ornl.gov>>.

6249 the latest operating system, correct errors found in prior versions, or incorporate updated
6250 methodologies.

6251
6252 The computer codes previously applied for DSS source and shielding analyses include:

- 6253
- 6254 • MicroSkyshine (air-scattering computer code);
- 6255
- 6256 • MORSE (Monte Carlo multigroup three-dimensional neutron and gamma
6257 transport computer code);
- 6258
- 6259 • MCBEND (Monte Carlo multigroup three-dimensional neutron and gamma
6260 transport computer code similar to MORSE developed by the United Kingdom
6261 (UK) National Radiation Protection Board (NRPB));
- 6262
- 6263 • MCNP (Monte Carlo n-particle transport computer code maintained by Los
6264 Alamos National Laboratory (LANL));
- 6265
- 6266 • RANKERN (three-dimensional point kernel gamma transport shielding computer
6267 code similar to QAD-CGGP);
- 6268
- 6269 • SCALE (a modular computer code system for performing standardized computer
6270 analyses for licensing evaluation maintained for the NRC by ORNL);
- 6271
- 6272 • SKYSHINE-II (air-scattering computer code); and
- 6273
- 6274 • STREAMING (computer code for calculation of attenuation of a gamma flux
6275 incident on a variety of shielding penetrations, such as ducts and voids).
- 6276

6277 Some other shielding computer code packages available through RSICC which have potential
6278 application to DSS sources include:

- 6279
- 6280 • DOORS3.2 (one-, two-, and three-dimensional discrete ordinates neutron/photon
6281 transport code system that includes ANISN for one-dimensional, DORT for two-
6282 dimensional, and TORT for three-dimensional analysis maintained by ORNL).
- 6283
- 6284 • DANTSYS (a code system maintained by the Los Alamos National Laboratory
6285 (LANL) that provides discrete ordinates solutions to the neutral particle transport
6286 equation that include ONEDANT for one-dimensional, TWODANT for two-
6287 dimensional, and THREEDANT for three-dimensional multigroup discrete-
6288 ordinate transport analysis.
- 6289

6290 Some of the above computer codes have been modified or improved to perform adjoint
6291 calculations. Examples of the computer codes with adjoint capability are as follows:

- 6292
- 6293 • DORT (part of the DOORS3.2 computer code package),
- 6294
- 6295 • A³MCNP (Automated Adjoint Accelerated MCNP),
- 6296
- 6297 • MCBEND.
- 6298

6299 The reviewer should verify that the SAR describes each of the numerical models of the
6300 computer codes used in the shielding evaluation. For each computer code used, the reviewer
6301 should ensure that an approved, validated, and verified version of the computer code is being
6302 applied by verifying that the following information has been provided in the SAR:

- 6303
- 6304 • The author, source, and dated version;
- 6305
- 6306 • A description of the numerical model applied in the computer code and the extent
6307 and limitation of its application; and
- 6308
- 6309 • The computer code solutions to a series of test problems, demonstrating
6310 substantial similarity to solutions obtained from hand calculations, analytical
6311 results published in the literature, acceptable experimental tests, a similar
6312 computer code, or benchmark problems.
- 6313

6314 The reviewer should examine the solution comparisons provided by the SAR and determine
6315 whether satisfactory agreement of computer and test solutions (or resolution of deviations) is
6316 evident. Ideally (though not a requirement), the computer code used for evaluation of shielded
6317 storage containers should have been validated with actual dose rate measurements from similar
6318 or prototypical SNF or high-level waste storage systems.

6319

6320 6.5.4.2 Flux-to-Dose-Rate Conversion (MEDIUM Priority)

6321

6322 The shielding analysis computer code may perform flux-to-dose-rate conversion using its own
6323 data library. For the conversions, the NRC accepts the use of ANSI/ANS 6.1.1-1977. While this
6324 standard was revised in 1991, the NRC has not adopted the methodology given in ANSI/ANS
6325 6.1.1-1991 principally for two reasons. First, the 10 CFR Part 20 radiation protection
6326 requirements are based on fluence-to-dose conversions that are essentially the same as those
6327 defined by ANSI/ANS 6.1.1-1977, and are conservative relative to those of
6328 ANSI/ANS 6.1.1-1991. Second, neutron dose rates determined on the basis of conversions
6329 performed according to ANSI/ANS 6.1.1-1991 may be significantly lower than those determined
6330 on the basis of 10 CFR Part 20 or ANSI/ANS 6.1.1-1977.

6331

6332 6.5.4.3 Dose Rates (MEDIUM Priority)

6333

6334 On the basis of experience, comparison to similar systems, or scoping calculations, the reviewer
6335 should make an initial assessment of whether the dose rates appear reasonable and whether
6336 their variation with location is consistent with the geometry and shielding characteristics of the
6337 cask system. The following guidance pertains to the selection of points at which the dose rates
6338 should be calculated.

6339

6340 For normal and off-normal conditions, the applicant should indicate the dose rate at all locations
6341 accessible to occupational personnel during cask loading, transport to the ISFSI, and
6342 maintenance and surveillance operations. Generally, these locations include points at or near
6343 various cask components and in the immediate vicinity of the cask. Example of locations
6344 include vent areas, trunnion areas, peak side of the cask, peak top of the cask, the canister-gap
6345 region, and the bottom of the transfer cask. The applicant should also calculate the dose rates
6346 at a distance of 1m from these locations because they typically contribute to occupational
6347 exposures.

6348

6349 The application for a cask design is required by 10 CFR 72.236(d) to demonstrate that the
6350 shielding and confinement features of the cask are sufficient to meet the requirements in
6351 10 CFR 72.104 for any real individual. The real individual is an individual at or beyond the
6352 controlled area, and the dose to any real individual must not exceed the limits specified in
6353 10 CFR 72.104 from both the storage facility and other surrounding fuel cycle activities. For
6354 example, a real individual may be anyone living, working, or recreating close to the facility for a
6355 significant portion of the year.

6356
6357 However, for approval of a cask design, the applicant should evaluate the shielding and
6358 confinement features of a single cask and a theoretical array of casks, assuming design-basis
6359 source terms and full-time occupancy. The applicant should also provide analyses to facilitate
6360 future site-specific evaluations for each general ISFSI licensee. The single cask analysis should
6361 identify the minimum distance that is required to meet the dose rates in 10 CFR 72.104. Past
6362 applications have shown this distance to be typically within 200m (656 ft.) of the cask. The
6363 applicant should include a dose rate versus distance curve for a single cask to facilitate a site-
6364 specific evaluation for general licensees. To satisfy 10 CFR 72.106(b), dose evaluations should
6365 be determined at a minimum of 100m (328 ft.) distance to the closest boundary of the controlled
6366 area. However, the applicant may use a longer distance, provided that the longer distance is
6367 made a condition of use.

6368
6369 The applicant should also include a dose rate-versus-distance curve for a theoretical cask array.
6370 The theoretical cask array should consist of at least 20 storage casks (typically in a 2x10 array),
6371 and should account for shadowing effect among casks.

6372
6373 It is important to note that the general ISFSI licensee is permitted to use additional engineering
6374 features, such as berms, to mitigate doses to real individuals near the site. If such features are
6375 used in the cask SAR to show compliance with the regulations, they should be included in the
6376 cask conditions of use. In addition, the SAR should determine the degree to which the normal
6377 condition dose rates could change for the identified off-normal conditions.

6378
6379 As required by 10 CFR 72.212(b)(2)(i)(C), a general licensee must perform a written evaluation
6380 to demonstrate that the dose limits in 10 CFR 72.104 are met. An evaluation similar to that for
6381 site-specific ISFSI should be performed. The licensee may use information provided in the cask
6382 SAR, as well as site specific information to perform the evaluation. Evaluations performed by
6383 the general ISFSI licensee are not reviewed for approval by NRC; however, they are subject to
6384 NRC inspection and must be recorded and maintained by the general licensee.

6385
6386 The general licensee should establish measures in the radiological protection program,
6387 environmental monitoring program, and/or operating procedures to identify and reevaluate
6388 potential increases in exposure to the real individuals. Compliance with the dose limits in
6389 10 CFR 72.104 will be verified by the environmental monitoring program with direct radiation
6390 measurements and/or effluent measurements, as appropriate.

6391
6392 The reviewer should review the technical specifications of Chapter 13 of this SRP to ensure
6393 appropriate requirements are addressed in the technical specifications of the cask. In addition,
6394 the degree to which the normal condition dose rates could change for the identified off-normal
6395 conditions should be verified. The need for additional calculations should be indicated in the
6396 Safety Evaluation Report (SER) and in the conditions set forth in the CoC.

6397
6398 If the above dose rate criteria are satisfied, NRC accepts that the direct-dose regulatory
6399 requirements can also be satisfied, although the exact details needed to comply with these

6400 limitations will vary from ISFSI site to site. Therefore, the SAR needs to address such
6401 requirements only in general terms. Detailed calculations need not be presented if Chapter 13
6402 of the SAR, "Technical Specifications and Operational Controls and Limits Evaluation," assigns
6403 ultimate compliance responsibilities to the ISFSI site licensee.

6404
6405 In addition, the applicant should calculate the dose rate at 1m (3.3 ft.) from the cask surface for
6406 accident-level conditions to assist in demonstrating the design is sufficient to meet the
6407 requirements of 10 CFR 72.106. The model used for these calculations should be consistent
6408 with the expected condition of the cask after an accident or natural phenomenon event.

6409
6410 The potential reconfiguration of damaged fuel within the damaged-fuel can, if applicable, must
6411 be analyzed to demonstrate that the cask/fuel meet the dose limits of normal and design basis
6412 events of storage. The shielding analysis should assume a worst case or bounding
6413 configuration of the canned fuel.

6414
6415 6.5.4.4 Confirmatory Calculations (HIGH Priority)

6416
6417 The reviewer should independently evaluate the dose rates in the vicinity of the cask for normal,
6418 off-normal, and accident-level conditions. In determining the level of effort appropriate for these
6419 calculations, the reviewer should consider the following factors:

- 6420 • the degree of sophistication in the SAR analysis;
- 6421
- 6422 • a comparison of SAR dose rates with those of similar casks that have previously
6423 been reviewed, if applicable;
- 6424
- 6425 • the typical variation in dose rates expected between different computer codes
6426 and cross-section sets;
- 6427
- 6428 • the fact that actual dose rates will be monitored and limited by the requirements
6429 of 10 CFR Part 20;
- 6430
- 6431 • the restrictions that can be placed on ISFSI operations affecting measured dose
6432 rates, as documented in SER Section 12, the site-specific license, or the CoC
6433 understanding that current technical specification guidance does not include
6434 dose rates restrictions for a general dry cask storage license;
- 6435
- 6436 • the applicant's experience in using the methods and computer codes in previous
6437 submittals;
- 6438
- 6439 • the use of new, or previously reviewed, computational methods or computer
6440 codes; and,
- 6441
- 6442 • the inclusion in the design of any significant departures from previous cask
6443 system designs (e.g., unusual shield geometry, new types of materials, or
6444 different source terms).
- 6445
- 6446

6447 At a minimum, the review should include examination of the applicant's input to the computer
6448 code used for the shielding analysis. The reviewer should verify use of proper dimensions,

6449 material properties, and an appropriate cross-section set. In addition, the reviewer should
6450 independently evaluate the use of gamma and neutron source terms.

6451
6452 If a more detailed review is required (e.g., a new and not previously reviewed shielding
6453 computer code), the reviewer should independently confirm the dose rates to ensure that the
6454 SAR results are reasonable and conservative. As previously noted, the use of a simple
6455 computer code for neutron calculations often does not provide results with sufficient accuracy
6456 and confidence. An extensive and more detailed evaluation may be necessary if large
6457 uncertainties are suspected. To the degree possible, the use of a different shielding computer
6458 code with a different analytical technique and cross-section set from that of the SAR analysis
6459 will usually provide a more independent evaluation.

6460
6461 A good reference regarding the treatment of uncertainty in thick-shielded cask analyses is the
6462 Electric Power Research Institute's "Evaluation of Shielding Analysis Methods in Spent Fuel
6463 Cask Environments," published in 1995 (Broadhead, 1995).

6464 6465 **6.5.5 Supplemental Information**

6466
6467 Supplemental information can include copies of applicable references (especially if a reference
6468 is not generally available to the reviewer), computer code descriptions, input and output files,
6469 and any other information that the applicant deems necessary. Likewise, the reviewer should
6470 request any additional information needed to complete the review process.

6471 6472 **6.6 Evaluation Findings**

6473
6474 The reviewer should review the 10 CFR Part 72 acceptance criteria and provide a summary
6475 statement for each. These statements should be similar to the following model:

6476
6477 F6.1 Section(s) _____ of the SAR describe(s) shielding structures, systems, and
6478 components (SSCs) important to safety in sufficient detail to allow evaluation of
6479 their effectiveness. The reviewer should cite specific drawings that are used to
6480 define the SSCs for shielding.

6481
6482 F6.2 Section(s) _____ of the SAR demonstrate the radiation shielding features are
6483 sufficient to meet the radiation protection requirements of 10 CFR Part 20,
6484 10 CFR 72.104 and 10 CFR 72.106.

6485
6486 F6.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR
6487 Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site
6488 licensee. The [cask designation] shielding features are designed to assist in
6489 meeting these requirements.

6490
6491 A summary statement similar to the following should be made:

6492
6493 "The staff concludes that the design of the shielding system of the [cask designation] is
6494 in compliance with 10 CFR Part 72 and that the applicable design and acceptance
6495 criteria have been satisfied. The evaluation of the shielding system design provides
6496 reasonable assurance that the [cask designation] will allow safe storage of spent fuel in
6497 accordance with 10 CFR 72.236(d). This finding is reached on the basis of a review that
6498 considered the regulation itself, appropriate regulatory guides, applicable codes and
6499 standards, and accepted engineering practices.

7 CRITICALITY EVALUATION

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7.1 Review Objective

The criticality review and evaluation ensures that spent nuclear fuel (SNF) to be placed into the dry storage system (DSS) remains subcritical under normal, off-normal, and accident conditions involving handling, packaging, transfer, and storage. The criticality review is designed to fulfill the strategic outcome of no inadvertent criticality events, part of the strategic goal of safety described in the agency's strategic plan (NUREG-1614).

7.2 Areas of Review

This portion of the DSS review evaluates the criticality design and analysis related to SNF handling, packaging, transfer, and storage procedures for normal, off-normal, and accident conditions. Consequently, this chapter of the DSS Standard Review Plan (SRP) provides guidance for use in conducting a comprehensive criticality evaluation that may encompass any or all of the following areas of review:

Criticality Design Criteria and Features

Fuel Specification

- Non-Fuel Hardware
- Fuel Condition

Model Specification

- Configuration
- Material Properties

Criticality Analysis

- Computer Codes
- Multiplication Factor
- Benchmark Comparisons

Burnup Credit

- Limits for the Licensing Basis
- Code Validation
- Licensing-Basis Model Assumptions
- Loading Curve
- Assigned Burnup Loading Value
- Estimate of Additional Reactivity Margin

Supplemental Information

7.3 Regulatory Requirements

SNF storage systems must be designed to remain subcritical unless at least two unlikely independent events occur. Moreover, the SNF cask must be designed to remain subcritical under all credible conditions. Regulations specific to nuclear criticality safety of the cask system are specified below. Normal and accident conditions to be considered are also identified in U.S. Code of Federal Regulations (CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, "Energy" (10 CFR

6551 Part 72). The reviewer should read the exact regulatory language. Table 7-1 matches the
 6552 relevant regulatory requirements associated with this chapter to the areas of review.
 6553

Table 7-1 Relationship of Regulations and Areas of Review			
Areas of Review	10 CFR Part 72 Regulations		
	72.124	72.236(a)	72.236(b), (c), (g), (h), (m),
Criticality Design Criteria and Features	•	•	•
Fuel Specification	•	•	
Model Specification	•	•	•
Criticality Analysis	•	•	•
Burnup Credit	•	•	

6554
 6555 **7.4 Acceptance Criteria**
 6556

6557 In general, the DSS criticality evaluation seeks to ensure that a subcritical condition is
 6558 maintained for the given design by fulfilling the following acceptance criteria:
 6559

- 6560 • The effective neutron multiplication factor, k_{eff} , including all biases and
 6561 uncertainties at a 95-percent confidence level, should not exceed 0.95 under all
 6562 credible normal, off-normal, and accident-level conditions.
 6563
- 6564 • At least two unlikely, independent, and concurrent or sequential changes to the
 6565 conditions essential to criticality safety, under normal, off-normal, and accident-
 6566 level conditions would need to occur before an accidental criticality is deemed to
 6567 be possible (i.e., double contingency principle).
 6568
- 6569 • When practicable, criticality safety of the design should be established on the
 6570 basis of favorable geometry, permanently fixed neutron-absorbing materials
 6571 (poisons), or both. Where solid neutron-absorbing materials are used, the design
 6572 should provide for a positive means to verify their continued efficacy during the
 6573 storage period. The neutron-absorbing materials' continued efficacy may be
 6574 confirmed by a demonstration or analysis before use, showing that significant
 6575 degradation of these materials cannot occur over the life of the facility.
 6576
- 6577 • Criticality safety of the cask system should not rely on credit for more than 75
 6578 percent of the boron in fixed neutron absorbers when subject to standard
 6579 acceptance tests. For greater credit allowance, special, comprehensive
 6580 fabrication tests capable of verifying the presence and uniformity of the neutron
 6581 absorber are needed.
 6582

6583 **7.5 Review Procedures**

6584

6585 The interrelationship of the criticality evaluation review with other disciplines is shown in Figure
6586 7-1. The figure shows that this review draws upon information from the general information
6587 section as well as information reviewed or developed for the design criteria, structural, and
6588 operating procedures evaluations. Information collected or developed during the review of this
6589 chapter is useful in the evaluation of the materials, operating procedures, acceptance tests and
6590 maintenance program, accident analysis, and technical specifications and operating controls for
6591 the DSS.

6592

6593 The reviewer should examine the criticality design features and criteria in SAR Chapter 1,
6594 "General Information," and SAR Chapter 2, "Principal Design Criteria," in addition to SAR
6595 Chapter 7, "Criticality Evaluation," for any additional details concerning criticality design features
6596 and criteria. The reviewer should assess the bounding specifications for the SNF and assure
6597 consistency with the models used by the applicant in the criticality analyses. The reviewer
6598 should verify that criticality safety considerations under normal, off-normal, and accident-level
6599 conditions are addressed by the applicant and that the cask system design complies with
6600 10 CFR Part 72. In addition, the reviewer should verify that the criticality calculations determine
6601 the highest k_{eff} that might occur for all loading states under normal, off-normal, and accident
6602 conditions involving handling, packaging, transfer, and storage. To the extent practicable, the
6603 use of independent methods to perform any k_{eff} calculations by the reviewer should be pursued
6604 to evaluate the applicant's design.

6605

6606 **7.5.1 Criticality Design Criteria and Features (HIGH Priority)**

6607

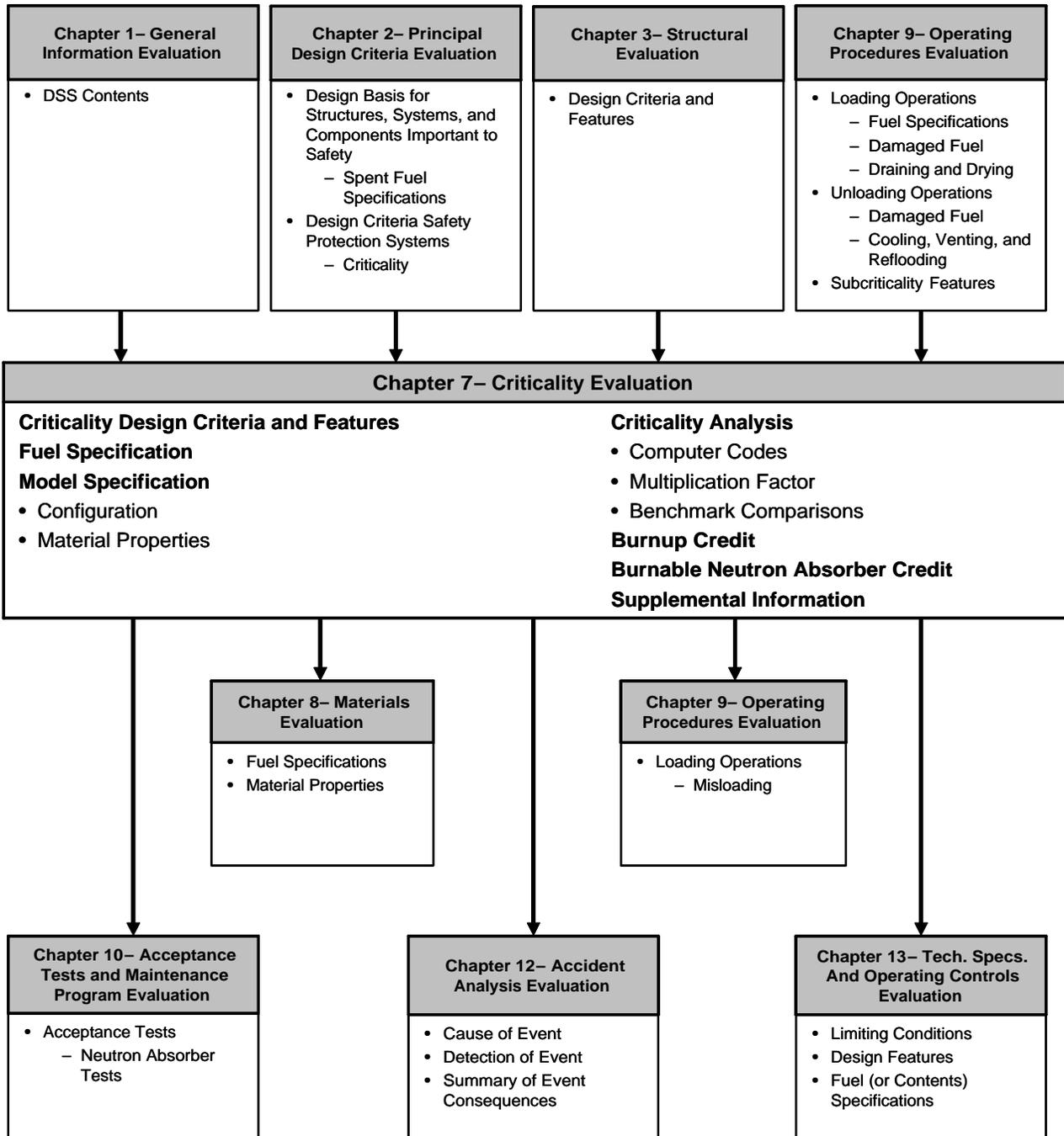
6608 The reviewer should examine the principal criticality design criteria presented in SAR Chapter 2
6609 as well as any related details provided in SAR Chapter 7, "Criticality Evaluation". The general
6610 cask description presented in SAR Chapter 1 should be examined for any relevant information.
6611 The information in Chapter 7 of the SAR should be verified to be consistent with the information
6612 in SAR Chapters 1 and 2. The reviewer should verify that all descriptions, drawings, figures,
6613 and tables are sufficiently detailed to support an in-depth staff evaluation.

6614

6615 The criticality design of the cask relies on the general dimensions of the cask components and
6616 the spacing of the fuel assemblies. The criticality design also often relies on neutron poisons.
6617 These may be in the form of fixed poisons in the basket structure, which may be used together
6618 with flux traps, and/or soluble poisons in the water of the SNF pool. During loading and
6619 unloading operations, NRC staff accepts the use of borated water as a means of criticality
6620 control if the applicant specifies a minimum boron content and strict controls are established to
6621 ensure that the minimum required boron concentration is maintained. This condition in turn
6622 becomes an operating control and limit in SAR Chapter 13, and in the Technical Specification
6623 (TS). The SER should also discuss these operating controls. Other design features significant
6624 to the criticality design, such as important basket dimensions that control the spacing of the fuel
6625 assemblies should also be included in the TS. These dimensions may be a minimum pitch for
6626 the basket cells or a minimum flux trap width.

6627

6628 If borated water is used for criticality control during loading and unloading operations,
6629 administrative controls and/or design features should be implemented to ensure that accidental
6630 flooding with unborated water cannot occur, or the criticality evaluation should consider
6631 accidental flooding with unborated water. If the cask is also intended for transport, borated
6632 water should not be relied upon for criticality control. Borated water and any other liquids are
6633 not acceptable as a means of criticality control for a cask in dry storage.



6634
6635

Figure 7-1 Overview of Criticality Evaluation

6636 This includes use of any credit in the criticality analysis for the presence of a liquid that may
6637 provide neutron shielding (and is external to the fuel basket); however, its presence and most
6638 reactive density should be assumed if it increases k_{eff} . Also, if more than one certified or
6639 licensed basket design of the same supplier could fit in the cask; the type of basket to be used
6640 with the cask should be stamped in a location on the cask system in a way that allows for easy
6641 identification of the basket. Thus, a licensee using the cask system will be able to easily verify
6642 the appropriateness of the fuel contents to be loaded in the basket.

6644 **7.5.2 Fuel Specification (HIGH Priority)**

6645
6646 The reviewer should examine the specifications for the ranges or types of SNF that will be
6647 stored in the cask as presented in SAR Chapters 1, "General Information Evaluation" and 2,
6648 "Principle Design Criteria Evaluation" as well as any related information provided in SAR
6649 Chapter 7, "Criticality Evaluation". The SNF specifications given in Chapter 7 of the SAR should
6650 be consistent with, or bound, the specifications given in SAR Chapters 1 and 2 and in the TS.
6651 The reviewer should also, keeping in mind that some specifications are more important than
6652 others, identify the specifications that are keys to criticality safety and verify that these are
6653 appropriately captured in the TS. NUREG-1745 provides a listing of some fuel specifications
6654 that may be keys to maintaining the system subcritical.

6655
6656 Of primary interest is the type of fuel assemblies and maximum fuel enrichment that should be
6657 specified and used in the criticality calculations. Some boiling-water reactors (BWR) use
6658 multiple fuel pin enrichments, in which case the criticality calculations should use the maximum
6659 fuel pin enrichment present. Depending upon the fuel design, an applicant may propose use of
6660 assembly averaged or lattice averaged enrichments. This may be acceptable if the applicant
6661 can demonstrate that the applicant's averaging technique is technically defensible and, for the
6662 criticality calculation, produces realistic or conservative results. Because of the natural uranium
6663 blankets present in many BWR designs, use of an assembly-averaged enrichment that includes
6664 the blankets is not normally considered appropriate or conservative for BWR fuel.

6665
6666 Another parameter of interest is the fuel density assumed in the analysis. The value of the fuel
6667 density used in the calculations should be justified to be realistic or conservative.

6668
6669 Although the burnup of the fuel affects its reactivity, many criticality analyses have assumed the
6670 cask to be loaded with fresh fuel (the fresh fuel assumption). Alternatively, the NRC staff has
6671 provided guidance for limited burnup credit for intact fuel. This guidance is currently limited to
6672 burnup credit available from actinide compositions associated with UO₂ fuel of 5.0 wt percent or
6673 less enrichment that has been irradiated in a PWR to an assembly-average burnup value not
6674 exceeding 50 GWD/MTU and cooled out-of-reactor for a time period between 1 and 40 years.
6675 Guidance regarding the review of a criticality analysis that involves burnup credit is provided in
6676 Section 7.5.5. Specifications for the fuel that will be stored in the cask, including those
6677 important for burnup credit, if applicable, should be included in Chapter 13, "Technical
6678 Specifications and Operational Controls and Limits Evaluation" of both the SAR and SER, with
6679 those specifications determined to be key to criticality safety also explicitly listed in the
6680 Technical Specifications.

6681
6682 For analyses that use the fresh fuel assumption, inadvertent loading of the cask with
6683 unirradiated fuel is not a major concern. However, inadvertent loading of the cask with
6684 unirradiated fuel is a major concern for casks that rely on criticality analyses that use burnup
6685 credit. Therefore, detailed loading procedures for these casks will need to include steps to
6686 prevent misloading of unirradiated fuel. Regardless of which analysis is used, detailed loading

6687 procedures may need to include steps to prevent misloading if fuel exceeding the design basis
6688 for the DSS is present in the pool at the time of loading.
6689

6690 Because casks are typically designed to store many types and configurations of fuel
6691 assemblies, the applicant should demonstrate that criticality requirements are satisfied for the
6692 most reactive case. A determination of which fuel is bounding in a criticality analysis depends
6693 on many factors and usually requires examination of several types of fuel assemblies and
6694 compositions. The design-basis fuel has often been the Westinghouse 17x17 optimized fuel
6695 assembly (OFA); however, this will not be the case for all cask designs because of cask-specific
6696 effects on reactivity. Therefore, the applicant should demonstrate and reviewers should verify
6697 that the fuel assembly used as the design basis is the most reactive for the specific cask design.
6698 Chapter 1, "General Information Evaluation" of the SAR and Chapter 13, "Technical
6699 Specifications and Operation Controls and Limits Evaluation" of the SER should either clearly
6700 indicate the design-basis assemblies or reference the SAR chapter in which they are identified.
6701

6702 7.5.2.1 Non-Fuel Hardware 6703

6704 Some fuel assemblies may also have non-fuel components that are positioned or operated
6705 within the envelope of the fuel assembly during reactor operation that an applicant may seek to
6706 store with the assemblies in the cask. These items include PWR control assemblies such as
6707 Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Burnable
6708 Poison Rod Assemblies (BPRAs) and Axial Power Shaping Rods (APSRs). Applicants may
6709 also seek approval of storage of fuel assemblies with other items that extend into an assembly's
6710 active fuel region, such as stainless steel rod inserts used to displace water in PWR assembly
6711 guide tube dashpots. For applications that propose to load assemblies containing non-fuel
6712 hardware, ensure that the analysis considers the effects of both inclusion and neglect of non-
6713 fuel hardware on system reactivity. If the application relies on the presence of the non-fuel
6714 hardware to meet the subcritical criterion, verify that the non-fuel hardware will remain in place
6715 under all normal and design basis conditions.
6716

6717 Generally, staff does not allow reliance on, or credit for, fuel-related burnable neutron
6718 absorbers. This restriction includes residual neutron-absorbing material remaining in the non-
6719 fuel hardware loaded with an assembly. However, credit for any negative reactivity for this latter
6720 absorbing material may be accepted if: (1) the remaining absorbing material content is
6721 established through physical measurement, where a sufficient margin of safety is included
6722 commensurate with the uncertainty in the method of measurement, (2) the axial distribution of
6723 the poison depletion is adequately determined with appropriate margin for uncertainties, and
6724 (3) adequate structural integrity and placement of the non-fuel hardware under accident
6725 conditions is demonstrated. Ensure that the fuel specifications, described in Chapter 13,
6726 "Technical Specifications and Operation Controls and Limits Evaluation" of both the SAR and
6727 SER, include the important details about the non-fuel hardware to be stored with the fuel
6728 assemblies and the associated residual neutron absorbing material, with those details key to
6729 criticality safety included in the TS, as appropriate. Also, verify that operating procedures are
6730 established that ensure that non-fuel hardware loaded with assemblies meets the approved
6731 specifications as well as remains in position.
6732

6733 7.5.2.2 Fuel Condition 6734

6735 Determine if the applicant has included any specifications regarding the fuel condition. To date,
6736 a number of applications have requested approval for storage of fuel that is damaged as well as
6737 intact, or undamaged. The reviewer should consult the most current staff guidance for detailed

6738 descriptions regarding what constitutes damaged, undamaged and intact fuel (e.g.,
6739 Section 8.5.4.3 and Attachment 8-3 of this SRP or more recent guidance). This guidance gives
6740 the applicant the latitude to define fuel with defects (such as missing rods but not loose rods or
6741 debris) as undamaged fuel as long as the fuel can meet all the fuel specific or system related
6742 functions. For purposes of the criticality function, undamaged fuel is fuel that: (1) is in the form
6743 of an assembly, (2) has structural and material properties such that the assembly can withstand
6744 normal and design basis events while maintaining its geometric configuration and (3) has had
6745 any damaged or missing fuel rods replaced with solid dummy rods that displace an equal
6746 amount of water as the original rods. Fuel that cannot meet these criteria is considered to be
6747 damaged. However, a fuel assembly with missing fuel rods may be considered undamaged fuel
6748 if analyses are performed that show the criterion for subcriticality will be met with the fuel rods
6749 missing.

6750
6751 A fuel assembly that is classified as damaged must be placed in a damaged fuel canister, or in
6752 an acceptable alternative, for loading into the cask. For a cask that is also intended for
6753 transport, it must be kept in mind that the more severe conditions of transport may require
6754 re-analysis of assemblies classified as undamaged under storage-only conditions prior to
6755 transport. Specifications concerning the condition of the fuel to be stored in the cask and the
6756 loading of damaged fuel, as applicable, should be included in Chapter 13, "Technical
6757 Specifications and Operation Controls and Limits Evaluation" of both the SAR and SER and in
6758 the Certificate of Compliance (in the TS).

6759
6760 The reviewer should verify that the criticality analysis addresses the conditions of the fuel to be
6761 stored in the cask system. Analyses for cask systems designed to store damaged fuel should
6762 bound the configuration of the damaged fuel assemblies under all credible normal and design
6763 basis conditions. For example, some analyses have performed calculations that model the
6764 damaged fuel as arrays of bare fuel rods (i.e., the cladding is assumed to be completely
6765 removed) having an optimized rod pitch.

6766
6767 **7.5.3 Model Specification (HIGH Priority)**
6768

6769 Manufacturing and fabrication tolerances should be specified, and the reviewer should verify
6770 that the applicant used the most reactive combination of tolerances, within the ranges of their
6771 acceptable values, in the cask system model.

6772
6773 **7.5.3.1 Configuration**
6774

6775 The reviewer should verify that the model used in the criticality evaluation is adequately
6776 described for normal, off-normal, and accident conditions. The reviewer should also coordinate
6777 with the structural, materials, and thermal reviewers to understand any damage that could result
6778 from accident or natural phenomena events.

6779
6780 The reviewer should examine the sketches or figures of the model used for criticality
6781 calculations. The reviewer should verify that the dimensions and materials of the model are
6782 consistent with the engineering drawings. Differences between the actual cask configuration
6783 and the models should be identified, and the models should be shown to be conservative.
6784 Substitution of end sections and support structures of the fuel with ordinary water is a common
6785 and usually conservative practice in criticality analysis. However, substitution with borated
6786 water is typically not conservative. Any such substitutions should be justified.

6787

6788 Tolerances for poison material dimensions and/or concentrations should be defined, and the
6789 most reactive conditions should be used in the criticality analysis. In addition, the analysis
6790 should identify all important design conditions and then address these conditions for potential
6791 variations during normal, off-normal, and accident-level conditions.

6792
6793 The reviewer should verify that the applicant has considered deviations from nominal design
6794 configurations. The evaluation of k_{eff} should not be limited to a model in which all of the fuel
6795 bundles are neatly centered in each basket compartment with the center line of the basket
6796 coincident with the center line of the cask. For example, a cask with steel confinement and lead
6797 shielding may have a higher k_{eff} when the basket and fuel assemblies are positioned as close as
6798 possible to the lead. However, in some designs, the most reactive configuration may be when
6799 all fuel assemblies are shifted toward the center of the basket.

6800
6801 In addition to a fully flooded cask, the SAR should address configurations in which the cask is
6802 filled with partial density water or is partially filled with water (borated, if applicable) and the
6803 remainder of the cask is filled with steam consisting of ordinary water at partial density. These
6804 configurations are considered to be possible during loading and unloading operations. The SAR
6805 should also consider the possibility of preferential or uneven flooding within the cask, if such a
6806 scenario is credible for the given cask design (e.g., because of blockage in small flow or drain
6807 paths). In particular, the reviewer should watch for situations where there is water in the fuel
6808 regions but not in the flux traps, if applicable. Cask designs for which this type of flooding is
6809 credible are generally unacceptable. The SAR should also consider flooding in the fuel rod
6810 pellet-to-clad gap regions with unborated water. Above all, the analysis must demonstrate that
6811 the cask remains subcritical for all credible conditions of moderation.

6812
6813 The reviewer should examine whether the applicant has prepared a heterogeneous model of
6814 each fuel rod or has homogenized the entire fuel assembly. With current computational
6815 capabilities, homogenization is now an uncommon practice and should not be used.

6816 7.5.3.2 Material Properties

6817
6818
6819 The reviewer should verify that the compositions and densities are provided for all materials
6820 used in the calculational model. The applicant should also cite, in the SAR Chapter 8,
6821 "Materials Evaluation", the source of all materials data, particularly the data for fuel and poison
6822 materials. In coordination with the materials reviewer, the criticality reviewer should determine
6823 the acceptability of the sources of data that are important to the criticality safety function of the
6824 cask. The criticality reviewer should, in coordination with the materials reviewer, ensure that the
6825 applicant addressed the validation of the poison concentration in the acceptance testing
6826 discussion in SAR Chapter 10, "Acceptance Tests and Maintenance Program Evaluation."
6827 Criticality computer codes generally will allow the densities to be input directly in units of g/cm^3
6828 or units of atoms/barn-cm. In either case, the reviewer should pay attention to the final value
6829 used directly by the code. Also, the reviewer should confirm that the analysis does not take
6830 credit for more than the minimum amount of neutron absorber verified by the acceptance
6831 testing, subject to the criteria in Section 7.4.

6832
6833 Among other specifications, 10 CFR Part 72 requires that a positive means to verify the
6834 continued efficacy of solid neutron-absorbing materials should be provided when these
6835 materials are used. The criticality reviewer should verify that the neutron flux from the irradiated
6836 fuel results in a negligible depletion of poison material over the storage period. In coordination
6837 with the materials and structural reviewers, the criticality reviewer should ensure that the
6838 applicant demonstrates that the required acceptance testing of the poisons during fabrication

6839 (specified in SAR Chapter10, "Acceptance Tests and Maintenance Program Evaluation") has
6840 been satisfactorily specified, and by analysis or demonstration, the applicant has shown the
6841 poison material's durability and resistance to degradation during the storage period.
6842
6843
6844

6845 The neutron flux used for this analysis should be the maximum that may be produced by
6846 feasible loadings of irradiated or unirradiated fuel. The reviewer should coordinate review of the
6847 applicant's acceptance testing and assessment of the poison material's durability with the
6848 materials reviewer to verify that the applicant provides a valid and accurate demonstration of the
6849 absorber material's continued efficacy. Consideration should be given to the effects of physical
6850 and chemical actions as well as irradiation (gamma and neutron). There may be other ways to
6851 provide positive means of verifying the neutron absorber's continued efficacy. For applications
6852 that propose an alternative method, the reviewer should verify that the proposed method is
6853 reasonable (considering any effects on meeting containment, shielding, or other system design
6854 criteria) and valid and accurate in demonstrating the absorber's continued efficacy.
6855

6856 **7.5.4 Criticality Analysis (Priority as indicated)**

6857 7.5.4.1 Computer Codes

6858 (MEDIUM Priority) Both Monte Carlo and deterministic computer codes may be used for
6859 criticality calculations. Monte Carlo computer codes are better suited to three-dimensional
6860 geometry and, therefore, are more widely used to evaluate spent fuel cask designs. The most
6861 frequently used Monte Carlo codes are SCALE/KENO (ORNL, 2005), MCNP (LANL, 2003), and
6862 MONK (ANSWERS, 2001). All three codes permit the use of either multigroup or continuous
6863 cross sections. The reviewer should determine that the applicant has used a computer code
6864 that is appropriate for the particular application and has used that code correctly.
6865
6866
6867

6868 (LOW Priority) The reviewer should determine whether the applicant has chosen an acceptable
6869 set of cross sections. Cross sections may be distributed with the criticality computer codes or
6870 developed independently from another source. The applicant should provide or reference the
6871 source of cross-section data. For user-generated cross sections, the applicant should specify
6872 the method used to obtain the actual data employed in the criticality analysis. For multigroup
6873 calculations, the neutron flux spectrum used to construct the group cross sections should be
6874 similar to that of the cask. If a multigroup treatment is used, the reviewer should ensure the
6875 applicant has appropriately considered the neutron spectrum of the cask. In addition to
6876 selecting a cross-section set collapsed with an appropriate flux spectrum, a more detailed
6877 processing of the energy-group cross sections is required to properly account for resonance
6878 absorption and self-shielding. The use of multigroup KENO as part of the CSAS sequences in
6879 SCALE will directly enable appropriate cross-section processing. Some cross-section sets
6880 include data for fissile and fertile nuclides (based on a potential scattering cross section, s_p) that
6881 can be input by the user. If the applicant has used a stand-alone version of KENO, the reviewer
6882 should ensure that potential scattering has been properly considered. Furthermore, information
6883 has been published concerning problems with some cross-section libraries once commonly
6884 distributed with SCALE/KENO. One library, the "working-format" library, was used for
6885 calculations of the code manual's sample problems but is not intended for criticality calculations
6886 of actual systems (IN 91-26, 1991). Another library, the SCALE 123-group library, has
6887 demonstrated inadequacies for non-thermalized, highly enriched systems (NUREG/CR-6328,
6888 1995).
6889

6890 MEDIUM Priority) The reviewer should pay particular attention to the proper selection of
6891 scattering cross section data for important compounds that may be in the system. Use of a free
6892 atom cross section for nuclides in a compound may not adequately account for the scattering
6893 effects of atoms bound in molecules and lattices. This misrepresentation can cause the
6894 underprediction of k_{eff} , particularly in the case of a well moderated system where energetic up
6895 scattering plays a significant role in the neutronics of the system.
6896

6897 (MEDIUM Priority) For analyses of a cask model with separate regions of water and steam, the
6898 use of a multigroup cross-section set raises additional concerns. The reviewer should verify
6899 that the applicant has addressed the differences in the flux spectra in the two regions. If the
6900 results of these calculations indicate that k_{eff} is close to 0.95, additional independent calculations
6901 using a different code and/or cross-section library (a library derived from a different cross-
6902 section database if possible and appropriate) may be helpful. The reviewer should also closely
6903 examine the applicant's benchmark analysis to verify the applicability of the critical experiments
6904 considered.
6905

6906 7.5.4.2 Multiplication Factor

6907
6908 (MEDIUM Priority) The reviewer should examine the results and discussion of the k_{eff}
6909 calculations for the storage cask. The reviewer should verify that the calculations determine the
6910 highest k_{eff} that might occur during all operational states under normal, off-normal and accident
6911 conditions. Sensitivity parametric analyses may be used to provide the required demonstration
6912 that the highest k_{eff} with a confidence level of 95 percent has been determined. Variations in the
6913 results caused by differences in the models and sensitivity analyses should be explained and
6914 found to be reasonable.
6915

6916 (MEDIUM Priority) For Monte Carlo calculations, the reviewer should assess if the number of
6917 neutron histories and convergence criteria are appropriate. As the number of neutron histories
6918 increases, the mean value for k_{eff} should approach a fixed value, and the standard deviation
6919 associated with each mean value should decrease. Depending on the code used by the
6920 applicant, a number of diagnostic calculations are generally available to demonstrate adequate
6921 convergence and statistical variation. For deterministic codes, a convergence limit is often
6922 prescribed in the input. The selection of a proper convergence limit and the achievement of this
6923 limit should be described and demonstrated in either the SAR or supporting criticality
6924 calculations. When burnup credit is included in the criticality analysis, the reviewer needs to be
6925 sure that proper neutron sampling and convergence have been achieved because the flux will
6926 be concentrated in the low burned ends of the fuel assemblies.
6927

6928 (HIGH Priority) Because of the importance and complexity of the criticality evaluation,
6929 independent calculations should be performed to ensure that the most reactive conditions have
6930 been addressed, the reported k_{eff} is conservative and the applicant has appropriately modeled
6931 the storage cask geometry and materials. In deciding the level of effort necessary to perform
6932 independent confirmatory calculations, the reviewer should consider the following factors:
6933 (1) the calculation method (computer code) used by the applicant, (2) uniqueness and
6934 complexity of the design and analysis, (3) the degree of conservatism in the applicant's
6935 assumptions and analyses, and (4) the extent of the margin between the calculated result and
6936 the acceptance criterion of $k_{\text{eff}} \leq 0.95$. As with any design and review, a small margin below the
6937 acceptance criterion and/or a small degree of conservatism may necessitate a more extensive
6938 staff analysis.
6939

6940 (HIGH Priority) The reviewer should develop a model that is independent of the applicant's
6941 model. If the reported k_{eff} for the most reactive case is substantially lower than the acceptance
6942 criterion of 0.95, a simple model known to produce very bounding results may be all that is
6943 necessary for the independent calculations.

6944

6945 (HIGH Priority) If possible and appropriate, the reviewer should perform the independent
6946 calculations with a computer code different from that used by the applicant. Likewise, use of a
6947 different cross-section set, derived from a different cross-section database where possible and
6948 appropriate (e.g., ENDF/B, JEF, JENDL, UKNDL, etc.), can provide a more independent
6949 confirmation. The continuous energy (CE) cross sections created for use with KENO in the
6950 SCALE code system are generated by the AMPX processing code rather than the more widely
6951 used NJOY code. Even though some cross section libraries may not have fully independent
6952 data bases because they are all derived from ENDF/B data, the CE library in SCALE still can
6953 provide some level of independence and is useful for checking computations performed with
6954 libraries which were generated by using NJOY. The reviewer should describe the staff's
6955 independent analysis and the analysis general results and conclusions in the SER.

6956

6957 (HIGH Priority) Although a k_{eff} of 0.95 or lower meets the acceptance criterion, the reviewer
6958 should watch for design features or content specifications where small changes could result in
6959 large changes in the value of k_{eff} . When the value of k_{eff} is highly sensitive to system
6960 parameters that could vary, the acceptable k_{eff} limit may need to be reduced below 0.95. When
6961 establishing a k_{eff} limit below 0.95, the reviewer should consider the degree of sensitivity to
6962 system parameter changes and the likelihood and extent of potential parameter variations.

6963

6964 7.5.4.3 Benchmark Comparisons (HIGH Priority)

6965

6966 Computer codes for criticality calculations should be benchmarked against critical experiments.
6967 A thorough comparison provides justification for the validity of the computer code, its use for a
6968 specific hardware configuration, the neutron cross sections used in the analysis, and
6969 consistency in modeling by the analyst. Ultimately the benchmarking process establishes a bias
6970 and uncertainty for the particular application of the code (using the benchmark results for
6971 calculations performed by another analyst does not address this last issue) . The calculated k_{eff}
6972 of the cask should then be adjusted to include the appropriate biases and uncertainties from the
6973 benchmark calculations.

6974

6975 The reviewer should examine the general description of the benchmark comparisons. This
6976 examination includes verifying that the analysis of the experiments used the same computer
6977 code, computer system, cross-section data, modeling methods, and code options that were
6978 used to calculate the cask system k_{eff} values.

6979

6980 The reviewer should also closely examine the applicant's benchmark analysis to determine
6981 whether the benchmark experiments are relevant to the actual cask design. No critical
6982 benchmark experiment will precisely match the fissile material, moderation, neutron poisoning,
6983 and configuration in the actual cask. However, the applicant can perform a proper benchmark
6984 analysis by selecting experiments that adequately represent cask and fuel features and
6985 parameters that are important to reactivity. Key features and parameters that should be
6986 considered in selecting appropriate critical experiments include the type of fuel, enrichment,
6987 hydrogen-to-uranium (H/U) ratio (dependent largely on rod diameter and pitch), reflector
6988 material, neutron energy spectrum, and poisoning material and placement. The applicant
6989 should justify, and the reviewer should verify, the suitability of the critical experiments chosen to
6990 benchmark the criticality code and calculations. Techniques such as the sensitivity/uncertainty

6991 method developed by Oak Ridge National Laboratory (ORNL/TM-2005/39, 2005) can be helpful
6992 when assessing the applicability of the critical experiments used to benchmark the design
6993 analysis. UCID-21830, the “International Handbook on Evaluated Criticality Safety Benchmark
6994 Experiments,” (NEA, 9/2003) and NUREG/CR-6361 provide information on benchmark
6995 experiments that may apply to the cask being analyzed.
6996

6997 The reviewer needs to assess whether the applicant analyzed a sufficient number of appropriate
6998 benchmark experiments and how the results of these benchmark calculations have been
6999 converted to a bias for the cask calculations. Simply averaging the biases from a number of
7000 benchmark calculations typically is not sufficient, such as when one benchmark yields results
7001 that are significantly different from the others, the number of experiments is limited, or
7002 benchmarks that over-predict k_{eff} are included. In addition, benchmark comparisons should be
7003 checked for bias trends with respect to parameter variations (such as pitch-to-rod-diameter
7004 ratio, assembly separation, reflector material, neutron absorber material, etc.). A Lawrence
7005 Livermore National Laboratory (LLNL) report by W.R. Lloyd (LLNL, 1/1990) and
7006 NUREG/CR-6361 provide some guidance, but other methods, when adequately explained, have
7007 also been considered appropriate.
7008

7009 For Monte Carlo codes, the statistical uncertainties of both benchmark and cask calculations
7010 also need to be addressed. The uncertainties should be applied to at least the 95-percent
7011 confidence level. As a general rule, if the acceptability of the result depends on these rather
7012 small differences, the reviewer should question the overall degree of conservatism of the
7013 calculations. Considering the current availability of computer resources, a sufficient number of
7014 neutron histories can readily be used so that the treatment of these uncertainties should not
7015 significantly affect the results.
7016

7017 The reviewer should verify that only biases that increase k_{eff} have been applied. For example, if
7018 the benchmark calculation for a critical experiment results in a neutron multiplication that is
7019 greater than unity, it should not be used in a manner that would reduce the k_{eff} calculated for the
7020 cask. Only corrections that increase k_{eff} should be applied to preserve conservatism.
7021

7022 The reviewer may have already performed a number of benchmark calculations applicable to
7023 storage casks and may have a reasonable estimation of the bias to be applied to the
7024 independent calculation of the cask. If such is not the case, or if the acceptability depends on
7025 small bias differences, the reviewer again needs to determine whether sufficient conservatism
7026 has been applied to the calculations.
7027

7028 **7.5.5 Burnup Credit (HIGH Priority)**

7029

7030 Unirradiated reactor fuel has a well-specified nuclide composition that provides a straightforward
7031 and bounding approach to the criticality safety analysis of transport and storage casks. As the
7032 fuel is irradiated in the reactor, the nuclide composition changes and, ignoring the presence of
7033 burnable poisons, this composition change will cause the reactivity of the fuel to decrease.
7034 Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from
7035 irradiation is typically termed burnup credit.
7036

7037 The following guidance (Sections 7.5.5.1 to 7.5.5.6) is applicable to fuel that is classified as
7038 undamaged fuel and is expected, based upon engineering evaluations, to remain undamaged
7039 under off-normal and accident-level conditions. If burnup credit is requested for mildly damaged
7040 fuel (basically undamaged and not debris; i.e., damaged fuel that has the same geometric form
7041 and structural integrity as undamaged fuel), this guidance may be applied, as appropriate, while

7042 accounting for uncertainties that can be associated with the damaged fuel, to establish an
7043 isotopic inventory and assumed fuel configuration for normal and accident conditions that bound
7044 the uncertainties.

7045
7046 7.5.5.1 Limits for the Licensing Basis
7047

7048 Available data supports allowance for burnup credit where the licensing safety analysis is based
7049 on actinide compositions associated with UO_2 fuel of an initial enrichment up to 5.0 wt. percent
7050 in Uranium-235 irradiated in a PWR to an assembly-average burnup value up to 50 GWd/MTU
7051 and cooled out-of-reactor for a time period between 1 and 40 years. The range of available
7052 measured assay data for irradiated UO_2 fuel indicates that an extension of the licensing basis
7053 beyond 5.0 wt. percent enrichment is not warranted. Even within this range of parameters, the
7054 reviewer needs to exercise care in assessing whether the analytical methods and assumptions
7055 used are appropriate, especially near the ends of the range. Use of actinide compositions
7056 associated with burnup values or cooling times outside these specifications should be
7057 accompanied by the measurement data and/or justified extrapolation techniques necessary to
7058 adequately extend the isotopic validation and quantify or bound the bias and uncertainty.

7059
7060 7.5.5.2 Code Validation
7061

7062 The computational methodologies used for predicting the actinide compositions and determining
7063 the k_{eff} should be properly validated. Bias and uncertainties associated with predicting the
7064 actinide compositions should be determined from benchmarks of applicable fuel assay
7065 measurements. Bias and uncertainties associated with the calculation of k_{eff} should be derived
7066 from benchmark experiments that closely represent the important features of the cask design
7067 and SNF contents. The particular set of nuclides used to determine the k_{eff} value should be
7068 limited to that established in the validation process. The licensing-basis safety analysis should
7069 utilize bias and uncertainty values that can be justified as bounding based on the quantity and
7070 quality of the experimental data. Particular consideration should be given to bias uncertainties
7071 arising from the lack of critical experiments that are highly prototypical of SNF in a cask.

7072
7073 7.5.5.3 Licensing-Basis Model Assumptions
7074

7075 The actinide compositions used to determine a value of k_{eff} for the licensing safety basis (as
7076 described in SRP Section 7.5.5.1) should be calculated using fuel design and in-reactor
7077 operating parameter values that appropriately encompass the range of design and operating
7078 conditions for the proposed contents. The calculation of the k_{eff} value should be performed
7079 using cask models, appropriate analysis assumptions, and code inputs that allow adequate
7080 representation of the physics. The following should be of particular concern:

- 7081
- 7082 • The need to account for and effectively model the axial and horizontal variation of
7083 the burnup within a SNF assembly (e.g., the selection of the axial burnup profiles,
7084 number of axial material zones, etc.).
 - 7085
 - 7086 • The need to consider the potential for increased reactivity due to the presence of
7087 burnable absorbers or control rods (fully or partially inserted) during irradiation.
 - 7088

7089 The axial burnup profile database in RSICC's Data Package DLC-201 (Cacciapouti, 1997)
7090 provides a source of realistic, representative data that can be used for establishing a profile to
7091 use in the licensing-basis safety analysis. However, care should be taken to select a profile that

7092 will encompass the range of potential k_{eff} values for the proposed contents, particularly near the
7093 upper end of the ranges described in SRP Section 7.5.5.1.

7094
7095 A licensing-basis modeling assumption where the assemblies are exposed during irradiation to
7096 the maximum (neutron absorber) loading of burnable poison rods for the maximum burnup is an
7097 appropriate analysis assumption that encompasses all assemblies that may or may not have
7098 been exposed to burnable absorbers (Sanders and Wagner, 2002b). Such an assumption in
7099 the licensing-basis safety analysis should also encompass the impact of exposure to fully
7100 inserted or partially inserted control rods in typical domestic PWR operations (Sanders, 2002a).
7101 Assemblies that are exposed to atypical insertions of poison rods (e.g., full control rod, CEA,
7102 RCCA, or APSR insertion for one full cycle of reactor operation) or that include integral poison
7103 rods (e.g., integral fuel burnable absorbers - IFBAs) or poisons coated on pellets should not be
7104 loaded unless the safety analysis explicitly considers such operational conditions. If the
7105 assumption on burnable poison rod exposure is less than the maximum for which overall burnup
7106 credit is requested, then a justification commensurate with the selected value should be
7107 provided (e.g., the lower the value, the greater the need to support the assumption with
7108 available data and/or indicate how administrative controls will prevent a misload of an assembly
7109 exposed beyond the assumed value).

7110 7111 7.5.5.4 Loading Curve

7112
7113 A loading curve shows the minimum allowable assembly burnup as a function of initial
7114 enrichment; fuel assemblies with greater burnup values may be loaded in the cask. Separate
7115 loading curves should be established for each set of applicable licensing conditions. For
7116 example, a separate loading curve should be provided for each minimum cooling time to be
7117 considered in the cask loading. The applicability of the loading curve to bound various fuel
7118 types or burnable absorber loadings should be justified. To limit the opportunity for misloading,
7119 only one loading curve should be used for each cask loading.

7120 7121 7.5.5.5 Assigned Burnup Loading Value

7122
7123 Administrative procedures should be established to ensure that the cask will be loaded with fuel
7124 that is within the specifications of the approved contents. The administrative procedures should
7125 include a measurement that confirms the reactor record for each assembly. Procedures that
7126 confirm the reactor records using measurement of a sampling of the fuel assemblies will be
7127 considered if a database of measured data is provided to justify the adequacy of the procedure
7128 in comparison to procedures that measure each assembly.

7129
7130 The measurement technique may be calibrated to the reactor records for a representative set of
7131 assemblies. For confirmation of assembly reactor burnup record(s), the measurement should
7132 provide agreement within a 95-percent confidence interval based on the measurement
7133 uncertainty. The assembly burnup value to be used for loading acceptance (termed the
7134 assigned burnup loading value) should be the confirmed reactor record value as adjusted by
7135 reducing the record value by a combination of the uncertainties in the record value and the
7136 measurement.

7137 7138 7.5.5.6 Estimate of Additional Reactivity Margin

7139
7140 The available experimental database relevant to use of burnup credit in the safety analysis of a
7141 PWR cask is not as extensive as the database available to support licensing with the
7142 unirradiated fuel assumption. The process of assuring that appropriate values and conditions

7143 have been applied in the safety analysis is also more difficult. For example, there may be
7144 uncertainties that are not directly evaluated in the modeling or validation processes for actinide-
7145 only burnup credit (e.g., k_{eff} validation uncertainties caused by a lack of critical experiments with
7146 either actinide compositions that match those in SNF or material distributions that represent the
7147 more reactive ends of SNF). Also, there may be potential uncertainties in the models that
7148 calculate the licensing-basis actinide inventories (e.g., caused by any outlier assemblies with
7149 higher-than-modeled reactivity such as may be caused by prolonged use of control rod insertion
7150 during irradiation, axial profiles not encompassed by the data in RSICC's Data Package
7151 DLC-201 [Cacciapouti, 1997], or exposure to unanticipated operating conditions that increase
7152 reactivity). Decisions on the adequacy of the safety analysis relevant to these difficult-to-
7153 quantify uncertainties are more straightforward if design-specific analyses are provided that
7154 estimate the additional reactivity margins available from absorber nuclides (fission products and
7155 actinides) not included in the licensing safety basis (as described in SRP Section 7.5.5.1). The
7156 reviewer should assess the estimated reactivity margins to determine their adequacy for
7157 offsetting any potential uncertainties introduced by the type of effects discussed above.

7158

7159 **7.5.6 Supplemental Information**

7160
7161 The reviewer should ensure that all supportive information or documentation is provided. This
7162 may include, but not be limited to, justification of assumptions or analytical procedures, test
7163 results, photographs, computer program descriptions, input/output, and applicable pages from
7164 referenced documents. In addition, the SAR should include a list of fuel designs with the
7165 acceptable parametric limits and the maximum enrichments for which the criticality analysis is
7166 valid. The reviewer should request any additional information needed to complete the review.

7167

7168 **7.6 Evaluation Findings**

7169

7170 The reviewer should review the 10 CFR Part 72 acceptance criteria and provide a summary
7171 statement for each. These statements should be substantially as follows:

7172

7173 F7.1 Structures, systems, and components important to criticality safety are described
7174 in sufficient detail in Chapters _____ of the SAR to enable an evaluation of their
7175 effectiveness.

7176

7177 F7.2 The _____ cask and its spent fuel transfer systems are designed to be
7178 subcritical under all credible conditions.

7179

7180 F7.3 The criticality design is based on favorable geometry, fixed neutron poisons, and
7181 soluble poisons of the spent fuel pool [as applicable]. An appraisal of the fixed
7182 neutron poisons has shown that they will remain effective for the term requested
7183 in the CoC application and there is no credible way for the fixed neutron poisons
7184 to significantly degrade during the requested term in the CoC application;
7185 therefore, there is no need to provide a positive means to verify their continued
7186 efficacy as required by 10 CFR 72.124(b).

7187

7188 F7.4 The analysis and evaluation of the criticality design and performance have
7189 demonstrated that the cask will enable the storage of spent fuel for the term
7190 requested in the CoC application.

7191

7192 The reviewer should provide a summary statement similar to the following:

7193

7194 “The staff concludes that the criticality design features for the [cask designation] are in
7195 compliance with 10 CFR Part 72, as exempted [if applicable], and that the applicable
7196 design and acceptance criteria have been satisfied. The evaluation of the criticality
7197 design provides reasonable assurance that the [cask designation] will allow safe storage
7198 of spent fuel. This finding is reached on the basis of a review that considered the
7199 regulation itself, appropriate regulatory guides, applicable codes and standards, and
7200 accepted engineering practices.”

8 MATERIALS EVALUATION*

8.1 Review Objective

The materials review ensures adequate material performance of components important to safety of a dry cask storage system (DSS), including the spent fuel canister or cask, under normal, off-normal, and accident-level conditions. To ensure an adequate margin of safety in the design basis of the DSS, the reviewer should obtain reasonable assurance that:

- The physical, chemical, and mechanical properties of materials for components important to safety (ITS) meet their service requirements including normal, off-normal, and accident-level conditions, and that the mechanical properties are Code accepted values.
- Materials for components ITS have sufficient requirements to control the quality of the production, fabrication, and test activities.
- Materials for ITS components are selected to accommodate the effects of, and to be compatible with, the independent spent fuel storage installation (ISFSI) site characteristics, environmental conditions, and duration of the license period.
- The spent nuclear fuel (SNF) cladding is protected from gross rupture and from conditions that could lead to fuel redistribution.
- The DSS is designed to maintain the spent fuel in a readily retrievable condition.
- Other materials which support or protect ITS components (such as coatings) are suitable for the application.

In reviewing the materials, the reviewer should consider the sources of information for the physical and mechanical properties of the materials used in the DSS construction and those materials which are part of the spent fuel payload. These material properties should be considered against both static and dynamic loadings for normal, off-normal, accident conditions, and other phenomena such as corrosion. The material properties and characteristics needed to satisfy these functional safety requirements should be maintained and are applicable over the complete licensing period.

Preferred materials information sources are U.S. industry consensus codes, standards, and specifications. The applicability and acceptability of all other sources, such as manufacturer's test data and handbooks, should be reviewed. The reviewer should also examine published articles, research reports, and texts as sources of information concerning material performance. Foreign standards are not generally acceptable and would only be reviewed for acceptability on a case-by-case basis.

* The technical guidance of this chapter is primarily arranged by subject matter, regardless of where in a SAR the material may appear.

7245 **8.2 Areas of Review**

7246
7247 The materials evaluation encompasses the following listed areas of review. Note, specifically
7248 items marked (*) are items that must also be addressed in the Technical Specifications (TS).
7249 The various materials engineering related topics requiring review may be addressed in different
7250 chapters of the SAR. However, the review guidance for all materials engineering related topics
7251 are provided in this chapter of the SRP.

7252
7253 Areas for materials review:

7254
7255 **General**

7256
7257 Cask Design/Materials
7258 Environmental Conditions
7259 Engineering Drawings

7260
7261 **Materials Selection**

7262
7263 *Applicable Codes and Standards and Alternatives to the Code
7264 Material Properties
7265 *Alternative or Substitute Materials (ITS components)
7266 *Weathering Steels for Coastal ISFSI Locations (specific DSS designs)
7267 Weld Design, Inspection
7268 Bolt Applications
7269 Coatings
7270 Neutron Shielding Materials
7271 Gamma shielding
7272 *Neutron Poison Materials for Criticality Control
7273 Concrete and Reinforcing Steel
7274 Seals
7275 *Low Temperature Ductility of Ferritic Steels
7276 Creep Properties/Analyses

7277
7278 **Corrosion**

7279
7280 Corrosion Resistance
7281 Galvanic/Chemical/Radiolytic Reactions of Fuel with Canister Internals

7282
7283 **Cladding Integrity/Fuel**

7284
7285 *Fuel Burn-up
7286 *Cladding Temperature Limits
7287 *Damaged Fuel Definition

7288
7289 **Operational Issues** (see Operating Procedures Chapter of SAR)

7290
7291 *Hydrogen gas monitoring/mitigation
7292 *Preventing oxidation of fuel during loading/unloading operations which can lead
7293 to Rod Splitting

7294

7295 **Examination and Testing** (see Acceptance Test Chapter of SAR)

7296
 7297 *Helium leakage testing of canister welds
 7298 Periodic Inspections

7299
 7300 **Code Case Acceptability**

7301
 7302 Refer to Regulatory Guide 1.193

7303
 7304 **8.3 Regulatory Requirements**

7305
 7306 This section presents a summary matrix of the portions of U.S. Code of Federal Regulations
 7307 (CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and
 7308 High-Level Radioactive Waste," Title 10, "Energy" (10 CFR Part 72) relevant to the review areas
 7309 addressed by this chapter. The U.S. Nuclear Regulatory Commission (NRC) staff reviewer
 7310 should read the exact referenced regulatory language. Table 8-1 matches the relevant
 7311 regulatory requirements associated with this chapter to the areas of review.
 7312

Table 8-1 Relationship of 10 CFR Part 72 Regulations and Areas of Review					
Chapter 8 Areas of Review	10 CFR Part 72 Regulations				
	72.104(a)	72.106(b)	72.122 (a), (b), (c)	72.122 (h)(1), (i), (l)	72.124
General					
Materials Selection	•	•	•		•
Corrosive Reactions					
Cladding Integrity				•	

7313

Chapter 8 Areas of Review	10 CFR Part 72 Regulations			
	72.236(g)	72.236(h)	72.236(i)	72.236(m)
General				•
Materials Selection	•		•	•
Corrosive Reactions		•		
Cladding Integrity				•

7314
 7315 **8.4 Review Procedures and Acceptance Criteria**

7316
 7317 Technical Specifications (TS) and license conditions are the legally enforceable portions of a
 7318 CoC. The body of the SAR and the staff SE are not enforceable. Therefore, any technical
 7319 aspect of the design which is deemed critical to nuclear safety must appear in the TS.
 7320 Incorporation by reference into the TS is acceptable and often employed to avoid having the TS
 7321 become unwieldy in size.
 7322

7323 In this chapter, those materials items which have been deemed necessary to incorporate into
7324 the TS are marked with an asterisk (*) at the beginning of the discussion to alert the reviewer.
7325 Metallic materials are primarily assumed in this guidance. The interrelationship of the materials
7326 evaluation review with other disciplines is shown in Figure 8-1.

7327

7328 **8.4.1 General Review Considerations (HIGH Priority)**

7329

7330 Survey the SAR (generally SAR Chapters 1 and 2) and especially the Technical Specifications
7331 (TS) and talk to the project manager (PM) to gain an overall understanding of the nature of the
7332 license application. Most license applications are submitted as “amendments” to existing (and
7333 thus previously reviewed) designs. However, the actual nature of the “amendment” may
7334 encompass anything from a minor change to a completely new design. Beware, not all design
7335 and license changes in the amendment will necessarily be separately identified by the applicant
7336 nor necessarily be obvious.

7337

7338 This means any amendment should be approached with the view that any topic in the
7339 amendment request, TS and supporting SAR is open to review by the technical staff. This
7340 applies regardless of whether or not it is considered within scope to the specific amendment
7341 request. Sometimes this creates a conflict with the PM and/or applicant, and management
7342 consultation can be required. Regardless, the technical staff should principally ensure the
7343 technical content of any amendment application and supporting TS or SAR is acceptable. Any
7344 technically incorrect or unacceptable design or operational aspect should be identified for
7345 resolution.

7346

7347 To give an amendment a “quick review,” the following Technical Specification (TS) items should
7348 be examined. Detailed discussion is provided in the following sections.

7349

7350 TS items to confirm:

7351

7352 Maximum fuel burn-up

7353

7353 Maximum cladding temperature

7354

7354 Definition of damaged fuel

7355

7355 Code of record and alternatives to specific Code requirements

7356

7356 Specification/requirements for alternative materials for ITS components

7357

7357 Manufacture and testing of neutron poison material(s)

7358

7358 Hydrogen monitoring/mitigation during wet loading/unloading

7359

7359 Helium leakage testing of cover welds

7360

7360 Maintaining inert atmosphere during canister draining/flooding

7361

7361 No Code Case N-595

7362

7362 Use of copper bearing structural carbon steel at coastal marine ISFSI sites (presently
7363 only for NUHOMS designs)

7364

7365 Non-TS items of recent interest include:

7366

7367 Design temperature for aluminum components used in the fuel basket or canister interior
7368 (creep issues)

7369

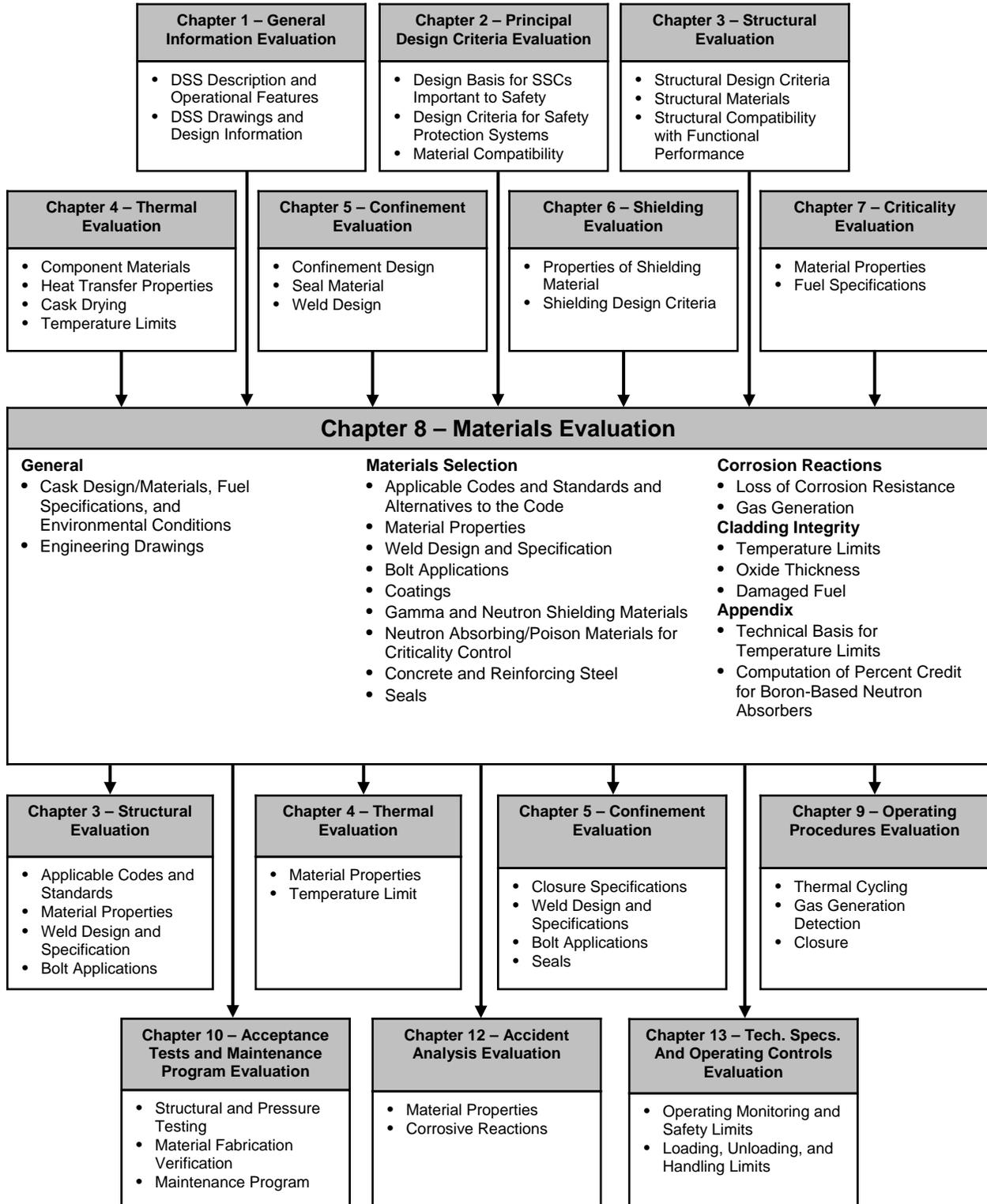


Figure 8-1 Overview of Materials Evaluation

7376 **8.4.2 Codes and Standards (HIGH Priority)**
7377
7378 *8.4.2.1 Usage and Endorsement
7379
7380 Codes (or “construction codes”) govern which materials may be used and how they may be
7381 employed. Standards detail how a material is produced and establishes chemical and material
7382 property requirements. All ASME and AWS materials are a subset of ASTM materials.
7383 However, not all ASTM materials are endorsed for use by the ASME or other codes which may
7384 be used for canister design.
7385
7386 The SAR must identify applicable codes and standards used in the design, selection, and use of
7387 materials. For important-to-safety (ITS) components, U.S. industry consensus codes and
7388 standards such as ASME, AWS, ANSI, ACI, and ASTM should be specified.
7389
7390 Foreign codes and standards are generally NOT acceptable for ITS components/materials and
7391 would only be approved on a case-by-case basis. However, foreign-produced materials which
7392 comply with U.S. codes and standards are acceptable.
7393
7394 Materials for ITS components will normally be ASME Section II approved materials. Alternative
7395 or substitute materials for ITS components should also be ASME Section II materials.
7396 Alternative materials must be specifically listed. Use of terms such as “equivalent” without a
7397 specification to a specific ASME Section II material is not acceptable for ITS component
7398 materials.
7399
7400 Materials for non-ITS components should be specified to ASTM standards. Alternative
7401 materials for non-ITS components should be other ASTM materials or specified sufficiently as to
7402 ensure equivalent performance. Equivalent performance means the alternative material(s) must
7403 have the same or higher ultimate tensile strength (UTS), yield strength (YS), elongation, and
7404 Charpy values in addition to chemical composition that falls within or close to the chemical
7405 composition range of the originally specified material. Foreign material specs would be
7406 acceptable for non-ITS component materials.
7407
7408 Foreign-produced materials which comply with U.S. codes and standards are acceptable.
7409
7410 * Proprietary materials which are ITS (specifically neutron poisons) must be described
7411 sufficiently in SAR Chapter 8, “Materials” so no changes to the materials composition,
7412 performance, or manufacturing methods are allowed without prior NRC review. Additionally, the
7413 governing quality assurance and quality control (QA/QC) documents, manufacturing
7414 procedures, and testing protocols for neutron poisons must be incorporated by reference into
7415 the TS.
7416
7417 Polymeric neutron shielding materials, which are usually proprietary, are not considered
7418 important-to-safety (ITS) materials. Thus no TS reference to these materials is warranted.
7419
7420 * The code of record and alternatives to the code for ITS components must be specified
7421 in the TS.
7422
7423 * Ensure substitute materials used for ITS components are specified in the TS.
7424
7425 * Ensure any invoked ASME Code Cases are specified in the TS.
7426

7427 8.4.2.2 Code Case Use/Acceptability

7428
7429 Review any referenced ASME Code cases against Regulatory Guide 1.193 for acceptability.
7430 Note that Code Case N-595 (any revision) has been found unacceptable to the staff per
7431 RG 1.193.

7432
7433 **8.4.3 Environment (Priority – as indicated)**

7434
7435 (MEDIUM Priority) Generally, the ISFSI site with associated storage canisters are subjected
7436 (long-term) to a mild atmospheric environment. Twenty or more years of ISFSI operational
7437 experience has verified that no significant corrosion issues generally exist during storage.
7438 However, note whether or not the site or potential site is a coastal marine location. Additional
7439 corrosion prevention measures may be applied when the ISFSI is located in a coastal marine
7440 environment. Detailed review guidance is provided in 8.4.6 Coastal Marine ISFSI Sites–
7441 Material Selections.

7442
7443 (LOW Priority) Underground structures require additional consideration due to soil corrosion
7444 issues. Additional guidance is provided in 8.4.14.3 Omission of Reinforcement.

7445
7446 (LOW Priority) Fuel loading/unloading conditions assume a borated, demineralized water
7447 environment at temperatures up to the boiling point. Experience with the conventional stainless
7448 steel and aluminum construction canister internals have verified no significant corrosion of fuel
7449 canister ITS components occur during the limited duration of a fuel loading/unloading operation.
7450 Pool water is buffered to a pH of about 8.5 to limit corrosion.

7451
7452 **8.4.4 Drawings (MEDIUM Priority)**

7453
7454 Licensing drawings usually appear in SAR Chapters 1 or 2. Examine the drawings and drawing
7455 notes for material specifications and alternatives. Ensure any materials substitutes are
7456 adequately specified, either on the drawing or in the SAR. ITS component material substitutes
7457 must appear in the TS.

7458
7459 **8.4.5 Material Properties (MEDIUM Priority)**

7460
7461 8.4.5.1 Structural Properties

7462
7463 The intent of this portion of the materials evaluation is to determine the acceptability of all
7464 material properties that have a structural role in confinement system structures and other
7465 structures important to safety (e.g., the basket, impact limiters, and shielding) and non-safety.
7466 The material properties and characteristics need to be applicable over the term requested in the
7467 CoC application. The reviewer should analyze the potential for corrosion and ensure that the
7468 applicant established and used appropriate corrosion allowances for the structural analyses.
7469 The range of some materials components properties may have to be evaluated over the range
7470 of life cycle conditions experienced during cask fabrication, loading, transportation,
7471 emplacement, storage, transfer, retrieval, unloading, and decontamination.

7472
7473 The information provided on structural materials must be consistent with the application of
7474 accepted design criteria, codes, standards, and specifications selected for the storage cask
7475 system and as described in this chapter and Chapter 3, “Structural Evaluation” of this SRP.
7476 Materials and material properties used for the design and construction of these safety-related
7477 structures should comply with the applicable codes and standards identified in Section

7478 3.5.2.2 (i). For example, if the applicant elects to use design criteria from Section III of the
7479 ASME B&PV Code, the materials selected for the cask must be consistent with those allowed
7480 by the ASME Code subsection related to design. Acceptable requirements include the ASME
7481 adopted specifications given in Section II, Part A, "Ferrous Metals;" Part B, "Nonferrous Metals;"
7482 Part C, "Welding Rods, Electrodes, and Filler Metals;" and Part D, "Properties." The review of
7483 structural materials should be coordinated with the structural discipline.
7484

7485 A list of all materials used and the proposed service conditions for those materials during
7486 loading, storage, and unloading is a useful aid during the review. These tables provide various
7487 types of information that the reviewer needs from an application to aid in determining the
7488 suitability of the materials for the structural evaluation. The tables include the name and safety
7489 classification of each component part of the DSS and, where applicable, the function, the
7490 material specification(s) to which it is produced, and the nominal values for structural
7491 parameters. The tabulation should include all materials used for components with an important-
7492 to-safety function (e.g., confinement, transfer, criticality control, shielding). Information in this
7493 table can aid the reviewer to formulate the types of performance-related questions that are
7494 important for each component of a storage system.
7495

7496 The SAR documentation should fully define the structural materials used for components
7497 important to safety. The reviewer may find it useful to tabulate the major structural materials to
7498 facilitate the review. The following information could be tabulated: specification number, grade,
7499 type, and class of the material, nominal composition, product form, yield strength, tensile
7500 strength, and notes about the materials, etc. The SAR should identify properties related to
7501 structural performance and resistance or response to thermal, radiation, or other applicable
7502 environments that may impact structural performance. The structural and material disciplines
7503 should coordinate their reviews as appropriate for these components.
7504

7505 The completeness, accuracy, and acceptability of the identification and stated properties of the
7506 safety-related materials should be reviewed. In reviewing the structural materials, the reviewer
7507 should consider the sources of information; properties used in the structural evaluation and
7508 suitability for term requested in the CoC application. The reviewer should verify that the SAR
7509 clearly references acceptable sources of all material properties.
7510

7511 Examine the SAR adopted material properties for ITS component materials and ensure ASME
7512 Section II, Part D, properties and stresses are employed. The longstanding staff position
7513 (developed by NRR) regarding material properties has held that ASME Code values must be
7514 used. Use of certified material test report (CMTR) values of UTS, yield, etc., is not permissible.
7515 Use of CMTR values is always at risk of being non-conservative. Steel producers are expert at
7516 knowing where to go in an ingot, billet, or forging to obtain samples with optimum properties for
7517 the certification record. These samples are usually archived for future reference if questions
7518 arise. Attempting to defeat a steel producer regarding tested values is practically impossible.
7519

7520 In the event more exact or confirmatory materials or chemical properties must be known, then a
7521 "product analysis" must be specified by the purchasing agent at the time of purchasing. Since
7522 this drives up the cost, it is infrequently invoked. Alternatively, a limited chemical analysis can
7523 be performed on a job site by use of a portable alloy analyzer.
7524

7525 8.4.5.2 Thermal Materials

7526
7527 The materials reviewer should coordinate with the thermal reviewer to determine the materials
7528 properties of the materials important to the thermal analysis. The material compositions and

7529 thermal properties such as thermal conductivity, thermal expansion, specific heat, and heat
7530 capacity should be verified as a function of the temperature over the range the components are
7531 to operate, for all components used in the safety analysis. Verify the change in these material
7532 properties due to potential degradation of materials over their service life has been evaluated by
7533 the applicant. Temperature and anisotropic dependencies of thermal properties should be
7534 considered.

7535

7536 * **8.4.6 Coastal Marine ISFSI Sites–Material Selections (MEDIUM Priority)**

7537

7538 At coastal marine locations, the heavy salt drift can significantly accelerate the normally slight
7539 atmospheric corrosion rate to unacceptable values of some canister storage module designs,
7540 such as those that employ carbon steel structural elements inside the canister storage module.
7541 Experience has shown ordinary grades of structural steel (such as A-36) withstand the
7542 nominally dry interior environment of the canister overpack very well over a 20 year operational
7543 period.

7544

7545 For such cases, the reviewer must verify that the corrosion allowance specified is adequate for
7546 the 20 to 40 year CoC period of the canister. Corrosion rates for carbon steel in air may be
7547 found in corrosion references such as Corrosion Engineering by Fontana and Greene,
7548 Corrosion Data Survey by the National Association of Corrosion Engineers (NACE), Corrosion
7549 and Corrosion Control by Uhlig, and the publications of the NASA Kennedy Space Center
7550 Corrosion Technology Laboratory. For exposures to coastal marine atmospheres, the corrosion
7551 rate data from the Kennedy Space Center Corrosion Technology Laboratory appears to be
7552 bounding for any location in the continental United States.

7553

7554 To address the increased atmospheric corrosion rates found at coastal marine sites, TN has
7555 specified the use of copper-bearing alloy steel (aka “weathering steels” such as Cor-Ten) with a
7556 minimum copper content of 0.20 percent. The Kennedy Space Center data and a proprietary
7557 study conducted for TN has shown a significant benefit (significantly reduced corrosion rate) by
7558 employing the weathering steels at coastal marine sites. For example, for coastal marine ISFSI
7559 sites, the use of weathering steels containing a minimum of 0.20 percent copper may be
7560 necessary. Such steels are covered by ASTM A-242 and other specifications.

7561

7562 Coatings may be specified to alleviate the coastal atmospheric corrosion issue. However,
7563 unless the coating is periodically inspected and maintained, no credit may be given for its
7564 presence.

7565

7566 **8.4.7 Weld Design/Inspection (MEDIUM Priority)**

7567

7568 8.4.7.1 Welding Codes–Background Discussion

7569

7570 The nationally recognized codes which have been used for spent fuel canister construction
7571 include:

7572

7573 • ASME B&PV Code, Section III, “Rules for Construction of Nuclear Facility
7574 Components,” Division 1.

7575

7576 • AWS D1.1 (current edition), “Structural Welding Code-Steel.”

7577

7578 The ASME B&PV Code Section III contains the design requirements for nuclear systems at a
7579 commercial nuclear power plant. It contains sections governing the design of welded nuclear
7580 components in the plant.

7581
7582 AWS D1.1 is the structural welding code for carbon steel structures such as bridges and steel-
7583 framed buildings.

7584
7585 The NRC staff accepts the use of the ASME B&PV Code, Section III, as the preferred
7586 construction code for storage casks. Some older cask designs used the AWS D1.1 Code.
7587 Note, the various construction codes (e.g., ASME Sections I, III, or VIII, and AWS D1.1) differ
7588 from one another in their requirements for materials and welding procedures, because each
7589 code is specialized with a particular application in mind.

7590
7591 The ASME construction codes are supplemented by “supporting codes” which detail how
7592 special processes such as welding and nondestructive examination (NDE) are to be qualified
7593 and executed. ASME B&PV Code Section IX, “Welding and Brazing Qualifications” details the
7594 requirements for specifying and qualifying a welding procedure and for testing and qualifying
7595 welders. ASME B&PV Code Section V, “Nondestructive Examination,” supports the various
7596 ASME construction codes by detailing the required qualifications for NDE examiners and the
7597 requirements and methods for performing the types of NDE specified by the various
7598 construction codes.

7599
7600 Standard welding and NDE symbols may be found in AWS A2.4 (latest edition), “Symbols for
7601 Welding, Brazing, and Nondestructive Testing,” to aid interpretation of such symbols found on
7602 the drawings submitted with the SAR.

7603
7604 Technical specification items related to the welds and testing are discussed separately.

7605
7606 8.4.7.2 Weld Design and Testing

7607
7608 Verify the canister shell welds (sides and bottom closure) are full penetration welds. Inspection
7609 of these welds must follow the ASME Code requirements of full volumetric examination
7610 [radiographic testing (RT) or ultrasonic testing (UT)] and a surface examination [liquid penetrant
7611 testing (PT), for austenitic stainless steel canisters]. A hydrostatic or pneumatic test is also
7612 required by the Code.

7613
7614 Stainless steel fillet welds can only be inspected by PT. Volumetric inspection of fillet welds is
7615 not feasible.

7616
7617 Due to the relatively benign operating conditions in storage, imposition of specific weld filler
7618 metals, or use/prohibition of certain welding processes is not presently necessary. Sensitization
7619 of the stainless steel is not an issue. Hence, solution annealing is unnecessary.

7620
7621 In order to comply with 10 CFR 72.122(h)(5), a helium leakage test is performed of the entire
7622 canister shell.

7623
7624 The hydrostatic or pneumatic test and the helium leakage test is performed in the fabrication
7625 shop before fuel basket installation. A temporary lid is placed on the canister and the tests
7626 conducted with the temporary lid in place.

7627

7628 8.4.7.3 Lid Welds and Closure Welds

7629

7630 The staff should verify the cask design is in compliance with Section 8.9 of this SRP or follows:

7631

7632 • This guidance only applies to canisters of all-welded construction, fabricated from
7633 austenitic stainless steel, and employing redundant welds for the confinement
7634 closure.

7635

7636 • The welded canister (i.e., the confinement boundary) must be leak tested in
7637 accordance with ANSI N14.5-1997, except as specified by this guidance. The
7638 exemption for leak testing only applies to the closure welds that are typically
7639 made in the field and all other welds should be leak tested.

7640

7641 • “Structures, systems, and components important to safety must be designed to
7642 withstand postulated accidents” (10 CFR 72.122(b)).

7643

7644 • Records documenting the lid welds shall comply with the provisions of 10 CFR
7645 Part 72.174, “Quality Assurance Records.” Records storage should comply with
7646 ANSI N45.2.9, “Requirements for Collection, Storage, and Maintenance of
7647 Quality Assurance Records for Nuclear Power Plants.”

7648

7649 • Activities related to inspection, evaluation, documentation of fabrication, and lid
7650 welding shall be performed in accordance with an NRC-approved quality
7651 assurance program as required in 10 CFR Part 72, Subpart G, “Quality
7652 Assurance.”

7653

7654 A redundant sealing of the canister is required by 10 CFR 72.236(e). One of the redundant
7655 seals in a welded canister design will involve a structural weld. The structural lid weld joint will
7656 be a full or partial penetration groove weld.

7657

7658 *Carbon and Alloy Steel Cask Designs*

7659

7660 The reviewer should verify the applicant has considered all the closure lid weld material and
7661 technique improvements that accrued from previous DSS design and fabrication experience.
7662 For example, the reviewer should refer to the technical evaluation in NRC Confirmatory Action
7663 Letter 97-7-001, 1998 (ADAMS ML 060620420). Some of the DSS improvements resulting from
7664 that action include:

7665

7666 • Shell plates made from low sulfur, calcium-treated, vacuum-degassed steel.

7667

7668 • Application of minimum 93°C (200°F) preheat.

7669

7670 • Use of low-hydrogen electrodes.

7671

7672 • Low carbon equivalent base metals and weld metals.

7673

7674 • Magnetic particle examination (MT) of the root pass.

7675

7676 • Maintenance of preheat as a postheat treatment for a minimum of one hour.

7677

- 7678 • Minimum of two-hour delay after postheat before performing final volumetric
7679 NDE.
7680

7681 The structural lid weld should be examined by UT or other volumetric methods. The applicant's
7682 evaluation of the critical flaw size using the linear-elastic fracture mechanics methodology
7683 should be reviewed based on service temperature, dynamic fracture toughness, and critical
7684 design stress parameters, as specified in ASME B&PV Code, Section XI, 2001.
7685

7686 Progressive surface examinations, utilizing a PT or magnetic particle testing (MT), are permitted
7687 only if unusual design and loading conditions exist. In addition, a stress-reduction factor of 0.8
7688 is imposed on the weld strength of the closure joint to account for imperfections or flaws that
7689 may have been missed by progressive surface examinations. The weld design should be
7690 approved by the NRC on a case-by-case basis.
7691

7692 8.4.7.4 Austenitic Stainless and Nickel-Base Alloy Steels Cask Design 7693

7694 For designs employing austenitic lid materials and welds, either volumetric or multi-pass PT
7695 inspection methods are acceptable.
7696

7697 For either UT or PT, the minimum detectable flaw size must be demonstrated to be less than
7698 the critical flaw size. The critical flaw size should be calculated in accordance with ASME B&PV
7699 Code, Section XI methodology; however, net section stress may be governing for austenitic
7700 stainless steels, and must not violate ASME B&PV Code Section III, Division 3 requirements.
7701 Flaws in austenitic stainless steels are not expected to exceed the thickness of one weld bead.
7702

7703 If using UT, the UT acceptance criteria are the same as those of paragraph NB-5332 of the
7704 ASME B&PV Code, Section III, for pre-service examination. In accordance with Code practice
7705 for supplementing volumetric examinations with a surface examination, UT must be performed
7706 in conjunction with a root pass and cover pass PT.
7707

7708 If PT is specified (i.e., no volumetric inspection) a stress reduction factor of 0.8 must be applied
7709 to the weld design.
7710

7711 **8.4.8 Galvanic/Corrosive Reactions (LOW Priority)** 7712

7713 8.4.8.1 Environmental considerations 7714

7715 Pursuant to NRC Bulletin 96-04 (1996), the reviewer should confirm the DSS will perform
7716 adequately under the operating environments expected (e.g., short-term loading/unloading or
7717 long-term storage) for the duration of the license period such that no adverse galvanic or
7718 corrosive reactions occur between the canister materials, fuel payload, and the operating
7719 environments.
7720

7721 8.4.8.2 Canister Contents 7722

7723 The staff has previously reviewed a number of non-fuel hardware components and materials for
7724 compliance with 10 CFR 72.120(d), meaning, compatibility with a canister interior composed of
7725 stainless steel and aluminum components. These components are various neutron source
7726 assemblies, burnable poison rod assemblies (BPRAs), thimble plug devices, and other types of
7727 control elements. The staff has found the following materials to be acceptable for storage when

7728 the canister is constructed of stainless steel with stainless steel and aluminum basket
7729 components:

7730
7731 * Neutron source materials composed of stainless steel or zirconium alloy cladding containing:
7732 antimony-beryllium, americium-beryllium, plutonium-beryllium, polonium-beryllium, and
7733 californium. Exposure of these various contents to the wet loading and dry storage environment
7734 was assessed and found to be satisfactory.

7735
7736 * Control elements composed of zircaloy or stainless steel cladding containing: boron carbide,
7737 borosilicate glass, silver-indium-cadmium alloy, or thorium oxide. Exposure of these various
7738 contents to the wet loading and dry storage environment was assessed and found to be
7739 satisfactory.

7740
7741 **8.4.9 Creep Behavior of Aluminum Components (HIGH Priority)**
7742

7743 Aluminum based metal matrix composites are employed for all presently utilized neutron poison
7744 materials. Also, aluminum components are frequently part of the spent fuel basket. More
7745 recent designs have specified ever higher design temperatures for the fuel basket components
7746 in order to accommodate higher loading densities and higher burn-up fuel. This trend has
7747 pushed the various aluminum components well into creep regime operating temperatures.

7748
7749 Review the design maximum temperatures and stress for any aluminum components and verify
7750 a creep analysis has been performed if any load bearing (including dead-weight loads)
7751 aluminum components operate at a design temperature above approximately 200°F. In the
7752 event temperatures exceed the ASME Section II nominal 400°F temperature limit for aluminum,
7753 other sources for creep data must be examined. One previously cited reference for this
7754 information is: D.W. Wilson, J.W. Freeman and H.R. Voorhees, Creep-Rupture testing of
7755 Aluminum Alloys to 100,000 Hours, First Progress Report, Prepared for the Metal Properties
7756 Council, New York, November 1969. The staff makes no judgment as to the acceptability of this
7757 reference. This is because the designs reviewed through the time of this writing have had
7758 design stresses (on the order of tens of PSI) which were substantially below the creep-rupture
7759 stresses provided in the referenced report. None-the-less, an assessment of creep deformation
7760 over a 20 to 40 year CoC period should be part of the design calculations.

7761
7762 Borated aluminum neutron poison materials must be considered on a case-by-case basis if they
7763 are subjected to any kind of loading. This is due to their inherently low ductility and generally
7764 unknown creep properties.

7765
7766 **8.4.10 Bolt Applications (MEDIUM Priority)**
7767

7768 If threaded fasteners are employed for ITS components, verify the bolt material(s) have
7769 adequate resistance to corrosion and brittle fracture and a coefficient of thermal expansion
7770 similar to the materials being bolted together.

7771
7772 **8.4.11 Exterior Protective Coatings (LOW Priority)**
7773

7774 Coatings in DSSs are used primarily as corrosion barriers or to facilitate decontamination. They
7775 may have additional roles, such as improving the heat rejection capability by increasing the
7776 emissivity of cask internal components. Protective coatings are occasionally specified for
7777 carbon steel components. Coatings are not ITS components. The structures or components
7778 that the coatings are applied to are generally ITS component. No coating should be credited for

7779 protecting the substrate material or extending the useful life of the substrate material unless a
7780 periodic coating inspection and maintenance program is required for the coating.

7781
7782 The staff has established this section to alleviate confusion regarding coatings on cask
7783 components. This guide outlines methods and procedures for appropriately assessing coatings.
7784 Within the assessment several areas are covered in detail including the scope of the coating
7785 application, type of coating system, surface preparation methods, applicable coating repair
7786 techniques, and coatings qualification testing.

7787
7788 U.S. Nuclear Regulatory Commission (NRC) staff reviewer should read 72.122(b), 72.122(h),
7789 72.122(l), 72.236(h), 72.236(i), 72.236(j).

7790
7791 8.4.11.1 Review Guidance

7792
7793 The reviewer should determine the appropriateness of the coating(s) for the intended
7794 application by reviewing the coating specification for each protective coating that is applied to an
7795 important to safety component. A specification that describes the scope of the work, required
7796 materials, the coating's purpose, and key coating procedures, should ensure that the
7797 appropriate and compatible coatings have been selected by the DSS designers. A coating
7798 specification should include the following:

- 7799
7800 • Scope of coating application;
7801 • Type of coating system;
7802 • Surface preparation methods;
7803 • Coating application method;
7804 • Applicable coating repair techniques;
7805 • Coatings qualification testing, as applicable.

7806
7807 8.4.11.2 Scope of Coating Application

7808
7809 The coating specification should identify the purpose of the coating, a list of the components to
7810 be coated, and a description of the expected environmental conditions (e.g., expected
7811 conditions during loading, unloading, and dry storage).

7812
7813 The reviewer should verify that the coatings will not react with the cask internal components and
7814 contents and will remain adherent and inert when exposed to the various environments of a
7815 SNF cask. The most prevalent, potentially degrading environments include the immersion in
7816 borated SNF pool water during loading and unloading operations, and high-temperature and
7817 high-radiation (including neutrons) environments encountered during vacuum drying evolutions
7818 and long-term storage.

7819
7820 8.4.11.3 Coating Selection

7821
7822 The reviewer should verify that the coating specification identifies the manufacturer's name, the
7823 type of primers and topcoat(s) comprising the coating system, and the minimum and maximum
7824 dry coating thickness(es). The coating manufacturer's technical literature for all coatings
7825 specified for cask interiors must be submitted in the SAR for staff review.

7826
7827 The reviewer should verify that the coating selected for cask components is capable of
7828 withstanding the intended service conditions over the design service life. Failures can be

7829 prevented by ensuring that the selection and the application of the coating is controlled by
7830 adhering to the coating manufacturer's recommendations.

7831
7832 8.4.11.4 Surface Preparation

7833
7834 The reviewer should verify that the coating specification identifies whether solvent or abrasive
7835 cleaning methods should be used to prepare surfaces prior to coating application. This
7836 information should ensure that proper surface preparation techniques can be implemented
7837 during cask fabrication.

7838
7839 The reviewer should confirm that the specified type and degree of surface cleaning and the
7840 required surface profile meet the coating manufacturer's specification. Any deviations from the
7841 manufacturer's standards for surface preparation must be supported by appropriate tests that
7842 demonstrate acceptable coating performance under all design conditions.

7843
7844 8.4.11.5 Coating Repairs

7845
7846 The reviewer should verify that the coating specification identifies the general requirements for
7847 repairing damage to the coating. This information will assist the reviewer in evaluating the
7848 effects of repairs on the integrity of the coating and whether the designated repair methods
7849 could be implemented during or after cask fabrication.

7850
7851 The reviewer should examine the design to determine whether the structure is assembled
7852 before or after its various parts are coated. If a complex structure is to be coated after
7853 assembly, it is very important that the consequences of a potential coating failure be analyzed to
7854 determine whether other cask functions or component features could be compromised by the
7855 failure.

7856
7857 The consequences of coating failure depend on the type of coating and service environment,
7858 and may include the following:

- 7859
- 7860 • Partial and/or complete coating failure that alters the corrosion resistance of DSS
7861 structural and shielding components (primarily during loading/unloading
7862 operations).
 - 7863
 - 7864 • Partial and/or complete coating failure that alters the emissivity and heat transfer
7865 of basket components.
 - 7866
 - 7867 • Particulates (cloudiness) that form in SNF pool water or cask during loading or
7868 unloading that may affect such operations.
 - 7869
 - 7870 • Aggressive or reactive chemical species that form and consequently impact the
7871 performance of other cask components during long-term exposure to radiation
7872 (e.g., gamma and neutron).
 - 7873

7874 8.4.11.6 Coating Qualification Testing

7875
7876 Coatings used on cask external surfaces may have been selected upon the basis of their
7877 performance requirements and exposure conditions. The applicant may have used related
7878 industrial conditions as a documented guide or basis for coating selection without performing
7879 further laboratory tests.

7880
7881 Any coating used inside a DSS must have been tested to demonstrate the coatings
7882 performance under all conditions of loading and storage. The conditions evaluated should
7883 include exposure to radiation, high temperature during vacuum drying and storage, and
7884 immersion during loading, unloading and transfer operations. The coating must be
7885 demonstrated to remain intact and inert for the full duration of the DSS design life.

7886
7887 There are a number of standardized ASTM tests for coatings performance. In reviewing ASTM
7888 (or other) tests used to qualify coatings for service in storage casks, consideration should be
7889 given to the applicability of a test to the service conditions.

7890
7891 Planning, execution, and interpretation of coating qualification tests must be performed by a
7892 qualified coatings engineer (e.g., certified by the National Association of Corrosion Engineers).
7893 The reviewer should ensure that appropriate, qualified expertise has been employed by the
7894 applicant for any coatings qualification program.

7895
7896 The reviewer should verify that the coating specification includes a description of the coating
7897 qualifications testing program, as applicable. The following information, which is important to
7898 qualifying a coating, includes, but is not limited to:

7899
7900 • The size and shape of samples used for the coating tests, as well as the type of
7901 material(s), and a description and results of any tests conducted on partial or full-
7902 size production mock-ups.

7903
7904 • The test sample surface preparation method(s) and expected or measured
7905 surface profile. Sample surface preparation should be performed in accordance
7906 with written production procedures, using the same equipment, materials, and
7907 qualified personnel as intended for production coating. Inspection methods and
7908 acceptance criteria should be included.

7909
7910 • Application method(s) and measured control parameters, including records of
7911 temperature and humidity, cure cycle and times, and any other monitoring or
7912 acceptance tests such as dry film thickness, hardness, and adhesion. The
7913 methods and parameters should be employed in accordance with written
7914 production procedures using the same equipment, methods, materials, and
7915 qualified personnel.

7916
7917 • A test plan description which clearly describes the rationale for and the types and
7918 sequences of all coating qualification tests, lab protocols, numbers of samples,
7919 inspection methods, and acceptance criteria. Raw test results should be
7920 tabulated or otherwise presented. The test plan should include: (1) laboratory
7921 coupons for demonstrating coating suitability/qualification; and (2) partial or full
7922 size production mock-up tests that demonstrate that the selected coating can be
7923 applied successfully to real production parts under production shop conditions to
7924 give reasonable assurance that field performance will meet laboratory, test-
7925 based expectations.

7926
7927 • An interpretation and discussion of the test program results by a certified
7928 coatings engineer. This evaluation should examine, at a minimum, the coating
7929 performance against the specific tests and the overall requirements for coating
7930 performance. The overall program must be assessed as to whether it is likely to

7931 be an effective predictor of actual performance. A recommendation for the use of
7932 the coating, with specific restrictions, if any, must be included.

7933
7934 The application should also include general requirements applying to all tests:

- 7935
- 7936 • Test durations for immersion must equal or exceed the combined maximum
7937 design (or technical specification) durations for loading and vacuum drying.
- 7938
- 7939 • An evaluation of any observed gasses, bubbles or other evidence that a gas was
7940 produced during the test. Coatings that produce flammable gas require a
7941 mitigation program to prevent burnable or explosive gas concentrations during all
7942 phases of cask operations.
- 7943

7944 **8.4.12 Neutron Shielding (MEDIUM Priority)**

7945 7946 8.4.12.1 Neutron Shielding Materials

7947
7948 Concrete, steel, depleted uranium, and lead typically serve as gamma shields. Boron-filled
7949 polymers are sometimes used for neutron shielding materials (as opposed to neutron poisons).
7950 These materials are not considered ITS since dose limits are calculated at the site boundary,
7951 not the canister surface. Further, in-service performance monitoring of these materials is
7952 possible during the required periodic radiation surveys. Should a decline in the shielding
7953 effectiveness be detected, there is ample time and opportunity for engineering evaluation and
7954 corrective action.

7955
7956 The SAR should describe the composition of shielding materials and geometries. References
7957 for all materials used, including nonstandard materials (e.g., proprietary neutron shield material),
7958 should be provided for the source of the material composition and density data along with
7959 validation of the data.

7960 7961 8.4.12.2 Assessing Previously Unreviewed (New) Neutron Shielding Materials

7962
7963 No new neutron shield materials have been introduced in several years. Should a new material
7964 be introduced, review may proceed as follows:

7965
7966 The reviewer should confirm that temperature-sensitive (e.g., polymeric) neutron shielding
7967 materials will not be subject to temperatures at or above their design limits during normal
7968 conditions. The reviewer should determine whether the applicant properly examined the
7969 potential for shielding material to experience changes in material densities at temperature
7970 extremes. For example, elevated temperatures may reduce hydrogen content through loss of
7971 water in concrete or other hydrogenous shielding materials.

7972
7973 With respect to polymeric neutron shields, the reviewer should verify that the application:

- 7974
- 7975 • Describes the test(s) demonstrating the neutron-absorbing ability of the shield
7976 material.
- 7977
- 7978 • Describes the testing program and provides data and evaluations that
7979 demonstrate the thermal stability of the resin over its design life while at the
7980 upper end of the design temperature range.
- 7981

- 7982 • Describes the nature of any temperature-induced degradation and its effect(s) on
7983 neutron shield performance.
- 7984
- 7985 • Describes what provisions exist in the neutron shield design to assure that
7986 excessive neutron streaming will not occur as a result of shrinkage under
7987 conditions of extreme cold. This description is required because polymers
7988 generally have a relatively large coefficient of thermal expansion when compared
7989 to metals.
- 7990
- 7991 • Describes any changes or substitutions made to the shield material formulation.
7992 For such changes, describes how they were tested and how that data correlated
7993 with the original test data regarding neutron absorption, thermal stability, and
7994 handling properties during mixing and pouring or casting.
- 7995
- 7996 • Describes the acceptance tests conducted to verify any filled channels used on
7997 production casks did not have significant voids or defects that could lead to
7998 greater than calculated dose rates.
- 7999
- 8000 * Describe the materials ability to withstand the combined aging effects of heat and
8001 radiation field.
- 8002

8003 The potential for shielding material to experience changes in material properties at temperature
8004 extremes should be described in the SAR. Temperature sensitivities of shielding materials
8005 should be referenced. The SAR should also address degradation from aging, accumulated
8006 radiation exposure, and manufacturing tolerances. Twenty years of operational experience has
8007 not resulted in any noticeable decline in the performance of previously accepted materials, as
8008 verified by examination of periodic radiation survey results on the ISFSI pads at Surrey and
8009 Robinson sites.

8010
8011 **8.4.13 Criticality Control (HIGH Priority)**
8012

8013 U.S. Nuclear Regulatory Commission (NRC) staff reviewer should read 72.104(a), 72.106(b),
8014 72.124, and 72.236(g).

8015
8016 Qualification testing is conducted to ensure that (1) the material used will have sufficient
8017 durability for the application for which it has been designed, (2) the physical characteristics of
8018 the components of the absorber materials will meet the design requirements, and (3) the
8019 uniformity of the distribution of ¹⁰B is sufficient to meet the requirements of the applications for
8020 which the absorber materials will be used. Qualification tests would be useful in establishing
8021 that the impurity concentration limits for borated absorbers are not exceeded. Agreement on
8022 these limits can be done by agreement between buyer and seller. Materials that have passed
8023 the qualification tests must be acceptance tested (See Chapter 10 of this SRP) for use in
8024 systems to be used in storage or transportation of nuclear fuel.

8025
8026 **8.4.13.1 Neutron-Absorbing/Poison Materials**
8027

8028 Various boron containing materials are used in the nuclear industry as neutron absorbers.
8029 Since these materials are used in storage containers for fissile materials, the materials should
8030 have excellent physical and chemical stability, including a high resistance to radiation and
8031 corrosion. Further, these materials should experience no reduction in effectiveness under
8032 normal/off-normal and accident-level conditions of storage. Neutron absorbers can consist of

8033 alloys of boron compounds with aluminum or steel in the form of sheets, plates, rods, liners, and
8034 pellets. Likewise, neutron absorbers can consist of a core containing mixed aluminum and
8035 boron carbide (B_4C) particles, clad on both sides with aluminum (a composite).
8036

8037 The neutron absorber material must be demonstrated to be adequately durable for the service
8038 conditions of the application. These assurances are usually obtained during qualification testing
8039 of the material. In addition, acceptance tests (see Chapter 10 of this SRP) are performed on
8040 samples from each production run of the material. This procedure will ensure the properties for
8041 the plates or other shapes produced are in compliance with the specifications and requirements
8042 of the application. The uniformity of the distribution of ^{10}B may be addressed in both the
8043 qualification and the acceptance tests.
8044

8045 For all boron-containing absorber materials, the reviewer should verify the SAR, with its
8046 supporting documentation, describes the absorber material's chemical composition, physical
8047 and mechanical properties, fabrication process, and minimum poison content. The
8048 manufacturer's data sheet should be submitted to supplement the above information. In the
8049 case of absorber plates or sheets, the minimum poison content should be specified as an areal
8050 density (e.g., milligrams of ^{10}B per cm^2).
8051

8052 8.4.13.2 Computation of Percent Credit for Boron-Based Neutron Absorbers 8053

8054 This section illustrates one method used by the materials reviewers to compute the level of
8055 credit to be allowed for 1/v neutron absorber materials, such as boron or lithium, in the criticality
8056 safety analysis of packages for storing fissile materials, including fresh and SNF. The
8057 computation of the allowed level of credit uses the results of neutron attenuation measurements
8058 performed on samples of the absorber material placed in a beam of thermal neutrons.
8059

8060 Where such validation uncertainties exist, an upper limit of 90 percent credit is applied to boron-
8061 based solid absorbers, meaning that the material is computationally modeled as containing only
8062 90 percent of the ^{10}B shown to be present. The staff has concluded that limiting the poison
8063 credit to 90 percent adequately accounts for the uncertainties arising in extrapolating the
8064 validation for boron-based absorber materials. Other remedies, beyond the scope of this
8065 guidance, may be necessary in addressing the potentially more complex neutron-spectral
8066 effects and validation uncertainties encountered with materials based on non-1/v-absorbers
8067 such as cadmium or gadolinium. The current guidance applies only to 1/v absorbers such as
8068 boron or lithium.
8069

8070 Neutron channeling has been shown to occur in a commercial product that uses coarse
8071 particles of natural B_4C dispersed in an aluminum matrix. For one material, neutron channeling
8072 effects reduced the measured attenuation of thermal neutrons by about 18 percent. Therefore,
8073 whenever uncertainty due to these materials factors exists in a product, it may be necessary to
8074 measure the neutron attenuation for that product to assess the expected material performance
8075 in service. Thus, in addition to the 90-percent limit on poison credit that is used to offset
8076 validation uncertainties, an additional penalty must be considered for material heterogeneity
8077 effects and uncertainties. In the absence of a fully documented understanding of non-
8078 uniformities and channeling effects in a heterogeneous absorber material, the staff recommends
8079 that the poison credit should continue to be limited to 75 percent.
8080

8081 A neutron absorber material is formulated to meet or exceed the neutron absorption effect
8082 computed to be required for a given service application. This guidance can be used to extend
8083 the range of credit for heterogeneous absorber materials from 75 to 90 percent, as follows:

- 8084
- 8085 • Material for which data is presented to show the measured attenuation for
- 8086 thermal neutrons to be at or above the acceptance attenuation (A_a), is given the
- 8087 full credit of 90 percent.
- 8088
- 8089 • Material for which data is presented to show the measured attenuation for
- 8090 thermal neutrons to be at levels between 75 and 100 percent of the acceptance
- 8091 attenuation (A_a) is given a fraction of the 90 percent credit allowed for fully
- 8092 effective absorber material.
- 8093
- 8094 • Material for which data is presented to show the measured attenuation for
- 8095 thermal neutrons to be at or below 75 percent of the acceptance attenuation (A_a)
- 8096 is not approved for use at any level of credit; the process used to produce such
- 8097 material is judged to be unsuitable.
- 8098

8099 The sampling, testing, and reporting of results shall be conducted according to the
 8100 specifications given in ASTM standard C1671-7.

8101 The applicable credit can be calculated by the following method. Using the following definitions:
 8102
 8103

- A = neutron attenuation, a measured value taken on a given absorber material in a beam of thermal neutrons with fixed energy spectrum. A is assumed to be normally distributed with mean μ and standard deviation σ .
- A_a = acceptance value of neutron attenuation, based on a qualified homogeneous absorber standard such as ZrB_2 , evaluated at 111 percent (i.e., 1/0.90) of the poison density assumed in the criticality computational model.
- $A_{t|}$ = attenuation tolerance limit, a statistic of the data
- n = number of coupon measures of attenuation
- P = probability
- μ = true mean of A
- x bar = estimate of μ
- σ = true standard deviation of A
- S = estimate of σ
- C_p = exact number of standard deviations required at probability P
- K_p = tolerance coefficient that is substituted for C_p when μ and σ are estimated by x bar and S, respectively
- γ = confidence level

8104
 8105 The attenuation data can be used to bound the probability P that the value of neutron
 8106 attenuation A at an arbitrary location on the material is greater than the acceptance attenuation
 8107 A_a . This is done by computing an attenuation tolerance limit, $A_{t|}$, such that, with 95-percent
 8108 confidence, the probability is less than 0.001 that $A < A_{t|}$.

8109
 8110 Let $P = 0.999$ and $\gamma = 0.95$. Compute $A_{t|} = (X \text{ bar} - K_p S)$, where $K_p = f(P, n, \gamma)$. The value of K_p
 8111 may be found in a table of one-sided tolerance coefficients for a normal distribution.

8112 If $A_{t|} \geq A_a$, then 90 percent credit is given.

8113
 8114 If $A_{t|} < A_a$, then compute the fractional credit from 0.75 to 0.90 as follows:
 8115
 8116

8117 Fractional Credit = $0.30 + 0.6(A_{t1} / A_a)$.

8118

8119 If the computed fractional credit is less than 0.75, the process is regarded as unsuitable and
8120 should be given no credit.

8121

8122 8.4.13.3 Qualifying the Neutron Absorber Material Fabrication Process

8123

8124 Qualification tests should be conducted at least once for a given manufacturing process and set
8125 of material specifications to demonstrate the quality and durability of the resulting neutron
8126 absorber product over its licensed service life. The full set of qualification tests should ensure
8127 the fabrication process results in an absorber material that meets or exceeds the service
8128 requirements. The following guidance discusses important material characteristics that can be
8129 demonstrated through tests.

8130

8131 Qualification testing should address the potential for damage under all service conditions of
8132 gamma-ray and neutron irradiation, temperature, material chemistry, and imposed stress over
8133 the licensed system lifetime. Test samples should be examined [i.e., the use of transmission
8134 electron microscopy (TEM) or scanning electron microscopy (SEM)] for the following changes:

8135

8136 • Redistribution or loss of boron.

8137

8138 • Dimensional changes (material instability).

8139

8140 • Cracking, spalling, or debonding of the matrix from the boron-containing
8141 particles.

8142

8143 • Weight changes caused by leaching, dissolution, corrosion, wear, or off-gassing.

8144

8145 • Embrittlement.

8146

8147 • Chemical changes such as oxidation or hydriding.

8148

8149 • Molecular decomposition of the material as a result of radiation (radiolysis).

8150

8151 Coupons should be taken so as to be representative of the neutron absorber. To the extent
8152 practical, test locations on coupons should be stratified to minimize errors due to location or
8153 position within the coupon. Some suggested locations should include the ends, corners,
8154 centers, and irregular locations. These locations represent the most likely areas to contain
8155 variances in thickness. Adequate numbers of samples should be taken from every other
8156 component (i.e., plate, rod, etc.) produced in a lot to obtain a good representation. A lot is
8157 defined as all plates from a single billet. Overall, the coupons should be a representative
8158 sample of the material.

8159

8160 For containers that will be loaded or unloaded in a SNF pool or similar environment, the
8161 reviewer should verify the absorber material has been evaluated or tested for environmental and
8162 galvanic interactions and the generation of hydrogen in the pool environment. If environmental
8163 testing is employed, the test conditions (time, temperature) should equal or exceed those
8164 expected for loading, unloading, and transfer operations. For environmental tests, the absorber
8165 materials should be coupled to dissimilar metals, as may be appropriate to the application. The
8166 environment may be borated or deionized water, as appropriate. The evaluation should also

8167 consider the effects of any residual pool water remaining in the container after removal from the
8168 pool.

8169
8170 Generally, for common engineering materials, an evaluation based upon consultation of a
8171 corrosion reference (galvanic series) should suffice for pool loading/unloading situations.
8172

8173 The reviewer should note the applicant must take appropriate measures to assess the strength
8174 or ductility of the material, depending on the structural requirements of the application.
8175

8176 Acceptance testing of the fabricated materials is discussed in Chapter 10, "Acceptance Tests
8177 and Maintenance Program Evaluation," of this SRP.
8178

8179 **8.4.14 Concrete and Reinforcing Steel (LOW Priority)**

8180 8.4.14.1 Embedment Materials

8181
8182 The materials discipline should review the material to be used for embedments, inserts,
8183 conduits, pipes, or other items embedded in the concrete. Embedments must satisfy the
8184 requirements of the code used in designing the reinforced concrete structure in which they are
8185 embedded (e.g., ACI 359, ACI 349, or ACI 318). Zinc, zinc rich coatings, zinc-clad materials,
8186 and aluminum should not be used for any embedded objects that will be in contact with wet
8187 concrete, because of the potential for concrete degradation from an adverse chemical reaction.
8188 Embedments and attachments are considered to include components cast or grouted into the
8189 reinforced concrete structure, inserts, embedded pipes, conduits, or lightning protection and
8190 grounding systems.
8191

8192
8193 Unless otherwise specified in this SRP, steel structural attachments must comply with the
8194 appropriate requirements of ACI-349.
8195

8196 8.4.14.2 Concrete Temperature Limits

8197
8198 The NRC accepts the use of ACI 318 for the design and material specifications for reinforced
8199 concrete structures subject to NRC approval, but are not important to safety. If ACI 349 is used
8200 for design of such structures, the NRC accepts the use of ACI 318 for construction. The NRC
8201 also accepts the following criteria as an alternative to the temperature requirements of ACI 349
8202 Section A.4, but only for the specified used and temperature ranges:
8203

8204 1. Concrete temperatures in general or local areas are a maximum of 93°C (200°F)
8205 in normal or off-normal conditions and/or occurrences, no tests are needed to
8206 prove capability for elevated temperatures or reduced concrete strength.
8207

8208 2. If concrete temperatures in general or local areas exceed 93°C (200°F) but are
8209 less than 149°C (300°F), no tests are required to prove capability for elevated
8210 temperatures or reduced concrete strength if Type II cement is used and
8211 temperature appropriate aggregates are used. The following criteria for fine and
8212 coarse aggregates are acceptable:
8213

8214 - Satisfy ASTM C33 requirements and requirements references in ACI
8215 349 for aggregates, and
8216

- 8217 - Have a demonstrated coefficient of thermal expansion (tangent in
8218 temperature range of 20-38°C (70-100°F) no greater than 11×10^{-6}
8219 mm/mm/°C (6×10^{-6} in./in./°F), or be one of the following materials:
8220 limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.
8221
- 8222 • If concrete temperatures in general or local areas under normal or off-normal
8223 conditions do not exceed 107°C (225°F), the requirements of 1 and 2 (above)
8224 apply to the coarse aggregate. Fine aggregate that meets 1 (above) and is also
8225 composed of quartz sands or sandstone sands may be used in place of 2
8226 (above) and be in compliance.
8227

8228 Water-to-cement ratios and use of air entraining materials are left to the discretion of the
8229 constructor.

8230

8231 8.4.14.3 Omission of Reinforcement

8232

8233 Frequently, designers specify the omission of reinforcing steel (“rebar”) in concrete above-
8234 ground structures which have the purpose of gamma shielding only. This is acceptable since it
8235 is to avoid the inadvertent formation of voids in the concrete due to the presence of the rebar,
8236 which can act to block the aggregate in the concrete from filling all intended areas.

8237

8238 Concrete applied around buried steel structures should be reinforced to alleviate shrinkage
8239 crack propagation. Concrete alleviates soil corrosion by creating a beneficial chemical buffering
8240 effect (high pH) around the steel. Cracks allow groundwater plus electrolyte intrusion which
8241 reduces the effectiveness of the concrete protective barrier.

8242

8243 **8.4.15 Seals**

8244

8245 Applicants for spent fuel storage canisters with metallic seals generally rely on seal
8246 manufacturer’s data to determine the maximum service temperatures for seals. Seals that may
8247 potentially be exposed to high temperature may not have been tested by independent
8248 laboratories (such as NIST and Factory Mutual). Due to the importance of the integrity of the
8249 seals, laboratory test results or data sheets that reference independent test results should be
8250 included in applications, if available.

8251

8252 8.4.15.1 Metallic Seals (MEDIUM Priority)

8253

8254 Bolted lid canisters employ redundant metallic seals as part of the confinement boundary.
8255 These seals are ITS components. The primary materials issue is the temperature resistance of
8256 the seal spring material. Generally this is a nickel-base alloy with excellent temperature and
8257 creep resistance. The seal cover material may be soft aluminum or silver. Aluminum faced
8258 seals have failed in service due to corrosion from inadvertent rainwater intrusion. Substitution of
8259 silver alloy faced seals appears to have alleviated the susceptibility of mechanical seals to this
8260 corrosion-induced failure mechanism.

8261

8262 8.4.15.2 Elastomeric Seals (LOW Priority)

8263

8264 Bolted lid canister designs may also employ a weather cover to preclude rainwater from the
8265 confinement boundary seals. These weather covers may be sealed against the weather with an
8266 elastomeric seal such as Viton. As such, these seals may be susceptible to thermally and
8267 radiation induced aging (hardening). Consequently, a replacement program may be warranted

8268 if the heat or radiation exposure is sufficient. Guidance as to radiation or thermal resistance is
8269 usually obtainable from the seal manufacturer. Elastomeric seals have never been ITS
8270 components in storage canisters.

8271
8272 Radiation will generally cause polymerization of elastomers to an extent that would adversely
8273 affect the performance when the dose reaches 10^5 Gy (10^7 rads). For higher dose rate
8274 environments, elastomer O-rings should not be specified. The use of fluorocarbons, which are
8275 known to be particularly susceptible to radiation damage, should be restricted if the expected
8276 dose exceeds 100 Gy (10^4 rads).

8277
8278 The reviewer should verify O-ring seals do not reach their maximum operating temperature limit
8279 during normal and off-normal conditions of storage. The O-ring manufacturer's data sheets
8280 specifying temperature and radiation tolerances should be included in the SAR.

8281
8282 The materials discipline should review the applicant's evaluation demonstrating the minimum
8283 normal operating temperature (usually -40°F) will neither fail the O-ring seal by brittle fracture
8284 nor stiffen the O-ring (lose elasticity) to an extent that prevents the seal from meeting its
8285 service requirements.

8286
8287 The reviewer should verify that under the environmental conditions expected in storage service,
8288 O-ring seals will not chemically react or decompose in a manner that would significantly affect
8289 other components of the DSS.

8290
8291 **8.4.16 Low Temperature Ductility and Fracture Control of Ferritic Steels**
8292 **(MEDIUM Priority)**
8293

8294 Regulatory Guides 7.11 and 7.12 specify acceptable ferritic steels for low temperature service
8295 where good toughness is required. Austenitic stainless steels are immune to low temperature
8296 toughness/ductility loss and thus are not a concern in this regard.

8297
8298 For designs that specify ferritic steels other than those listed in Reg. Guides 7.11 and 7.12, the
8299 Reg. Guide specifies the types of tests and data needed to qualify a material. Those tests and
8300 data include dynamic fracture toughness and nil-ductility or fracture appearance transition
8301 temperature test data. Samples are normally required of weld metal, heat-affected zone (HAZ),
8302 and base materials having been taken from welds that use the same thicknesses and materials
8303 of construction and welding procedures as used for construction.

8304
8305 **8.4.17 Cladding**
8306

8307 (MEDIUM Priority) This guidance will allow all commercial spent fuel that is currently licensed by
8308 the Nuclear Regulatory Commission (NRC) for commercial power plant operations to be stored
8309 in accordance with the regulations contained in 10 CFR Part 72. However, cask vendors'
8310 requests for the storage of spent fuel with burnup levels in excess of those levels licensed by
8311 the Office of Nuclear Reactor Regulation (NRR), or for cladding materials not licensed by NRR,
8312 may require additional justifications by the applicant.

8313
8314 The most important issues regarding spent fuel and cladding that must be considered are:

- 8315
8316 • The maximum cladding temperature during loading/unloading operations and
8317 normal conditions of storage. For high burn-up fuel, defined as any fuel with a
8318 burn-up greater than 45GWd/MTU, the maximum allowable cladding temperature

8319 limit is 400°C. The maximum fuel burn-up is to be specified as the peak rod
8320 average.

8321
8322 • Compatibility of fuel bundle materials and non-fuel component materials such as
8323 burnable poison rod assemblies (BPRAs) with the loading/unloading environment
8324 and the cask interior components. Refer to the separate discussion of this in
8325 Section 8.4.8.1.

8326
8327 • The fuel is maintained in a water or inert environment during loading/unloading
8328 operations to prevent excessive oxidation of fuel pellets. This is discussed in
8329 more detail in Section 8.7 of this SRP.

8330
8331 • A definition of damaged fuel is adequate for the intended fuel load and fuel with
8332 more severe damage (if any) is precluded from loading.

8333

8334 8.4.17.1 Cladding Temperature Limits (MEDIUM Priority)

8335
8336 The requirements of 10 CFR 72.122(h)(1) seek to ensure safe fuel storage and handling and to
8337 minimize post-operational safety problems with respect to the removal of the fuel from storage.
8338 In accordance with this regulation, the spent fuel cladding must be protected during storage
8339 against degradation that leads to gross rupture of the fuel and must be otherwise confined such
8340 that degradation of the fuel during storage will not pose operational problems with respect to its
8341 removal from storage. Additionally, 10 CFR 72.122(l) and 72.236(m) require that the storage
8342 system be designed to allow ready retrieval of the spent fuel from the storage system for further
8343 processing or disposal.

8344
8345 Spent fuel storage casks and systems must be designed to meet four safety objectives:

8346
8347 • Ensure doses from the spent fuel in the casks and systems are less than limits
8348 prescribed in the regulations.

8349
8350 • Maintain subcriticality under all credible conditions.

8351
8352 • Ensure there is adequate confinement and containment of the spent fuel under
8353 all credible conditions of storage.

8354
8355 • Allow the ready retrieval of the spent fuel from the storage systems.

8356
8357 The regulations that underpin these objectives will continue to be the foundation from which
8358 safety is ensured for the storage of spent fuel at all burnup levels. The following Part 72
8359 regulations pertain to the configuration control of spent fuel under various conditions of storage.

8360
8361 These acceptance criteria and review procedures are designed to provide reasonable
8362 assurance the spent fuel is maintained in the configuration analyzed in the storage SARs.
8363 These criteria are applicable to all commercial spent fuel burnup levels and cladding materials.
8364 In order to assure integrity of the cladding material, the following criteria should be met:

8365
8366 • For all fuel burnups (low and high), the maximum calculated fuel cladding
8367 temperature should not exceed 400°C (752°F) for normal conditions of storage
8368 and short-term loading operations (e.g., drying, backfilling with inert gas, and

8369 transfer of the cask to the storage pad). However, for low burnup fuel, a higher
8370 short-term temperature limit may be used, if the applicant can show by
8371 calculation the best estimate cladding hoop stress is equal to or less than
8372 90 MPa (13,053 psi) for the temperature limit proposed.

- 8373
- 8374 • During loading operations, repeated thermal cycling (repeated heatup/cooldown
8375 cycles) may occur but should be limited to less than 10 cycles, where cladding
8376 temperature variations are more than 65°C (117°F) each.
- 8377
- 8378 • For off-normal and accident conditions, the maximum cladding temperature
8379 should not exceed 570°C (1058°F).
- 8380

8381 Given the conservatism used in calculating peak clad temperatures for low burnup fuel, the staff
8382 has reasonable assurance that storage cask systems which use the 570°C temperature limit for
8383 low burnup fuel loading operations will continue to perform as expected when the casks were
8384 originally certified. Therefore, there is no need to require the licensees of storage-only or dual-
8385 purpose cask systems to repackage spent fuel loaded using the 570°C temperature limit.

8386

8387 The maximum allowable temperature should be based upon the peak rod temperature, not the
8388 average rod temperature. By employing the peak rod temperature, only a small fraction of the
8389 rods will experience the temperature and stress conditions that could lead to the formation of
8390 radial hydrides during normal conditions of storage.

8391

8392 High burnup fuel (i.e., fuel with burnups generally exceeding 45 GWd/MTU) may have cladding
8393 walls that have become relatively thin from in-reactor formation of oxides or zirconium hydride.
8394 For design basis accidents, where the structural integrity of the cladding is evaluated, the
8395 applicant should specify the maximum cladding oxide thickness and the expected thickness of
8396 the hydride layer (or rim). Cladding stress calculations should use an effective cladding
8397 thickness that is reduced by those amounts. The reviewer should verify that the applicant has
8398 used a value of cladding oxide thickness that is justified by the use of oxide thickness
8399 measurements, computer codes validated using experimentally measured oxide thickness data,
8400 or other means that the staff finds appropriate. Note that oxidation may not be of a uniform
8401 thickness along the axial length of the fuel rods.

8402

8403 Since the hoop stress is dependent on the rod internal pressure, cladding geometry, and the
8404 temperature of the gases inside the rod, the staff will verify that the applicant has calculated the
8405 best estimate hoop stress corresponding to the rod internal pressure of the highest burnup fuel
8406 assemblies of the specific type of assembly.

8407

8408 The intent of the thermal cycling acceptance criteria is to prevent licensees from applying cask
8409 drying, loading and transfer operations that could inadvertently enhance an undesirable hydride
8410 reorientation to form radial hydrides. Accordingly, these criteria pertain only to periods of fuel
8411 loading and transfer operations of the casks to the storage pads.

8412

8413 In general, the materials reviewer should coordinate with the structural reviewer to assure the
8414 spent fuel is maintained in the configuration analyzed in the Safety Analysis Reports (SARs) in
8415 order to meet the objectives described above.

8416

8417 The materials reviewer should coordinate with the thermal reviewer to assure the temperature
8418 criteria stated above are met. If higher peak temperatures are proposed by the applicant,
8419 additional justification for the higher temperatures must be supplied.

8420
8421 This guidance will allow all commercial spent fuel that is currently licensed by the Nuclear
8422 Regulatory Commission (NRC) for commercial power plant operations to be stored in
8423 accordance with the regulations contained in 10 CFR Part 72. However, cask vendors' requests
8424 for the storage of spent fuel with burnup levels in excess of those levels licensed by the Office of
8425 Nuclear Reactor Regulation (NRR), or for cladding materials not licensed by NRR, may require
8426 additional justifications by the applicant.

8427
8428 Background justification for these temperature limits can be found in Sec 8.8 of this SRP.
8429

8430 8.4.17.2 Fuel Classification (HIGH Priority)

8431
8432 The staff should verify that the definitions below are used in the SAR, and where appropriate
8433 are also included in the CoC.

8434
8435 Spent Nuclear Fuel (SNF) - See 10 CFR Part 72.3 for definition. This term has been used in the
8436 nuclear industry, at different times, to mean the fuel pellets, the rod, or entire fuel assembly.
8437 Unless specifically modified, the term will refer to both the rods and fuel assembly.

8438
8439 Damaged SNF - Any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system-
8440 related functions.

8441
8442 Undamaged SNF - SNF that can meet all fuel-specific and system-related functions. As
8443 shown in Figure 8-2, undamaged fuel may be breached. Fuel assembly classified as
8444 undamaged SNF may have "assembly defects."

8445
8446 Breached spent fuel rod - Spent fuel rod with cladding defects that permit the release of gas
8447 from the interior of the fuel rod. A breached spent fuel rod may also have cladding defects
8448 sufficient to permit the release of fuel particulate. A breach may be limited to a pinhole leak,
8449 hairline crack, or may be a gross breach.

8450
8451 Pinhole leaks or hairline cracks - Minor cladding defects that will not permit significant release of
8452 particulate matter from the spent fuel rod, and therefore present a minimal as low-as-is-
8453 reasonably-achievable concern, during fuel handling and retrieval operations. (See discussion
8454 of gross defects for size concerns.)

8455
8456 Grossly breached spent fuel rod - A subset of breached rods. A breach in spent fuel cladding
8457 that is larger than a pinhole leak or a hairline crack. An acceptable examination for a gross
8458 breach is a visual examination that has the capability to determine the fuel pellet surface may be
8459 seen through the breached portion of the cladding. Alternatively, review of reactor operating
8460 records may provide evidence of the presence of heavy metal isotopes indicating that a fuel rod
8461 is grossly breached. (See discussion for size concerns.)

8462
8463 Intact SNF - Any fuel that can fulfill all fuel-specific and system-related functions, and that is not
8464 breached. Note that all intact SNF is undamaged, but not all undamaged fuel is intact, since
8465 under most situations, breached spent fuel rods that are not grossly breached will be considered
8466 undamaged.

8467
8468 Can for Damaged Fuel - A metal enclosure that is sized to confine one damaged spent fuel
8469 assembly. A fuel can for damaged spent fuel with damaged spent-fuel assembly contents must

8470 satisfy fuel-specific and system-related functions for undamaged SNF required by the applicable
8471 regulations.

8472
8473 Assembly Defect - Any change in the physical as-built condition of the assembly with the
8474 exception of normal in-reactor changes such as elongation from irradiation growth or assembly
8475 bow. Examples of assembly defects: (a) missing rods; (b) broken or missing grids or grid
8476 straps (spacers); and (c) missing or broken grid springs, etc. An assembly with a defect is
8477 damaged only if it can't meet its fuel-specific and system-related functions required by the
8478 applicable regulations.

8479
8480 A fuel-specific regulation - a characteristic or performance requirement of the fuel specifically
8481 named in the applicable Code of Federal Regulations (CFR). These are regulations that specify
8482 capabilities that the spent nuclear fuel (SNF) must have. Examples include 10 CFR
8483 72.122(h)(1) and 10 CFR 72.122(l).

8484
8485 A system-related regulation - a performance requirement placed on the fuel so that the storage
8486 system can meet its regulatory requirements. Examples include 10 CFR 72.122(h)(5) and
8487 10 CFR 72.124(a)..

8488
8489 Previous definitions of damaged fuel have identified specific characteristics of the fuel that
8490 classify it as damaged, irrespective of whether the fuel is being stored or transported and
8491 independent of the design of the storage or transportation system. The current staff position is
8492 that damaged fuel is defined in terms of the characteristics needed to perform the fuel-specific
8493 and system-related functions. The materials properties, and possibly the physical condition, of
8494 a fuel rod or assembly can be altered during irradiation or storage. If this alteration is large
8495 enough to prevent the fuel or assembly from performing its fuel-specific or system-related
8496 functions during storage, then the fuel assembly is considered damaged.

8497
8498 To determine whether a fuel assembly is undamaged, the following should be stated in the
8499 SAR:

- 8500
- 8501 1) The functions the applicant has imposed on the fuel rods and assembly by either fuel
8502 specific or system-related functions to meet a regulatory requirement for the designated
8503 phase (storage, transportation, or both);
 - 8504
 - 8505 2) The mechanisms of change (alteration mechanisms) or the characteristics of the fuel
8506 that could potentially cause the fuel to fail to meet its fuel-specific or system-related
8507 functions;
 - 8508
 - 8509 3) An acceptable analysis showing that the fuel with the designated characteristics will
8510 meet the fuel-specific and system-related functions when the mechanisms considered in
8511 item #2, above, are evaluated; and
 - 8512
 - 8513 4) The physical characteristics of the fuel, based on item #3, above, that could cause the
8514 fuel or assembly to be classified as "damaged."

8515
8516 A "default" definition of damaged SNF, derived from ANSI N14.33-2005, is provided for those
8517 that do not want to perform the assessment outlined in item numbers 1 through 4 above. The
8518 default definition, however, may not take full advantage of the flexibility of the performance-
8519 based definition of damaged fuel provided in this guidance. This default definition may be more
8520 restrictive than necessary, depending on the design of the storage or transportation cask. For

8521 example, the default definition of damaged SNF indicates that SNF must be classified as
8522 damaged if an individual fuel rod is missing from an assembly. However, if an analysis shows
8523 that all fuel-specific and system-related functions will be met (e.g., subcriticality will be
8524 maintained, that the SNF assembly will be retrievable and that the structural properties of the
8525 assembly are not compromised by the missing rod) the assembly may be classified as
8526 undamaged. An alternative default definition of damaged Spent Nuclear Fuel (SNF) is: SNF
8527 assemblies must be classified as damaged if any one of the following conditions exist:
8528

8529 On removal of SNF selected for dry storage or transport from the spent fuel pool, any of the
8530 following apply:

- 8531
- 8532 • There is visible deformation of the rods in the SNF assembly. Note: This is not
8533 referring to the uniform bowing that occurs in the reactor. This refers to bowing
8534 that significantly opens up the lattice spacing.
8535
 - 8536 • Individual fuel rods are missing from the assembly. Note: The assembly may be
8537 reclassified as intact if a dummy rod that displaces a volume equal to, or greater
8538 than, the original fuel rod, is placed in the empty rod location.
8539
 - 8540 • The SNF assembly has missing, displaced, or damaged structural components
8541 such that either:
8542
 - 8543 a. Radiological and/or criticality safety is adversely affected (e.g.,
8544 significantly changed rod pitch).
8545
 - 8546 b. The assembly cannot be handled by normal means (i.e., crane and
8547 grapple).
8548
 - 8549 • Reactor operating records (or other records) indicate that the SNF assembly
8550 contains fuel rods with gross breaches.
8551
 - 8552 • The SNF assembly is no longer in the form of an intact fuel bundle (e.g., consists
8553 of, or contains, debris such, as loose fuel pellets or rod segments).
8554

8555 Additional background and examples of defining damaged fuel can be found in Section 8.6 of
8556 this SRP.

8557
8558 **8.4.17.3 Reflood Analysis (HIGH Priority)**
8559

8560 The NRC accepts that the total stress on the cladding is maintained below the material's
8561 minimum yield stress. The total stress includes the thermal stress combined with the cladding
8562 hoop stress from internal rod pressure and the rod-gas plenum temperature. The analysis also
8563 should account for high burnup effects on the fuel (e.g., waterside corrosion, high internal rod
8564 pressure) and minimum manufacturing wall thickness. Other assembly components should also
8565 be examined in a similar manner. Engineering judgment, combined with relevant industry
8566 operational experience with unloading SNF from transportation and storage casks, may support
8567 the basis for limits on quench fluid temperature and flow rate. This review should be
8568 coordinated with the materials reviewer.
8569

8570 **8.4.18 Prevention of Oxidation Damage During Loading of Fuel (MEDIUM Priority)**
8571

8572 The guidance in this section is only applicable to irradiated LWR fuel or other uranium oxide
8573 based fuel. The reviewer should make sure that the oxidation of other types of fuels during
8574 loading is evaluated. The information given in this section and Section 8.7 of this SRP may not
8575 be applicable to other fuel types. The characteristics of those fuel types must be considered
8576 when evaluating their analysis.
8577

8578 Once the fuel rods are placed inside of the storage cask and water is removed to a level that
8579 exposes any part of the rods to a gaseous atmosphere, reasonable assurance the spent fuel
8580 cladding will be protected against splitting due to fuel oxidation might occur must be
8581 demonstrated. If oxidation occurred, it may lead to loss of retrievability, or to a configuration not
8582 adequately analyzed for radiation dose rates or criticality. Further, the release of fuel fines or
8583 grain-sized powder into the inner cask environment from ruptured fuel may be a condition
8584 outside the licensing basis for the cask system. Three possible options exist to address the
8585 potential for and consequences of fuel oxidation:
8586

- 8587 1. Maintain the fuel rods in an appropriate environment such as Ar, N₂, or He to prevent
8588 oxidation.
8589
- 8590 2. Assure there are not any cladding breaches (including hairline cracks and pinhole leaks)
8591 in the fuel pin sections that will be exposed to an oxidizing atmosphere. This can be
8592 done by a review of records (for example, sipping records) or 100 percent eddy current
8593 inspection of assemblies. Note that inspection of rods by either eddy current or visual
8594 inspection, to the extent needed to assure there are no pinholes or hairline cracks is
8595 difficult, time consuming, and subject to error.
8596
- 8597 3. Determine the time-at-temperature profile of the rods while they are exposed to an
8598 oxidizing atmosphere and calculate the expected oxidation to determine if a gross
8599 breach would occur. The analysis should indicate the time required to incubate the
8600 splitting process will not be exceeded. Such an analysis would have to address
8601 expected differences in characteristics between the fuel to be loaded and the fuel tested
8602 to determine the basis for the analysis. Conversely, the maximum allowable
8603 temperature of the rods could be limited to the temperature that calculations show
8604 cladding splitting will not be expected to occur. Such evaluations must incorporate the
8605 effects of uncertainty in the data base. Calculation of the possibility of cladding splitting,
8606 is fraught with all the uncertainties discussed above. Lowering the maximum allowable
8607 temperature may impose an economic penalty by limiting the heat load in the cask. The
8608 selection of the methodology used to address this issue is up to the applicant. The use
8609 of a non-oxidizing atmosphere in the fuel canister to prevent fuel oxidation is one method
8610 accepted by the staff to address the issue.
8611

8612 If Option 3 is chosen, the materials reviewer should coordinate with the thermal reviewer to
8613 determine that the operating procedures, technical specification, and associated licensing
8614 documentation, as submitted by the applicants, provide a supportable analysis of the potential
8615 for cladding splitting, should fuel rods be exposed to an oxidizing gaseous atmosphere. For fuel
8616 with burnup below ~45 GWd/MTU and Zircaloy cladding, the time-at-temperature (TT) curves
8617 developed to date (R.E. Einziger and R.V. Strain, "Oxidation of Spent Fuel at Between 250° and
8618 360°C," EPRI Report NP-4524, 1986, for example) can be used to determine the allowable
8619 exposure duration to an oxidizing atmosphere if the fuel temperature is known, or conversely
8620 the maximum allowable temperature if the exposure time is known. For example, using
8621 Figure 3-9 of the above reference, at 360°C one would expect to incur splitting between 2 and

8622 10 hours. On the other hand, if one expected to stay at temperature for 100 hours then the fuel
8623 temperature should be kept below 290°C.

8624
8625 Additional information on oxidation of damaged fuel can be found in Section 8.7 of this SRP.
8626 Please refer to this reference for additional detail and background.

8627
8628 *** 8.4.19 Flammable Gas Generation (MEDIUM Priority)**
8629

8630 The reviewer should assume the generation of hydrogen or other gases during wet
8631 loading/unloading operations occurs. Field experience has amply demonstrated that any
8632 canister design employing aluminum components as part of the fuel basket construction will
8633 have a propensity to generate hydrogen. Efforts to passivate the aluminum components have
8634 proven inadequate to eliminate the generation of hydrogen. The use of zinc, zinc-rich coatings,
8635 or zinc-clad materials (e.g., galvanized steel) in particular, is known to generate potentially
8636 dangerous quantities of hydrogen gas during wet-loading in SFP.

8637
8638 * Consequently, the reviewer should verify the operating procedures contain adequate guidance
8639 for detecting the presence of hydrogen and preventing the ignition of combustible gases during
8640 cask loading and unloading operations. These procedures must be incorporated by reference
8641 into the TS.

8642
8643 *** 8.4.20 Helium Leakage Testing (MEDIUM Priority)**
8644

8645 Helium leakage testing of the entire confinement boundary is performed to assure various
8646 attributes of the confinement boundary:

- 8647
- 8648 • The fuel payload is protected from the deleterious oxidizing effects of moisture by
8649 excluding intrusion of such.
 - 8650
 - 8651 • Ensure the helium inerting gas will remain in the canister in sufficient amount
8652 over the license period.
 - 8653
 - 8654 • Ensure the helium gas heat transfer medium will remain in sufficient quantity over
8655 the license period to assure the cladding temperatures are controlled at safe
8656 levels.

8657
8658 This guidance addresses all welds associated with the redundant closures of a spent fuel
8659 canister and describes how each individual closure weld must be considered from the overall
8660 design and testing standpoint. It only applies to canisters of all-welded construction, fabricated
8661 from austenitic stainless steel, employing redundant welds for the confinement closure.

8662
8663 The staff should verify that the cask design under review is in compliance with the guidance of
8664 this document. In order for any closure weld to be exempt from the helium leak testing to
8665 demonstrate compliance with 10 CFR 72.236, the staff should verify all of the following
8666 conditions are satisfied:

- 8667
- 8668 • The welded canister (i.e., the confinement boundary) must be leak tested in
8669 accordance with ANSI N14.5-1997, except as specified by this guidance.
 - 8670
 - 8671 • Closure welds must conform with the guidance of this SRP, as appropriate.

8672

- 8673 • “Structures, systems, and components important to safety must be designed to
8674 withstand postulated accidents.” [10 CFR 72.122(b)(1)].
- 8675
- 8676 • Records documenting the lid welds shall comply with the provisions of 10 CFR
8677 Part 72.174, “Quality Assurance Records.” Records storage should comply with
8678 ANSI N45.2.9, “Requirements for Collection, Storage, and Maintenance of
8679 Quality Assurance Records for Nuclear Power Plants.”
- 8680
- 8681 • Activities related to inspection, evaluation, documentation of fabrication, and lid
8682 welding shall be performed in accordance with an NRC-approved quality
8683 assurance program as required in 10 CFR Part 72, Subpart G, “Quality
8684 Assurance.”
- 8685

8686 In addition for exemption of large multi-pass welds from helium leak testing the following must
8687 be satisfied.

- 8688
- 8689 (1) The weld must be multi-pass, with a minimum weld depth comprised of at least 3
8690 distinct weld layers.
- 8691
- 8692 (2) Each layer of weld may be composed of one or more adjacent weld beads.
- 8693
- 8694 (3) The layer must be complete across the width of the weld joint.
- 8695
- 8696 (4) If only 3 weld layers comprise the full thickness of the weld, each layer must be
8697 PT examined.
- 8698
- 8699 (5) For more than 3 weld layers, not all weld layers need be PT examined. The
8700 maximum weld deposit depth allowed before a PT examination is necessary is
8701 based upon flaw-tolerance calculations in accordance with Section 8.9 of this
8702 SRP. Note: This criteria does not supersede the flaw acceptance criteria of any
8703 construction code. Instead, this criteria is used to establish the maximum
8704 allowable weld deposit depth before an in-process PT examination is necessary.
- 8705
- 8706 (6) Regardless of conditions (4) or (5) above, at least 3 different weld layers must be
8707 examined, e.g., the root pass, a mid-layer, and the cover pass.
- 8708
- 8709 (7) The weld cannot have been executed under conditions where the root pass
8710 might have been subjected to pressurization from the helium fill in the canister
8711 itself. Credit may not be taken for closure valves, quick-disconnects, or similar.
8712 It is assumed that mechanical closure devices (e.g., a valve or quick-disconnect)
8713 permit helium leaks. Practical experience has shown such leaks occur and have
8714 been responsible for causing leak paths through the weld. Consequently, welds
8715 potentially subjected to helium pressure (by way of leakage through a
8716 mechanical closure device) during the welding process must be subsequently
8717 helium leak tested.
- 8718

8719 Other closure issues the materials reviewer should evaluate are: Hydrostatic Testing, ASME
8720 Code Case N-595-4, and the limiting root pass criteria for the weld.

8721

8722 Closure welds must be hydrostatically or pneumatically tested in accordance with ASME Code
8723 Section III requirements to the extent practicable. The two designs discussed in Section 8.9 of
8724 this SRP meet this criteria.

8725
8726 ASME Code Case N-595-4 is not endorsed by the NRC staff, per Regulatory Guide (RG) 1.183
8727 and consequently is not permitted as an alternative to the Code requirements.

8728
8729 Cask lid welding is governed in part by the limiting flaw size analysis. The welding method
8730 described herein controls the depth of weld deposit for the intermediate passes before the
8731 required PT examination is performed. However, the root pass thickness is not addressed by
8732 this guidance, as a single layer root pass was assumed. Occasionally, multi-layer root passes
8733 are employed to smooth the weld surface to avoid false positives from the PT.

8734
8735 A multi-layer root pass is acceptable provided the same method of limiting the weld deposit
8736 depth is followed as for the intermediate weld passes. Stress analysts should note that the
8737 intermediate layer critical flaw size calculation assumes a buried flaw, not a surface connected
8738 flaw. For the root pass calculation, a surface connected flaw must be assumed. This will result
8739 in a smaller critical flaw size, and, consequently a smaller permissible weld deposit thickness
8740 before a PT exam is considered necessary.

8741
8742 The staff should verify that if the licensee desires to use a thicker root pass, they must limit the
8743 amount of weld deposit to the ratio of the fracture toughness K values (or, J values) for the
8744 different flaw types (buried K divided by surface K) multiplied by the maximum depth. This will
8745 limit the depth of the root pass to the critical flaw size for a surface connected flaw. Thus, if a
8746 licensee desires to use a thicker weld deposit for the root pass, then a limiting flaw size analysis
8747 establishes a structural basis.

8748
8749 The staff recognizes that for stainless steel, K, or even J, is not entirely correct for evaluating
8750 failure in austenitic stainless steel due to the large capacity for plastic deformation. Generally
8751 the result is failure due to net section stress, not fracture. However, the stress intensity ratio
8752 suggested above is acceptable for this purpose.

8753
8754 The regulatory requirements governing this review are: 10 CFR 72.122(a), 72.122(h)(5),
8755 72.104(a), 72.106(b), 72.236(d), 72.236(e), 72.236(j), and 72.236(l).

8756
8757 Please refer to the additional information in Section 8.9 of this SRP to supplement the review
8758 criteria.

8759
8760 **8.4.21 Periodic Inspections (LOW Priority)**

8761
8762 Review the SAR operations and acceptance testing chapters for appropriate periodic inspection
8763 programs which may be included for the purpose of monitoring materials conditions or
8764 performance. Some cask vendors are now anticipating future license renewal for the designs
8765 and are incorporating into the SAR the currently specified limited inspections that are required
8766 as part of a license renewal application.

- 8767
8768
 - A one-time inspection of normally inaccessible portions of the canister exterior for
8769 unanticipated corrosion or other degradation. A single canister (or several) may
8770 be selected based upon engineering criteria such as longest time in service,
8771 hottest operating temperature, etc. and used to “bound” other canisters of that
8772 type of material of construction.

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- The periodic (usually monthly) ISFSI radiation survey results should be reviewed to determine if any significant degradation of any neutron shielding material (if used) has occurred.

8.5 Evaluation Findings

The evaluation findings are prepared by the reviewer on satisfaction of the regulatory requirements of Section 8.3. The reviewer should examine these requirements and provide a summary statement for each. These statements should be similar to the following examples:

- F8.1 Section(s) _____ of the SAR adequately describe(s) the materials used for SSCs important to safety and the suitability of those materials for their intended functions in sufficient detail to evaluate their effectiveness.
- F8.2 The applicant has met the requirements of 10 CFR 72.122(a). The material properties of SSCs important to safety conform to quality standards commensurate with their safety function.
- F8.3 The applicant has met the requirements of 10 CFR 72.104(a), 72.106(b), and 72.124. Materials used for criticality control and shielding are adequately designed and specified to perform their intended function.
- F8.4 The applicant has met the requirements of 10 CFR 72.122(h)(1) and 72.236(h). The design of the DSS and the selection of materials adequately protects the SNF cladding against degradation that might otherwise lead to damaged fuel.
- F8.5 The applicant has met the requirements of 10 CFR 72.236(h) and 72.236(m). The material properties of SSCs important to safety will be maintained during normal, off-normal, and accident conditions of operation so the SNF can be readily retrieved without posing operational safety problems.
- F8.6 The applicant has met the requirements of 10 CFR 72.236(g). The material properties of SSCs important to safety will be maintained during all conditions of operation so the SNF can be safely stored for a minimum of 20 years and maintenance can be conducted as required.
- F8.7 The applicant has met the requirements of 10 CFR 72.236(h). The [cask designation] employs materials that are compatible with wet and dry SNF loading and unloading operations and facilities. These materials should not degrade over time or react with one another during any conditions of storage.

The reviewer should provide a summary statement similar to the following:

“The staff concludes the material properties of the structures, systems, and components 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 material properties provides reasonable assurance the [cask designation] will allow safe storage of SNF for a licensed (certified) life of _____ years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.”

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8.6 Supplemental Information for Methods for Classifying Fuel (HIGH Priority)

A. Grossly Breached SNF Cladding

The regulations in 10 CFR 72.122(h) state “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.”

In dry cask storage and transportation systems, a gross cladding breach should be considered as any cladding breach that could lead to the release of fuel particulate greater than the average size fuel fragment. A pellet is ~1.1 centimeters in diameter in 15 x 15 Pressurized-Water Reactor (PWR) assemblies. Pellets from a Boiling-Water Reactor (BWR) are somewhat larger, and those from 17 x 17 PWR assemblies are somewhat smaller. The pellet's length is slightly longer than its diameter. During the first cycle of irradiation in-reactor, the pellet fragments into 25-35 smaller interlocked pieces, plus a small amount of finer powder, due to, pellet-to-pellet abrasion. When the rod breaches, about 0.1 gram of this fine powder may be carried out of the fuel rod at the breach site. Modeling the fragments as either spherical- or pie-shaped pieces indicates that a cladding-crack width of at least 2-3 millimeters would be required to release a fragment. Hence, gross breaches should be considered to be any cladding breach greater than 1 millimeter.

A review of reactor operating records, ultrasonic testing, and sipping (if done in a timely fashion) can be used to classify rods as unbreached or, breached. Evidence of only gaseous or volatile decay products (no heavy metals) in the reactor coolant system is accepted as evidence that a cladding breach is no larger than a pinhole leak or hairline crack. Records that show the presence of heavy metal isotopes that are characteristic of fuel release in the reactor coolant system indicate gross breaches in the cladding. Likewise, visual examination may also be used to determine if a cladding breach is gross, if the breached rod can be positively identified. Because cladding openings larger than 1 millimeter should expose the fuel pellet to visual sighting, visual examination of the breached rod can be used to determine if a breach is gross. However, visual examination is not an acceptable method of confirming intact (undamaged) fuel for assemblies that have indicated leakage.

It should be noted, however, that undamaged spent-fuel rods with pinhole leaks and/or hairline cracks will expose the fuel pellets to the canister or cask atmosphere. If that atmosphere is oxidizing, then the fuel pellet may oxidize and expand, placing stress on the cladding. The expansion may eventually cause a large split in the cladding, resulting in spent fuel that must be classified as damaged (for storage and possibly also for transportation) due to gross breaches in the cladding. Since fuel oxidation and cladding splitting follow Arrhenius time-at-temperature behavior, fuel rods with pinholes or hairline cracks that are exposed to an oxidizing atmosphere may experience this type of additional cladding damage. Section 8.7 of this SRP, “Supplemental Information for Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or other Uranium Oxide Based Fuel,” provides information regarding prevention of this phenomenon. Before handling undamaged rods with pinhole leaks and/or hairline cracks in an oxidizing atmosphere, the potential fuel and cladding degradation at the temperature of interest for the duration of the process should be assessed.

8874 B. Fuel Assembly with Defects

8875

8876 Damage under this guidance refers to alterations of the fuel assembly that prevent it from
8877 fulfilling its fuel-specific or system-related functions. Defects such as dents in rods, bent or
8878 missing structural members, small cracks in structural members, missing rods, etc., need not be
8879 considered damaged if the applicant can show that the fuel assembly with these defects still
8880 fulfils its fuel-specific and system-related functions. This may be done using calculations based
8881 on approved codes, situation-specific data, or reasoned engineering arguments.

8882

8883 C. Canning Damaged Fuel

8884

8885 Spent fuel that has been classified as damaged for storage must be placed in a can designed
8886 for damaged fuel, or in an acceptable alternative. The purpose of a can designed for damaged
8887 fuel is to (1) confine gross fuel particles, debris, or damaged assemblies to a known volume
8888 within the cask; (2) to demonstrate that compliance with the criticality, shielding, thermal, and
8889 structural requirements are met; and (3) permit normal handling and retrieval from the cask.
8890 The can designed for damaged fuel may need to contain neutron-absorbing materials, if results
8891 of the criticality safety analysis depend on the neutron absorber to meet the requirements of
8892 10 CFR 72.124(a).

8893

8894 D. Relationship of Spent Fuel Populations

8895

8896 The applicant will designate the population of spent fuel for which the cask system was
8897 designed (e.g., type of fuel, minimum cooling time, burnup limitations, arrays, manufacturers,
8898 cladding types, etc.). This population may contain breached rods. Some of these breached
8899 rods may be grossly breached. It may also contain assemblies with defects, such as missing
8900 rods, missing grid spacers, or damaged spacers. The populations of breached rods, grossly
8901 breached rods, and assemblies with defects are determined by in-reactor behavior and ex-
8902 reactor handling.

8903

8904 Each of these populations must be classified as damaged or undamaged after the storage or
8905 transportation system has been designated. For example, an applicant might propose the use
8906 of air as a cover gas in its design of a storage cask. The applicant might also propose this cask
8907 for use in storing spent fuel with cladding breaches that are hairline cracks or pinhole leaks.
8908 However, if the spent fuel in the cask will operate at a sufficiently high temperature for a long
8909 enough time, then oxidation of fuel pellets in breached rods could occur resulting in gross
8910 breaches. If this is the case, the breached spent fuel should be considered damaged because
8911 grossly breached rods do not meet the requirements of 10 CFR 72.122(h)(1). Also, in this case
8912 because the geometric form of the package contents could be substantially altered, the spent
8913 fuel would also be classified as damaged for transportation because the requirements of
8914 10 CFR 71.55(d)(2) might not be met. If an inert atmosphere was used instead of air, only
8915 grossly breached rods would be considered damaged for storage. This concept is illustrated in
8916 Figure 8-2, "Relationship of Spent Fuel Populations."

8917

8918 Example of Methodology

8919

8920 The following example is given to illustrate the general methodology. This is only an example of
8921 the methodology and should not be construed as approved characterization of damaged fuel.

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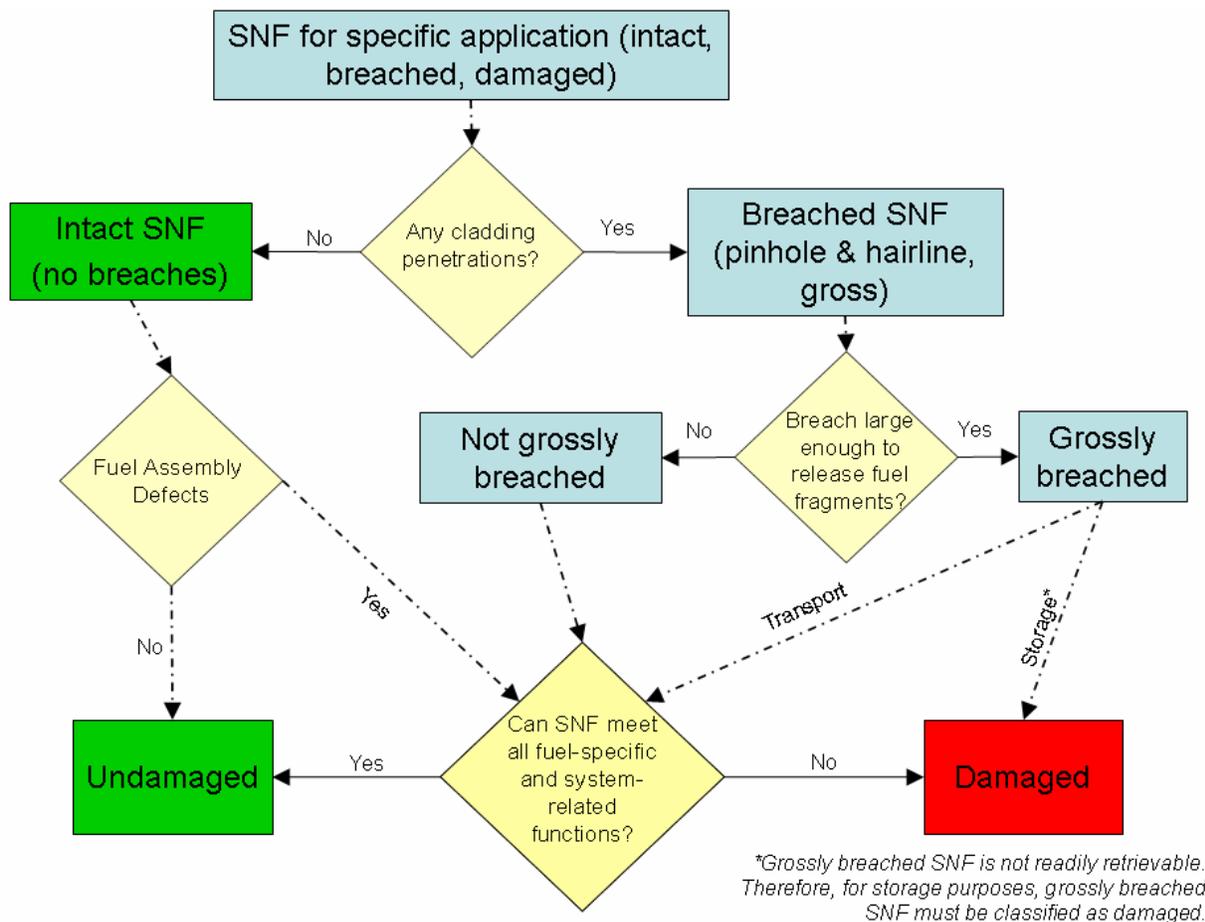


Figure 8-2 Relationship of Spent Fuel Populations

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Example of Methodology:

Situation - The vendor of a dual-purpose cask wants to store and transport low-burnup PWR fuel in an inert atmosphere and within the temperature limits recommended in Section 8.4.17.1. The vendor wants to store assemblies having rods with breaches containing only pinholes or hairline cracks, and assemblies having one or more outer grid straps with defects at three or more grid locations without canning them. The vendor is only applying for a storage license at this time but wants to be reasonably certain that the fuel will also be transportable.

Activity - Storage of Spent Fuel

Fuel-specific or system-related functions imposed on rods and assemblies - 10 CFR 72.122(h)(1), regarding gross ruptures, and 10 CFR 72.122(l), concerning retrievability, must be met for storage. 10 CFR 71.55(d), requiring the system to remain subcritical and unchanged during normal transport, must be met. The vendor believes that all the remaining system requirements, except for the subcriticality requirement, can be met, without imposing any limitations on the fuel, if the fuel is within the bounds stated in the situation.

Mechanisms - There are no mechanisms for the pinhole leaks and hairline cracks to evolve into gross breaches since the atmosphere is inert and the temperature is controlled. To be

8947 retrievable, the assemblies with missing grid straps must be able to withstand design basis
8948 events in a storage cask. Since the applicant also wants these assemblies to be considered
8949 undamaged for transportation, the behavior of the assemblies under both normal and
8950 hypothetical accident transportation conditions in 10 CFR Part 71 must be evaluated. For
8951 example, for normal transportation conditions, the applicant must show that the assemblies with
8952 the most missing grid straps in the worst locations can withstand both normal vibration and a
8953 one-foot drop and remain in their original physical configuration. Additionally, for hypothetical
8954 accident conditions, the analysis must indicate, among other things, that the system will meet
8955 shielding and subcriticality requirements when placed under the mechanical and thermal loads
8956 specified in 10 CFR Part 71.

8957
8958 Analysis - The applicant conducts an analysis to satisfactorily demonstrate that the assembly
8959 with three missing grid straps in the worst configuration remains intact for 1) normal
8960 transportation conditions; 2) cask tip-over; and 3) regulatory accident conditions. Further
8961 acceptable analysis indicates that all the system-related regulations are met, if the fuel with the
8962 characteristic limitations (as noted in Characteristics section below), stays structurally intact.

8963
8964 Characteristics - Assemblies containing breached rods with up to three grid straps missing will
8965 be considered undamaged for the purposes of storage. Analysis shows that these assemblies
8966 could probably also be considered undamaged for transportation, but fuel with these
8967 characteristics will be evaluated and approved as part of a later application for the transportation
8968 cask certification.

8969
8970 **8.7 Supplemental Information for Potential Rod Splitting Due to Exposure to an**
8971 **Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or**
8972 **Other Uranium Oxide Based Fuel (MEDIUM Priority)**
8973

8974 The definition of undamaged fuel includes fuel rods containing no cladding defects greater than
8975 pinhole leaks or hairline cracks. During the cask water removal process parts of, or all of, the
8976 fuel rods will be exposed to a gaseous atmosphere. If the gaseous atmosphere is oxidizing,
8977 oxidation of fuel pellets or fuel fragments can occur if a cladding breach exists (such as a
8978 pinhole). Oxidation may occur rapidly and cause significant swelling of fuel pellets and
8979 fragments, which could result in gross fuel cladding breaches if the time-at-elevated-
8980 temperature after water removal is excessive.

8981
8982 **8.7.1 Fuel Oxidation and Cladding Splitting**
8983

8984 Irradiated uranium dioxide exposed to an oxidizing atmosphere will eventually oxidize to U₃O₈.
8985 The time it takes to oxidize is a function of temperature that follows an Arrhenius function and
8986 burnup. However, at temperatures that may be expected for some spent fuel, this reaction can
8987 occur within a matter of hours.

8988
8989 The grain boundaries of irradiated fuel are highly populated with voids and gas bubbles. Initially
8990 the grain boundaries are oxidized to U₄O₉ resulting in a slight matrix shrinkage and further
8991 opening of the pellet structure. Oxidation then proceeds into the grain until there is complete
8992 transformation of the grains to U₄O₉ [Einziger, 1992]. The grains remain in this phase for a
8993 temperature dependent duration until the fuel resumes oxidizing to the U₃O₈ state. The
8994 transformation to U₃O₈ occurs with ~33 percent lattice expansion that breaks the ceramic
8995 fragment structure into grain sized particles. At higher temperatures, the two transformations
8996 occur so rapidly that they are difficult to distinguish. The mechanism of oxidation in irradiated
8997 fuel appears to be different than in unirradiated fuel where U₃O₇ is formed and oxidation

8998 proceeds from the fragment surface and not down the grain boundaries. This mechanistic
8999 change occurs between ~10 and 30 Gwd/MTU.

9000
9001 When the UO₂ is in the form of a fuel rod, the expansion of the fuel, when it transforms to
9002 U₃O₈, induces a circumferential stress in the cladding. Due to the swelling of the fuel, the
9003 process is usually initially localized to the original cladding crack site. The cladding strains due
9004 to this stress range from 2-6 percent before the initial crack starts to propagate along the rod.
9005 The incubation time to initiate the propagation and the rate of propagation have an Arrhenius
9006 temperature dependence. Axial propagation, spiral propagation and a combination of the
9007 modes that result in splitting have been observed in PWR rods [Einziger, 1986].
9008

8.7.2 Data Base

9009
9010
9011 The data base for oxidation was developed mostly in the 1980s in the US, Canada, England,
9012 and Germany. The data can usually appear in four forms: 1) O/M ratio (ratio of oxygen to metal
9013 content of the oxide) vs. time, 2) time to the UO_{2.4} plateau vs. time, 3) cladding splitting
9014 incubation vs. time, and 4) cladding splitting rate vs. time. Some later work was done by the
9015 Japanese on the effects of oxygen depletion [Nakamura, 1995], and most recently work is on-
9016 going by the French primarily on MOX fuel. Much of the work was done on unirradiated fuel. All
9017 the work on cladding splitting was done in the early 1980s by the US [Einziger, 1984, 1986;
9018 Johnson, 1984] and Canadians [Novak, 1984; Boase, 1977] and is limited. Recently DOE
9019 [Bechtel, 2005] has issued an analysis of the oxidation issue in relationship to handling of
9020 potentially breached fuel in their proposed handling facility at the repository. This analysis
9021 depends on variables such as the gap between the fuel and the cladding, and burnup in a
9022 manner that is currently under technical review. In total, this research has shown that there are
9023 a number of variables that can affect the rates at which the fuel oxidizes and the cladding splits:
9024 burnup, moisture content of the air, cladding material, and type of initial defect.
9025

9026 The DOE developed a model for fuel oxidation and cladding splitting [Bechtel, 2005] for use
9027 during long durations at the Yucca Mountain facility that tries to account for the fuel-to-cladding
9028 gap and burnup of the fuel. The gap is the as-measured cold gap and does not account for the
9029 closing of the gap due to differential thermal expansion of the cladding and fuel material, which
9030 could be calculated. There are inadequate data to verify correctness of the DOE model. Plots
9031 in the Einziger document [Einziger, 1986] present actual data and comparisons with the data
9032 taken by other researchers at 30 GWd/MTU. The gap closure is implicitly accounted for in the
9033 measurements of splitting. However, no burnup effects can be inferred from this data.
9034

9035 No oxidation or cladding splitting studies have been conducted on fuel with burnup greater than
9036 45 GWd/MTU. Data between 30 and 45 GWd/MTU, shows a decrease in the oxidation rate due
9037 to the presence of certain actinides and fission products that are burned into the fuel. There is
9038 no reason that this should not continue at higher burnups, but the strength of the effect may
9039 change with burnup. Higher burnup fuel (>55 GWd/MTU) forms an external rim on the pellets
9040 that consists of very fine grains (1 micron). As indicated earlier, the oxidation process is a grain
9041 boundary effect. The fuel pellet must be divided into two regions for the purpose of oxidation
9042 analysis; the center of the pellet where the grains have grown slightly, and the rim. While the
9043 rate of the oxidation may decrease with burnup, the total amount of fuel that is oxidized may
9044 increase due to a much greater intergranular surface area in the rim region. The DOE model
9045 [Bechtel, 2005] uses a linear decrease in oxidation with burnup but this has, as yet, not been
9046 substantiated. A burnup effect is supported by Hanson's analysis [Hanson, 1998] of Einziger
9047 and Cook's data from the NRC whole-rod tests in which defect propagation was observed to
9048 occur earlier at the defects at the lower end of the rod where the burnup was lower.

9049
9050 Studies using a low partial pressure of water vapor in air have not shown any dependence of
9051 the oxidation rate on the moisture content of the air [Ferry, 2005]. On the other hand, there are
9052 some studies that have shown a large increase in the oxidation rate when the moisture content
9053 is above 50 percent of the dew point. Oxidation in a 100 percent steam atmosphere is a
9054 different process. There are also studies that indicate that the oxidation rate will decrease if the
9055 oxygen content in the atmosphere drops into the range of a few torr or less [Nakamura, 1995].
9056 It does not appear that there is an effect of oxygen content at higher oxygen levels but the data
9057 is sparse.

9058
9059 Oxidation studies on fuel, with few exceptions, have been conducted on LWR fuel [Einziger,
9060 1986; Johnson, 1984]. However, the UO₂ matrix is essentially the same in both PWR and BWR
9061 fuel. At the higher burnups, oxidation behavior may vary slightly as the actinide and fission
9062 product burn-in varies. The effect of the process on the splitting of the cladding may vary
9063 considerably due to the difference in gap size between the cladding types, and the thicker
9064 cladding in BWR rods.

9065
9066 The limited cladding splitting studies have been conducted on Zircaloy clad PWR [Einziger,
9067 1984, 1986; Johnson, 1984] and CANDU fuel. Defects were put in the fuel either by an SCC
9068 (stress corrosion cracking) process producing small sharp holes more typical of those found in
9069 reactor initiated SCC and by drilling that produced a larger duller hole. Most of the defects used
9070 in the studies were of the latter type. No measurements were made in cladding above
9071 30 GWd/MTU. Very few data points were measured to determine the splitting rate; therefore,
9072 the time to start splitting has to be determined by interpolation. As a result, there is large
9073 uncertainty in both measurements. No measurements have been made on other alloy types
9074 (e.g., M5 and Zirlo) or at higher burnups where the cladding may be more brittle. Fuel oxidation
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9076
9077 **8.7.3 References**

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9119 **8.8 Supplemental Information for Background justification for Cladding Temperature**
9120 **Considerations for the Storage of Spent Fuel (MEDIUM Priority)**

9121
9122 **8.8.1 Basis for Guidance**

9123
9124 Creep is the dominant mechanism for cladding deformation under normal conditions of storage.
9125 The relatively high temperatures, differential pressures, and corresponding hoop stress on the
9126 cladding will result in permanent creep deformation of the cladding over time. Several
9127 laboratory programs have demonstrated that spent fuel has significant creep capacity even after
9128 15 years of dry cask storage. Einziger, et al., [2003] reported that irradiated Surry-2 PWR fuel
9129 rods (35.7 GWd/MTU) that were stored for 15 years at an initial temperature of 350°C (with
9130 temperatures reaching as high as 415°C for up to 72 hours) experienced thermal creep, which
9131 was estimated to be less than 0.1 percent. Post-storage creep tests were conducted to assess
9132 the residual creep capacity of the Surry-2 fuel rods. One-rod segment experienced a creep
9133 strain of 0.92 percent without rupture at 380°C and 220 MPa in 1820 hours (75.8 days). A
9134 different rod segment was tested at 400°C and 190 MPa for 1873 hours (78 days) followed by
9135 693 hours (28.9 days) at 400°C and 250 MPa, and experienced a creep strain of more than
9136 5 percent without failure [Tsai, 2002]. Profilometry measurements on that fuel rod indicated that
9137 the creep deformation was uniform around the circumference of the cladding with no signs of
9138 localized bulging, which can be a precursor for rupture. A report of the literature [Beyer, 2001]
9139 also indicates that some spent fuel cladding can accommodate creep strains of 2.87.5 percent
9140 at temperatures between 390 and 420°C and hoop stresses between 225 and 390 MPa. Other
9141 significant contributions to the understanding of the effects of creep on spent fuel cladding can
9142 be found in several references [Einziger, et al., 1982; Rashid, et al., 2000; Hendricks, 2001;
9143 Rashid and Dunham, 2001; Machiels, 2002]. In general, these data and analyses support the
9144 conclusions that (1) deformation caused by creep will proceed slowly over time and will
9145 decrease the rod pressure, (2) the decreasing cladding temperature also decreases the hoop
9146 stress, and this too will slow the creep rate so that during later stages of dry storage, further
9147 creep deformation will become exceedingly small, and (3) in the unlikely event that a breach of
9148 the cladding due to creep occurs, it is believed that this will not result in gross rupture.
9149

9150 Based on these conclusions, the staff has reasonable assurance that creep under normal
9151 conditions of storage will not cause gross rupture of the cladding and that the geometric
9152 configuration of the spent fuel will be preserved provided that the maximum cladding
9153 temperature does not exceed 400°C (752°F). As discussed below, this temperature will also
9154 limit the amount of radially oriented hydrides that may form under normal conditions of storage.
9155

9156 The effects of normal conditions of storage (i.e., the decaying temperature and hoop stress on
9157 the cladding with time) can affect the metallurgical condition of spent fuel cladding containing
9158 significant amounts of hydrogen (e.g., spent fuel with high burnup levels). As the burnup level
9159 of the fuel increases beyond 45 GWd/MTU during reactor operation, the thickness of the oxide
9160 layer on the cladding increases. With increasing oxidation during reactor operation, the
9161 cladding absorbs more hydrogen. As discussed in Garde, et al., [1996], Chung and Kassner
9162 [1997], and Newman [1986], high burnup fuels tend to have relatively higher concentrations of
9163 hydrogen in the cladding. The hydrogen is present in the cladding predominantly as zirconium
9164 hydride precipitates, or particles. After the fuel is removed from the reactor, the zirconium
9165 hydrides are generally elongated and oriented circumferentially and are predominantly present
9166 in the outer rim of the cladding. At elevated temperatures, a percentage of the zirconium
9167 hydrides will dissolve, and under decreasing temperatures, zirconium hydrides will precipitate,
9168 or re-form.
9169

9170 The materials phenomenon of **hydride reorientation** in zirconium-based alloys usually involves
9171 the dissolution of hydrides and the formation of zirconium-hydrides oriented perpendicular to the
9172 hoop stress (also referred to as radially oriented or radial hydrides) [Chung, 2000]. This occurs
9173 under sufficiently high hoop stresses along with the decrease in solubility of hydrogen that
9174 accompanies decreasing temperatures. The extent of the formation of radially oriented hydrides
9175 is a function of many parameters including the solubility of hydrogen in irradiated cladding
9176 material, cladding temperature, hoop stress, cooling rate, hydrogen concentration, thermal
9177 cycling, and materials characteristics. Among these parameters, the formation of radial
9178 hydrides is highly dependent on the hoop stress in the cladding. Data obtained from irradiated
9179 cladding [Einzigler and Kohli, 1984; Cappelaere, et al., 2001; and, Goll, et al., 2001] indicate that
9180 stresses greater than 120 MPa (17.4 ksi) are required to initiate the formation of radial hydrides.
9181 Other data obtained from unirradiated zirconium-based cladding materials [Kese, 1998] indicate
9182 that radial hydrides can form at stresses as low as 90 MPa. Therefore, until the effects of
9183 reorientation are better understood, the hoop stress on the cladding should be controlled to
9184 preclude the formation of radially oriented hydrides.
9185

9186 In general, a temperature limit of 400°C that is specified for normal conditions of storage and for
9187 short-term fuel loading and Part 72 storage operations (which includes drying, backfilling with
9188 inert gas, and transfer of the cask to the storage pad) will limit cladding hoop stresses and limit
9189 the amount of soluble hydrogen available to form radial hydrides. The use of a 400°C
9190 temperature limit for normal conditions of storage and for short-term fuel loading and storage
9191 operations will simplify the calculations in SARs while assuring that hydride reorientation will be
9192 minimized.
9193

9194 For low burnup fuel, a higher temperature limit may be used for short-term fuel loading and
9195 storage operations only, as long as the applicant can demonstrate that the best estimate
9196 cladding hoop stresses are equal to or less than 90 MPa for the temperature limit that is
9197 justified. For example, if the calculated best estimate hoop stress is equal to 90 MPa at 540°C,
9198 then 540°C is the maximum allowable temperature for loading operations. In this example,
9199 570°C is not the maximum allowable temperature limit. If the applicant can show that the best
9200 estimate hoop stress is less than or equal to 90 MPa at 570°C, then 570°C is the maximum

9201 allowable temperature. For some fuel types, short-term fuel loading and storage operation
9202 temperature limits as high as 570°C (1058°F) should be justified by the applicant. The materials
9203 reviewer should coordinate with the thermal reviewer to assure that either of the following
9204 criteria are used: (1) for low and high burnup fuel, the maximum calculated temperatures for
9205 normal conditions of storage and for fuel loading operations do not exceed 400°C, or (2) for low
9206 burnup fuel, a higher temperature limit may be used for loading and transfer operations, if the
9207 best estimate cladding hoop stress is less than 90 MPa for the temperature specified by the
9208 applicant. If the applicants use the latter approach, the materials reviewer should verify that the
9209 cladding hoop stresses are less than 90 MPa for each fuel assembly type (e.g., 14x14, 17x17,
9210 9x9, etc.) proposed for storage. Since the hoop stress is dependent on the rod internal
9211 pressure, cladding geometry, and the temperature of the gases inside the rod, the materials
9212 reviewer should coordinate with the thermal reviewer to verify that the applicant has calculated
9213 the best estimate hoop stress corresponding to the rod internal pressure of the highest burnup
9214 fuel assemblies of the specific type of assembly. It should be noted that during normal
9215 conditions of storage there will be a range of cladding temperatures that are less than the
9216 maximum allowable cladding temperature of 400°C, and this leads to a range of the internal rod
9217 pressures and the cladding hoop stresses, in any one storage cask. In general, the maximum
9218 allowable temperature will be 400°C or the maximum allowable temperature specified and
9219 supported (as discussed above) by the applicant. The maximum allowable temperature should
9220 be based upon the **peak** rod temperature, not the average rod temperature. By employing the
9221 peak rod temperature, only a small fraction of the rods will experience the temperature and
9222 stress conditions that could lead to the formation of radial hydrides during normal conditions of
9223 storage.

9224
9225 It also has been observed and reported that thermal cycling (repeated heatup/cooldown cycles)
9226 can enhance the amount of hydrogen that eventually re-precipitates in the form of radial
9227 hydrides [Kammenzind, et al., 2000]. The extent of the formation of radial hydrides is
9228 dependent on many factors including the maximum temperature, change in temperature,
9229 number of thermal cycles, applied stress, hydrogen concentration, and solubility of hydrogen in
9230 the material. Kammenzind, et al., [2000] indicates that the formation of radial hydrides in spent
9231 fuel cladding can be minimized by restricting the change in cladding temperatures to less than
9232 65°C and minimizing the number of cycles to less than 10. The 65°C temperature limit is based
9233 upon the temperature drop required to obtain the degree of supersaturation required for the
9234 precipitation of hydrides in a short thermal cycle.

9235
9236 For short-term accidents and short-term off-normal conditions that lead to an increase in
9237 temperature of the cladding, the dominant cladding failure mechanism is expected to be creep
9238 (stress rupture) of the cladding. To limit the amount of spent fuel that could be released from
9239 the cladding under off-normal conditions or accidents, the materials reviewer should coordinate
9240 with the thermal reviewer to verify that the maximum calculated cladding temperatures are
9241 maintained below 570°C (1058°F). The basis for using 570°C is established by the creep tests
9242 conducted on irradiated Zircaloy-4 rods [Einziger, et al., 1982]. The results from these
9243 experiments indicated that no cladding ruptures were observed for test times of 30 and 73 days.

9244 **8.8.2 Review Guidance**

9245
9246
9247 Prior to this guidance the short-term cladding temperature limit applicable to fuel loading
9248 operations was 570°C. All storage casks were certified using this limit. The current guidance to
9249 maintain cladding temperatures less than 400°C during fuel loading operations put into question
9250 whether the licensees who use certified storage casks (certified for fuel having average
9251 assembly burnups less than 45 GWd/MTU) would have to change their loading procedures and

9252 Technical Specifications to comply with this new temperature limit. Based on staff's evaluation,
9253 it is expected that fuel assemblies with burnups less than 45 GWd/MTU are not likely to have a
9254 significant amount of hydride reorientation due to limited hydride content. Further, most of the
9255 low burnup fuel has hoop stresses below 90 MPa. Even if hydride reorientation occurred during
9256 storage, the network of reoriented hydrides is not expected to be extensive enough in low
9257 burnup fuel to cause fuel rod failures.

9258
9259 Given the conservatism used in calculating peak clad temperatures for low burnup fuel, the staff
9260 has reasonable assurance that storage cask systems which use the 570°C temperature limit for
9261 low burnup fuel loading operations will continue to perform as expected when the casks were
9262 originally certified. Therefore, there is no need to require the licensees of storage-only or dual-
9263 purpose cask systems to repackage spent fuel that was loaded using the 570°C temperature
9264 limit. Nevertheless, the 400°C limit is intended, with exceptions as stated above, to be generally
9265 applicable to all future loadings. Therefore, licensees are not required to modify their Technical
9266 Specifications or fuel loading procedures (i.e., vacuum drying) to meet the new 400°C limit for
9267 loading low burnup fuel into storage casks previously certified with the 570°C limit. Note that for
9268 future amendments to certified designs, the applicants may be required to comply with the
9269 400°C temperature limit as discussed above.

9270

9271 **8.8.3 References**

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9356

9357 **8.9 Supplemental Information for the Design and Testing of Lid Welds on Austenitic**
9358 **Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage**
9359 **(MEDIUM Priority)**
9360

9361 **8.9.1 Basis for the Review**
9362

9363 10 CFR 72.236(e) states: "The spent fuel storage cask [note: also called "canister"] must be
9364 designed to provide redundant sealing of confinement systems." For a bolted lid canister
9365 design, the staff has accepted a dual seal arrangement as meeting the intent of this regulation.
9366 For a welded canister design, the staff has accepted closure designs employing redundant lids
9367 or covers, each with independent field welds. Thus, for either closure type, bolted or welded, a
9368 potential leak path must breach two independent seals or welds, sequentially, before the
9369 confinement system would be compromised.
9370

9371 The construction codes specify the types of non-destructive examinations (NDE) required for
9372 the confinement boundary during canister fabrication and loading operations. In addition to the
9373 code required NDE, a helium leakage test of the confinement boundary is considered necessary
9374 to satisfy regulatory requirements. Whereas bolted lid canister designs incorporate a helium
9375 monitoring system during storage, the welded closure designs must rely on weld integrity to
9376 assure continued confinement effectiveness. Consequently, at least one of the redundant
9377 welded closures must be helium leak tested per the method of ANSI N 14.5, with one exception
9378 permitted.
9379

9380 When the large, multi-pass weld joining the canister shell to the structural lid of an austenitic
9381 stainless steel spent fuel canister is executed and examined consistent with the guidance
9382 provided herein, the staff has reasonable assurance that no flaws of significant size will exist
9383 such that they could impair the structural strength or confinement capability of this weld. For a
9384 spent nuclear fuel canister, such a flaw would be the result of improper fabrication or welding
9385 technique, as service-induced flaws under normal and off-normal conditions of storage are not
9386 credible. Any such fabrication flaw would be reasonably detectable during the in-process and
9387 post-weld examination techniques described herein.
9388

9389 Based on evaluation, these described techniques should detect any such flaw which could lead
9390 to a failure or credible leakage of radioactive material. Therefore, the staff believes that there is
9391 reasonable assurance that no credible leakage of radioactive material would occur through the
9392 structural lid to canister shell weld of an austenitic stainless steel canister, and that helium
9393 leakage testing of this specific weld is unnecessary provided the weld is executed and
9394 examined in accordance with the methods described herein.
9395

9396 Conversely, it is the staff position that other welds associated with the lid assemblies of spent
9397 fuel canisters must be subject to the helium leak test of ANSI N 14.5, in addition to the ASME
9398 Code required pressure test and surface NDE in order to demonstrate compliance with
9399 10 CFR 72.236.
9400

9401 Note the criteria outlined above does not supercede the flaw acceptance criteria of any
9402 construction code. Instead, this criteria is used to establish the maximum allowable weld
9403 deposit depth before an in-process penetrant test (PT) examination is required.
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8.9.2 Helium Leak Test

The helium leak test was established to provide assurance that:

- No leakage occurred after the closure welds of the cask system were executed. This was viewed as necessary since no active or passive methods are employed to confirm or monitor the presence of helium within an all-welded spent fuel canister over its licensed lifetime. “No leakage” in this case means measured leak rate performed per ANSI N14.5, at a predetermined sensitivity that shows hypothetical doses would not exceed 10 CFR Part 72 limits.
- If the weld(s) meets the criteria of ANSI N14.5, the staff has assurance that radio nuclide leakage would not exceed the regulatory dose limits in 10 CFR Parts 72.104 and 72.106.
- No oxygen in-leakage could occur, thereby assuring the presence of the inert helium atmosphere which prevents oxidation and corrosion induced degradation of the spent fuel assemblies and enhances cooling of the spent fuel.

Helium Leak-Testing of the Confinement Boundary

The redundant weld requirement for the confinement system closure creates two closure boundaries. The staff should verify that at least one of the redundant boundaries is helium leak tested, or, some closure welds leak tested and the remaining closure welds of the same boundary designed so that the “large weld” exemption criteria of this guidance are met. Only a boundary which is testable or excluded from testing, per this guidance, should be considered the confinement boundary of the redundant closures. Refer to sketches A and B and the following narrative for application of this criteria to two currently approved designs:

Leak Testing a Single Lid With Cover Plate Design - Sketch A.

In sketch A, located at the end of Chapter 8, “Materials Evaluation” of this SRP, the dotted line marked (1) defines one closure boundary. Starting on the left side of the sketch, the closure boundary can be traced from the canister wall, up through the large, multi-pass weld joining the canister wall to the heavy section, combined shield and structural lid. The boundary continues through the lid to the small weld joining the heavy lid to the vent-and-drain port closure plate, and back to the heavy lid again. The remainder of the boundary (and sketch) is assumed to be symmetrical with or similar to the half-sketch portion that is shown, for all cases.

This boundary demonstrates confinement integrity by means of the large weld exemption criteria for one weld and by helium leak testing the small cover plate weld.

The large, canister-shell-to-lid weld is exempted from the helium leak test. This is because the canister shell to lid weld is a large, multi-pass weld meeting the flaw tolerance and other appropriate portions of this guidance. Note that this weld is executed prior to filling the canister with helium (excluding purging/welding gas).

Before the remaining welds of this first closure boundary are executed, the canister is drained, dried, purged, and filled with helium to the design operating pressure. The helium line connection is closed off and the cover plate fitted and welded into place. Since the cover plate

9455 weld may have potentially been pressurized from underneath due to assumed leakage from the
9456 closure valve, it must be helium leak tested in accordance with the methods described in ANSI
9457 N14.5-1997. If there are other cover plates and welds, they would also be helium leak tested.
9458

9459 This completes the first closure boundary. Note again that one weld was exempted from the
9460 helium leak test by the design criteria. The other weld was leak tested. Thus, this closure
9461 boundary demonstrates compliance with regulatory requirements and is consistent with the staff
9462 guidance by ensuring at least one of the two redundant closure boundaries is leak tested or
9463 exempted from leak testing by conformance with the large-weld exemption guidance. This
9464 boundary thus also qualifies as the confinement boundary.
9465

9466 The second boundary, delineated by line 2 in diagram A, can be traced from the canister wall on
9467 the left side of the sketch up through the cover plate fillet weld joining the canister wall to the
9468 structural lid cover plate. The boundary continues through the cover plate to the fillet weld
9469 joining the cover plate to the canister lid. The weld joining the cover plate to the canister wall
9470 and lid cannot be helium leak tested since there is no feasible means to do so. However, since
9471 the first closure boundary, delineated by line 1, was tested (or exempted thru design), the need
9472 to helium leak test at least one of the closure boundaries has been satisfied. Since this second
9473 boundary does not meet all the criteria for a confinement boundary, it may not be designated as
9474 the confinement boundary. The first closure is thereby the confinement boundary in this design,
9475 as it meets all the applicable criteria for a confinement boundary.
9476

9477 Leak Testing a Dual Lid Design - Sketch B

9478
9479 In sketch B, located at the end of Chapter 8, "Materials Evaluation" of this SRP, the dotted line
9480 marked (1) defines one of the redundant closure boundaries. It may be traced from the canister
9481 wall on the left side of the sketch. The boundary proceeds through the partial penetration weld
9482 joining the canister wall to the shield lid and into the shield lid. It continues through the small
9483 fillet weld joining the vent/drain port cover plate, the cover plate, and back through the same
9484 fillet weld to the shield lid.
9485

9486 This closure boundary may satisfy the leak test guidance by several methods, depending on
9487 details of the weld design. The canister shell to shield lid weld may be designed several ways.
9488 The weld may be a small seal weld which would necessitate subsequent helium leak testing.
9489 Conversely, it could be a large, multi-pass weld consistent with the guidance described herein.
9490 In that case, the weld would qualify for the leak test exemption. Either way, note that this weld
9491 (canister to shield lid weld) is executed prior to filling and pressurizing the canister with helium
9492 (use of purge or backing gas for welding operations is not considered filling or pressurizing).
9493

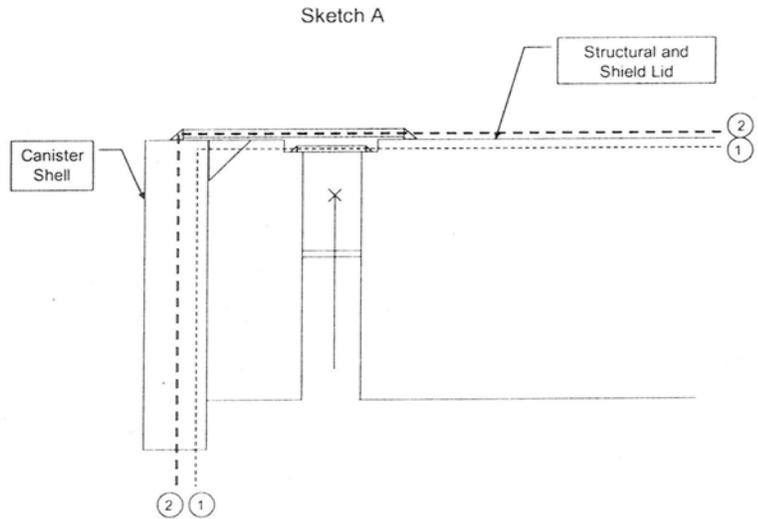
9494 Next, the canister is drained, dried, purged, and filled with helium to the design operating
9495 pressure. The helium line connection is closed off. The cover plate is fitted and welded into
9496 place. Since this weld may potentially be pressurized from underneath due to assumed leakage
9497 through the closure valve, it must be helium leak tested regardless of weld size (thickness).
9498

9499 This completes the first closure boundary. Note that one weld was either tested, or, exempted
9500 from the helium leak test by the design criteria. The other weld was leak tested. Thus, this
9501 closure boundary demonstrates compliance with regulatory requirements and is consistent with
9502 staff guidance by ensuring at least one of the two redundant closures is leak tested or exempted
9503 by conformance to this guidance. This closure may therefore be designated as the confinement
9504 boundary.

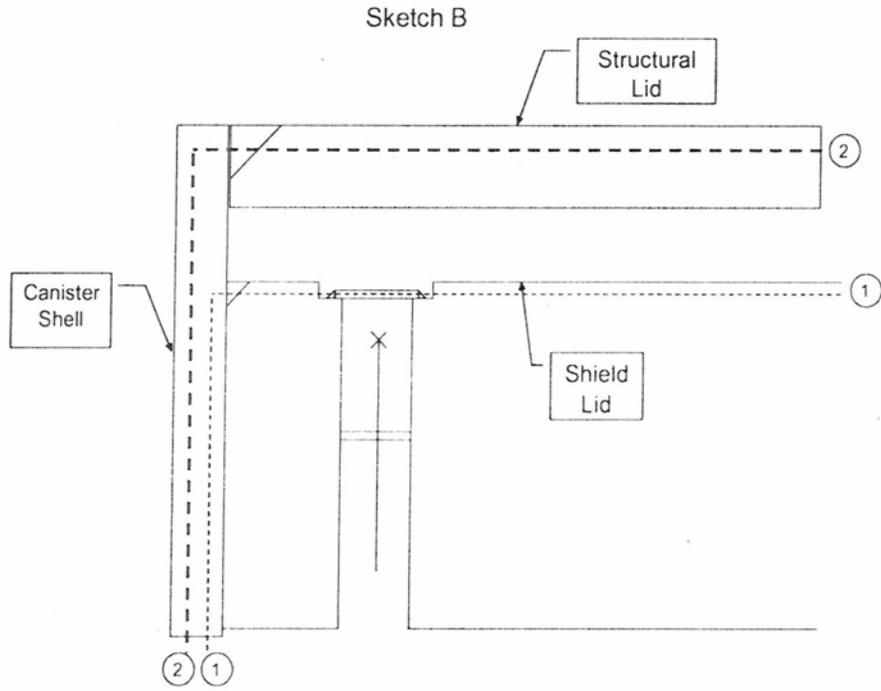
9505 The secondary boundary, delineated by line 2 in sketch B, can be traced from the canister wall
9506 on the left side of the sketch up through the canister wall-to-structural lid weld and into the
9507 structural lid.
9508

9509 The weld joining the canister wall and structural lid cannot be helium leak tested because
9510 helium is not present. Note, however, that this weld complies by design with the criteria
9511 described herein due to its size, structural requirements and weld examination requirements of
9512 the governing construction code.
9513

9514 In this case, the second closure also qualifies for designation as the confinement boundary
9515 because the single large weld involved may be exempted from the helium leak test. In this
9516 design, the designer therefore has the freedom to designate either of the redundant closures as
9517 the confinement boundary. Only one of the two closures is designated as the confinement
9518 boundary.



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9 OPERATING PROCEDURES EVALUATION

9.1 Review Objective

The operating procedures review ensures that the applicant's safety analysis report (SAR) presents acceptable operating sequences, guidance, and generic procedures for the key operations shown in Section 9.2, "Areas of Review." The review also ensures that the SAR incorporates and is compatible with the applicable operating control limits in the technical specifications.

The operating sequences described in the SAR should provide an effective basis for the development of the more detailed operating and test procedures by the cask user when preparing and implementing detailed site-specific procedures. The NRC normally inspects selected site-specific procedures. Such procedures are important aspects of the site's radiation protection program and allow the cask user to safely store spent nuclear fuel (SNF).

This chapter applies to all discipline reviews. Figure 1-1 presents an overview of the evaluation process and can be used as a guide to assist in coordinating with other review disciplines.

9.2 Areas of Review

This chapter of the dry storage system (DSS) Standard Review Plan (SRP) provides guidance in evaluating the applicant's general operating sequences and generic procedures related to cask operations. Within each area of cask operations, the NRC staff assesses the effectiveness of the applicant's generic procedures on a technical and safety basis for the subsequent development of detailed operating procedures. As required by U.S. Code of Federal Regulations (CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, "Energy" (10 CFR Part 72) 72.234(f), these procedures are to be provided to each cask user for the subsequent preparation and implementation of detailed site-specific procedures by the cask system user acting under a general license. Areas of review addressed in this chapter include the following:

Loading Operations

- Fuel Specifications
- Damaged Fuel
- Subcriticality Features
- ALARA
- Offsite Release
- Draining and Drying
- Filling and Pressurization
- Welding and Sealing
- Administrative Programs

Cask Handling and Storage Operations

Cask Unloading

- Damaged Fuel
- Cooling, Venting, and Reflooding
- Fuel Crud
- ALARA
- Offsite Release

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9.3 Regulatory Requirements

This section presents a summary matrix of the portions of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should read the exact referenced regulatory language. Table 9-1 matches the relevant regulatory requirements associated with this chapter to the areas of review.

Table 9-1 Relationship of Regulations and Areas of Review						
Areas of Review	10 CFR Part 72 Regulations					
	72.104(b)	72.122(f), (h)(1), (l)	72.212 (b) (9)	72.234 (f)	72.236 (c)	72.236(h), (i)
Cask Loading Operations	•		•	•	•	•
Cask Handling and Storage Operations	•	•	•	•		•
Cask Unloading		•	•	•		•

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9.4 Acceptance Criteria

Chapter 9, "Operating Procedures Evaluation," of the SAR should identify and describe the sequence of significant operations and actions that are important to safety for cask loading, cask handling, storage operations, and cask unloading. A sufficient level of detail is needed in Chapter 9 of the SAR for the reviewer to conclude that operating procedures will adequately protect health and minimize danger to life or property, protect the fuel from significant damage or degradation, and provide for the safe performance of tasks and DSS operations.

This portion of the DSS review seeks to ensure that the generic procedure descriptions and operational sequences described in the SAR include the following information:

- Major operating procedures should apply to the principal activities expected to occur during dry storage. The expected scope of activities for the SAR operating procedure descriptions is previously described in Section 9.2 as well as Chapter 8 of Regulatory Guide (RG) 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask." Operating procedure descriptions should be submitted to address the cask design features and planned operations.
- Operating procedure descriptions should identify measures to control processes and mitigate potential hazards that may be present during planned normal operations. Section 9.5, "Review Procedures," in this chapter discusses previously identified processes and potential hazards.

- 9607 • Operating procedure descriptions should ensure conformance with the applicable
9608 operating controls and limits described in the cask system's Technical
9609 Specifications provided in Chapter 13, "Technical Specifications and Operating
9610 Controls and Limits Evaluation," of the SAR.
- 9611 •
- 9612 • Operating procedure descriptions should reflect planning to ensure that
9613 operations will fulfill the following acceptance criteria:
9614
 - 9615 - Occupational radiation exposures will remain as low as is reasonably
9616 achievable (ALARA).
 - 9617 -
 - 9618 - Effective measures will be taken to preclude potential unplanned and
9619 uncontrolled releases of radioactive materials.
 - 9620 -
 - 9621 - Offsite dose rates will be maintained within the limits of 10 CFR Part 20
9622 and 10 CFR 72.104 for normal operations, and 10 CFR 72.106 for
9623 accident-level conditions.
 - 9624
- 9625 In addition, the operating procedure descriptions should support and be
9626 consistent with the bases used to estimate radiation exposures and total doses
9627 as defined in Chapter 11, "Radiation Protection Evaluation," of this SRP.
- 9628 •
- 9629 • Operating procedure descriptions should include provisions for the following
9630 activities:
9631
 - 9632 - Testing, surveillance, and monitoring of the stored material and casks
9633 during storage and loading and unloading operations.
 - 9634 -
 - 9635 - Contingency actions triggered by inspections, checks, observations,
9636 instrument readings, and so forth. Some of these may involve off-normal
9637 conditions addressed in Chapter 12, "Accident Analyses Evaluation," of
9638 the SAR.
 - 9639

9640 **9.4.1 Cask Loading**

9641 In addition to the acceptance criteria above, additional acceptance criteria for cask loading are
9642 as follows:
9643

- 9644 •
- 9645 • The operating procedure descriptions should facilitate reducing the amount of
9646 water vapor and oxidizing material within the confinement cask to an acceptable
9647 level to protect the SNF cladding against degradation that might otherwise lead
9648 to gross ruptures.
- 9649 •
- 9650 • Operating procedures should specify methods for placing damaged fuel in a
9651 damaged-fuel can prior to loading into a cask, if applicable.
- 9652

9653 **9.4.2 Cask Handling and Storage Operations**

9654 In addition to the acceptance criteria stated above, operating procedure descriptions should
9655 include provisions for maintenance of casks and cask functions during storage.
9656
9657

9658 **9.4.3 Cask Unloading**

9659
9660 In addition to the acceptance criteria stated above, operating procedures should facilitate ready
9661 retrieval of SNF stored in a storage cask.

9662
9663 **9.5 Review Procedures**

9664
9665 Introduction (MEDIUM Priority)

9666
9667 The interrelationship of the operating procedures evaluation with other disciplines is shown in
9668 Figure 9-1.

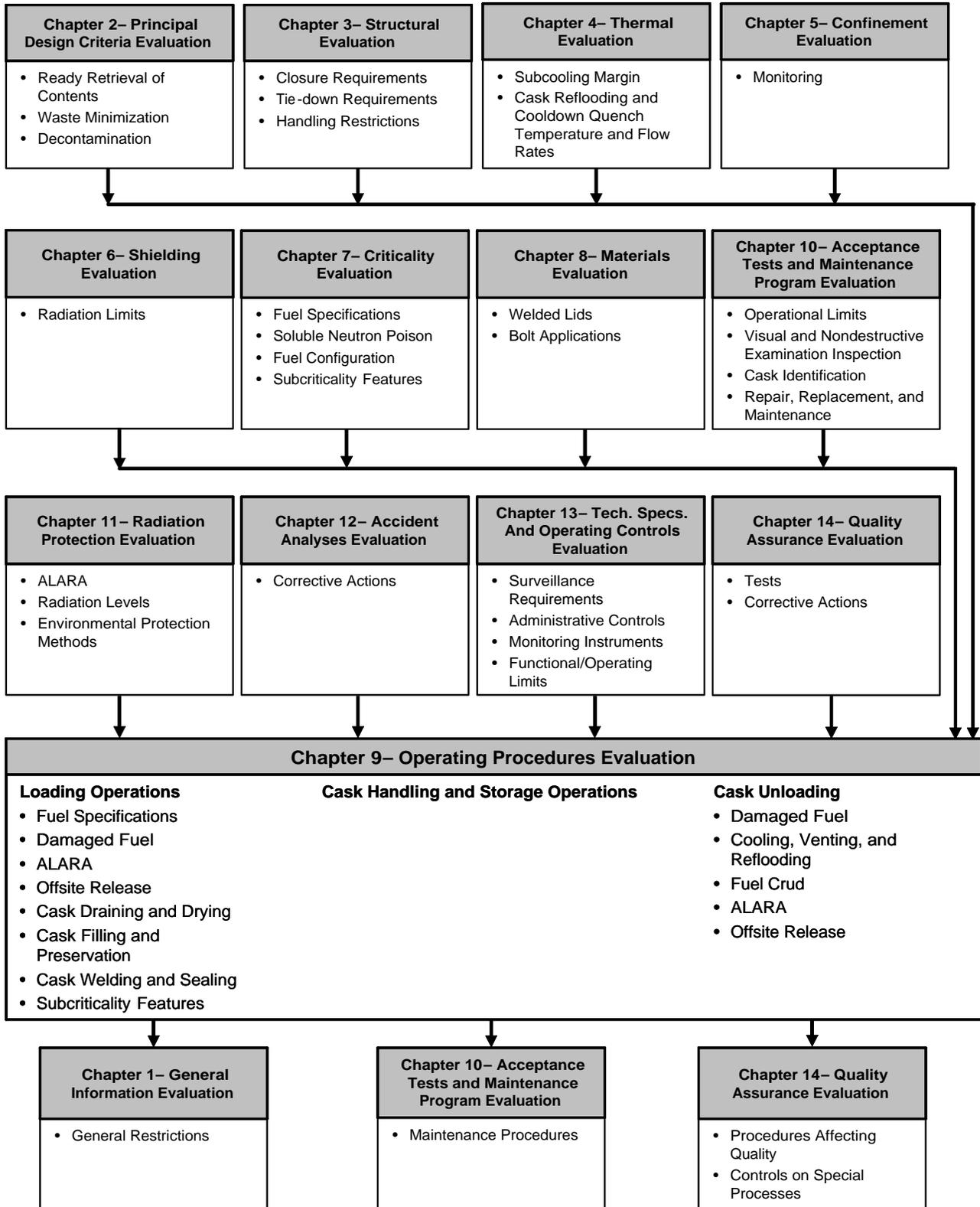
9669
9670 The review procedures described in this section are presented in a format intended to facilitate
9671 an independent review. Even though several individuals may actually be tasked with preparing
9672 the chapter of the safety evaluation report (SER) related to operating procedures, all review
9673 team members should examine the operating procedure descriptions presented in the SAR. If
9674 the descriptions included in the SAR are not sufficiently detailed to allow a complete evaluation
9675 concerning fulfillment of the acceptance criteria, reviewers should request additional information
9676 from the applicant.

9677
9678 The operating procedure sequences are described in Chapter 9 of the SAR, and the direct dose
9679 rate information in Chapter 6, "Shielding Evaluation," of the SAR is used to assess compliance
9680 with radiation protection requirements in Chapter 11 of the SAR. The reviewer should verify that
9681 the evaluation of Chapter 9 of the SAR is coordinated with the shielding and radiation protection
9682 evaluations covered in Chapters 6, "Shielding Evaluation" and 11, "Radiation Protection
9683 Evaluation," of this SRP.

9684
9685 In addition, the following review procedures are based on the assumption that the ISFSI
9686 operations are at a reactor facility licensed under 10 CFR Part 50, "Domestic Licensing of
9687 Production and Utilization Facilities," and that loading and unloading activities will be performed
9688 in the facility's SNF pool. Review procedures for dry fuel transfers and/or ISFSI operations at
9689 sites away from a reactor will be developed at a later date, if necessary.

9690
9691 Reviewers should be familiar with ANSI/ANS 57.9, "Design Criteria for an Independent Spent
9692 Fuel Storage Installation (Dry Type)," which applies to DSS operating procedures. Background
9693 information is available in NUREG/CR-4775, "Guide for Preparing Operating Procedures for
9694 Shipping Packages," which provides guidance on preparing operating procedures for shipping
9695 packages. Although NUREG/CR-4775 specifically addresses 10 CFR Part 71, most of the
9696 guidance can be adapted for storage casks that are governed by 10 CFR Part 72.
9697 Consequently, reviewers should be familiar with this information before initiating the DSS
9698 operating procedures review.

9699
9700 Since many of the detailed procedures may be developed by facilities licensed under 10 CFR
9701 Part 50 or 72, further background information on site-specific procedure requirements may be
9702 found in RG 1.33, "Quality Assurance Program Requirements (Operation)," and its associated
9703 standard ANSI/ANS 3.2. Reviewers of Chapter 9, "Operating procedures Evaluation" of the
9704 SAR should also be familiar with Chapter 11, "Conduct of Operations Evaluation," of NUREG-
9705 1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities." Specifically, Section
9706 11.4.3, "Normal Operations," in NUREG-1567 provides NRC review acceptance criteria for
9707 facility-developed procedures.



9708
9709

Figure 9-1 Overview of Operating Procedures Evaluation

9710 In general, reviewers should perform the following steps in the process of evaluating all of the
9711 operating procedure descriptions and operational sequences provided in the SAR.
9712

- 9713 • Verify that the proposed operating procedure descriptions incorporate and are
9714 compatible with the applicable operating limits and controls in Chapter 13, "Technical
9715 Specifications and Operational Controls and Limits Evaluation" of the SAR. Coordinate with the review of operating controls and limits, as described in Chapter
9716 13, "Technical Specifications and Operating Controls and Limits Evaluation," of this
9717 SRP.
9718
- 9719 • Ensure that the proposed operating procedure descriptions properly consider the
9720 prevention of hydrogen gas generation from any cause (including the reaction of zinc
9721 primer coating with acidic pool water, radiolysis, or other causes). Prevention of
9722 hydrogen generation or adequate purging of hydrogen is essential during loading
9723 and unloading operations that involve seal welding, seal cutting, grinding, or other
9724 forms of hot work.
9725
- 9726 • Determine whether the descriptions include appropriate precautions to minimize
9727 occupational radiation exposures in accordance with ALARA principles and the limits
9728 given in 10 CFR Part 20, as mandated by 72.126(a)(5). Provisions may include use
9729 of remotely controlled equipment, monitoring, and use of portable shielding.
9730
- 9731 • Verify that the operating procedure descriptions include a general listing of the major
9732 tools and equipment needed to support ISFSI loading, storage, and unloading
9733 operations (including those at the pool facility). The descriptions should also address
9734 installation, use, and removal of the cask and fuel, tools, and equipment. In addition,
9735 the descriptions should describe any specialized tools and equipment in sufficient
9736 detail to enable users to understand their use and operation. Examples include
9737 lifting yokes, transporter equipment, welding and cutting equipment, and vacuum
9738 drying equipment. The use of any such equipment that is classified as being
9739 important to safety is subject to approval as part of the application review. Such
9740 equipment should be identified and described in detail, its performance
9741 characteristics should be defined, and the design should be evaluated.
9742

9743 In addition to these generic review procedures, all disciplines should evaluate each of the
9744 specific areas of operating procedure review as described in the following subsections.
9745

9746 **9.5.1 Cask Loading (Priority - as indicated)** 9747

9748 (MEDIUM Priority) The operating procedure descriptions in the SAR should present the
9749 activities sequentially in the anticipated order of performance. The generic procedures in
9750 Chapter 9, "Operating Procedures Evaluation" of the SAR should be reviewed to ensure that
9751 they include appropriate key prerequisite, preparation, and receipt inspection activities to be
9752 accomplished before cask loading. The reviewer should verify that tests, inspections,
9753 verifications, and cleaning procedures required in preparation for cask loading are specified. In
9754 addition, where applicable, the reviewer should verify that the procedure descriptions include
9755 actions needed to ensure that any fluids such as shield water and primary coolants fill their
9756 respective cavities according to design specifications.
9757
9758

9759 Fuel Specifications (MEDIUM Priority)

9760

9761 The reviewer should verify that the loading procedure description appropriately addresses the
9762 SNF specifications (e.g., burnup, cooling period, source terms, heat generation, cladding
9763 damage, associated non-fuel hardware, etc.) in Chapter 2, "Principal Design Criteria," and
9764 Chapter 13, "Technical Specifications and Operation Controls and Limits Evaluation" of the
9765 SAR. For cask systems relying upon burnup credit, the loading procedure description should
9766 include verification that assemblies selected for loading meet the specifications for assembly
9767 operational history and the loading curve as well as include performance of measurements to
9768 confirm assembly burnup values. Depending on the types and specifications of fuel assemblies
9769 stored in the reactor SNF pool, detailed site-specific procedures may be necessary to ensure
9770 that all fuel loaded in the cask meets the fuel specifications for the cask design. These
9771 procedures can be evaluated only on a site-specific basis and will generally be evaluated
9772 through inspections rather than during the licensing review. The SAR should indicate, however,
9773 that such procedures may be necessary.

9774

9775 Damaged Fuel (MEDIUM Priority)

9776

9777 The reviewer should verify that the SAR includes appropriate measures for the loading of
9778 damaged fuel, if damaged fuel is included in the proposed cask contents. Chapter 2, "Principal
9779 Design Criteria Evaluation," and Chapter 8, "Materials Evaluation," of this SRP provide criteria
9780 for the storage of damaged fuel. Information in Section 8.6, "Supplemental Information for
9781 Methods for Classifying Fuel," of this SRP should be used to identify the conditions that
9782 determine when SNF is to be classified as damaged fuel. Section 8.4.17.2 of this SRP should
9783 be reviewed to determine the classification, documentation, and special handling requirements
9784 for damaged fuel and determine if operating procedures address these requirements.

9785

9786 Subcriticality Features (MEDIUM Priority)

9787

9788 Where applicable, the reviewer should verify that the procedure descriptions include the use of
9789 features important to criticality safety that may require installation by the DSS user. Such items
9790 include fuel spacers and items (e.g., blocks) used to prevent loading of contents in selected
9791 basket locations. The procedure descriptions should include installation, or verification of the
9792 installation, of these items prior to cask loading for casks that rely upon these features in the
9793 criticality analysis. Additionally, the procedure descriptions should include verification, in
9794 accordance with Technical Specification requirements, of the minimum soluble boron level
9795 necessary for cask loading for casks requiring soluble boron to meet subcriticality.

9796

9797 ALARA (LOW Priority)

9798

9799 The reviewer should verify that the procedure descriptions incorporate ALARA principles and
9800 practices. These may include provisions to perform radiological surveys as well as exposure
9801 and contamination control measures, temporary shielding, and suggested caution statements
9802 related to actions that could change radiological conditions. In addition, the reviewer should
9803 verify that any recommended surveys incorporate the applicable operating controls and limits
9804 described in Chapter 13, "Technical Specifications and Operating Controls and Limits
9805 Evaluation" of the SAR.

9806

9807 Offsite Release (LOW Priority)

9808

9809 Where applicable, the reviewer should verify that the SAR describes methods to minimize offsite
9810 releases such as decontamination, filtered ventilation, temporary containments (tents), and so
9811 forth. The procedure descriptions should also provide for minimizing generation of radioactive
9812 waste.

9813
9814 Draining and Drying (MEDIUM Priority)

9815
9816 The reviewer should evaluate the descriptions related to methods for use in draining and drying
9817 the cask for ISFSI operations at a reactor facility or at sites away from a reactor with a transfer
9818 pool. In particular, the descriptions should clearly describe the procedures for removing water
9819 vapor and oxidizing material to an acceptable level, and the reviewer should assess whether
9820 those procedures are appropriate.

9821
9822 The NRC staff has accepted vacuum drying methods comparable to those recommended in
9823 PNL-6365 (Knoll, 1987). This report evaluates the effects of oxidizing impurities on the dry
9824 storage of light-water reactor (LWR) fuel and recommends limiting the maximum quantity of
9825 oxidizing gasses (such as O₂, CO₂⁴, and CO) to a total of 1 gram-mole per cask. This
9826 corresponds to a concentration of 0.25 volume percent of the total gases for a 7.0m³ (about
9827 247 ft³) cask gas volume at a pressure of about 0.15 MPa (1.5 atm) at 300°K (80.3°F). This
9828 1 gram-mole limit reduces the amount of oxidants below levels where any cladding degradation
9829 is expected. Moisture removal is inherent in the vacuum drying process, and levels at or below
9830 those evaluated in PNL-6365 (about 0.43 gram-mole H₂O) are expected if adequate vacuum
9831 drying is performed.

9832
9833 If alternative methods other than vacuum drying are used (such as forced helium recirculation),
9834 the reviewer should ensure that additional analyses or tests are provided to sufficiently justify
9835 that cover gas moisture and impurity levels as specified in Chapter 9, "Operating Procedures
9836 Evaluation" of the SAR are met and will not result in unacceptable cladding degradation.

9837
9838 The following examples illustrate the accepted methods for cask draining and drying in
9839 accordance with the recommendations of PNL-6365 (Knoll, 1987):

9840
9841 • The cask should be drained of as much water as practicable and evacuated to
9842 less than or equal to 4.0E-04 MPa (4 millibar, 3.0 mm Hg or Torr). After
9843 evacuation, adequate moisture removal should be verified by maintaining a
9844 constant pressure over a period of about 30 minutes without vacuum pump
9845 operation (or the vacuum pump is running but it is isolated from the cask with its
9846 suction vented to atmosphere). The cask is then backfilled with an inert gas
9847 (e.g., helium) for applicable pressure and leak testing. The cask is then re-
9848 evacuated and re-backfilled with inert gas before final closure. Care should be
9849 taken to preserve the purity of the cover gas and, after backfilling, cover gas
9850 purity should be verified by sampling.

9851
9852 • The procedures should reflect the potential for blockage of the evacuation
9853 system or masking of defects in the cladding of non-intact rods, as a result of
9854 icing during evacuation. Icing can occur from the cooling effects of water
9855 vaporization and system depressurization during evacuation. Icing is more likely
9856 to occur in the evacuation system lines than in the cask because of decay heat
9857 from the fuel. A staged draw down or other means of preventing ice blockage of

⁴ Can be broken down by radiolysis.

- 9858 the cask evacuation path may be used (e.g., measurement of cask pressure not
9859 involving the line through which the cask is evacuated).
- 9860 • The procedures should specify a suitable inert cover gas (such as helium) with a
9861 quality specification that ensures a known maximum percentage of impurities to
9862 minimize the source of potentially oxidizing impurity gases and vapors and
9863 adequately remove contaminants from the cask.
 - 9864
 - 9865 • The process should provide for repetition of the evacuation and repressurization
9866 cycles if the cask interior is opened to an oxidizing atmosphere following the
9867 evacuation and repressurization cycles (as may occur in conjunction with
9868 remedial welding, seal repairs, etc.).
9869

9870 Reviewers should ensure that the drying specifications are consistent with the proposed
9871 operating controls and limits described in the technical specifications provided in Chapter 13 of
9872 the SAR. In addition, reviewers should assess the need for any additional technical
9873 specifications.
9874

9875 Welding and Sealing (HIGH Priority) 9876

9877 Structural and materials disciplines should coordinate their review of welded lids as described in
9878 Section 8.4.7, "Weld Design/Inspection," of this SRP for application of the proper weld joint,
9879 welding procedures, and non-destructive examination methods (NDE) to ensure the appropriate
9880 operating procedures are in place and acceptable. Reviewers should verify that procedures are
9881 acceptable for NDE and welding of the closure welds. While the NRC accepts progressive
9882 surface examinations utilizing dye penetrant testing (PT) or magnetic particle (MT) examination,
9883 it is only permitted if unusual design or loading conditions exist. In addition, if a PT or MT
9884 examination is used, a stress-reduction-factor of 0.8 is imposed on the weld strength for the
9885 reasons presented in Section 8.4.7.3. The SAR should also ensure ALARA principles are
9886 followed and include acceptable provisions for correcting weld defects and any additional drying
9887 and purging that may be necessary.
9888

9889 The reviewer should verify that provisions for placing and tightening any closure bolts, such as
9890 those associated with concrete casks, are consistent with information presented in Chapters 2,
9891 3, and 10 of the SAR that address applicable design criteria, structural evaluation, and the
9892 acceptance tests and maintenance program, respectively. The materials discipline should
9893 ensure that the closure bolts satisfy the conditions given in Section 8.4.10, "Bolt Applications," of
9894 this SRP. The SAR should specify the torque required to properly seal the closure lid. The
9895 inner seal should be tested using a helium leak test with the interior of the cask pressurized as
9896 previously described. The outer seal should also be tested using a helium leak test with the
9897 between-seal volume pressurized as required by the respective subsection of the ASME B&PV
9898 Code, Section III.
9899

9900 Filling and Pressurization (LOW Priority) 9901

9902 The reviewer should verify that the procedure recommendations address steps to fill and
9903 pressurize the cask with inert gas such as helium with a known maximum percentage of
9904 impurities. The operating procedures should state that the filling and pressurization (or
9905 evacuation and backfill) process be repeated if the cask cavity is exposed to the atmosphere.
9906 Also, the reviewer should ensure that the procedure recommendations include the requirements
9907 in Chapter 13, "Technical Specifications and Operation Controls and Limits Evaluation" of the
9908 SAR.

9909
9910 The SAR should specify the leak rate criteria (e.g., total leakage, leakage per closure,
9911 sensitivities of tests, etc.), and the reviewer should verify that these criteria are consistent with
9912 those presented in Chapters 2, 9, and 13 of the SAR. In addition, the reviewer should assess
9913 the general methods of leak testing (e.g., pressure rise, mass spectrometry) as they apply to the
9914 leak rate being tested. Particular attention should be paid to the possible use of quick-
9915 disconnect fittings for draining and filling operations. Although no credit is usually taken for
9916 these devices as part of the confinement boundary, their presence can negate the results of the
9917 leak test, and the SAR should provide guidance regarding their use. In addition, the guidelines
9918 presented in the SAR should note that leak testing is in accordance with ANSI N14.5,
9919 "Radioactive Materials – Leakage Tests on Packages for Shipment."

9920
9921 The reviewer should ensure that the SAR presents applicable pressure testing criteria (e.g., test
9922 pressure, hold periods, inspections) and that these criteria are consistent with those presented
9923 in Chapter 9 of the SAR.

9924
9925 Administrative Programs (HIGH Priority)

9926
9927 The applicant may request that one or more administrative programs be approved by the NRC
9928 in lieu of the requirements set forth in Section 9.5.1 above for offsite releases, draining and
9929 drying, filling and pressurization, and welding and sealing. Requirements for such
9930 administrative programs are provided in NUREG-1745, "Standard Format and Content for
9931 Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance," and are
9932 summarized in this section.

9933
9934 The applicant may request the NRC approve an administrative program for offsite releases. In
9935 this case, the reviewer should verify that the SAR describes a Radioactive Effluent Control
9936 Program and related operating procedures that shall be established, implemented, and
9937 maintained to:

- 9938
- 9939 • Implement the requirements of 10 CFR 72.126.
 - 9940
 - 9941 • Limit the surface contamination and verification of meeting those limits prior to
9942 removal of the cask from the Part 50 structure.
 - 9943
 - 9944 • Limit the leakage rate and verification of meeting those limits prior to removal of
9945 the cask from the Part 50 structure.
 - 9946
 - 9947 • Show compliance with the requirements of 10 CFR 72.104 and 72.106.

9948
9949 In addition, the applicant may request the NRC approve an administrative program for cask
9950 loading. In this case, the reviewer should verify that the SAR requirements are implemented for
9951 loading fuel and components into the cask and preparing the cask for storage. The
9952 requirements of the program for loading and preparing the cask should be completed prior to
9953 removing the cask from the 10 CFR Part 50 structure. (Items 1, 5, and 6 below are associated
9954 with requirements that will remain in the technical specifications; however, the process for
9955 establishing the specified action limit may be moved to this administrative program if a method
9956 of evaluation acceptable to the NRC is presented in the SAR. Items 2, 3, and 4 have been
9957 relocated from the Limiting Conditions of Operations [LCO] section to this administrative
9958 program because it is felt that NRC-approved methods of evaluation will be relatively easy to
9959 develop. If appropriate methods are not presented in the SAR, these items will retain LCOs.)

9960
9961 At a minimum, the cask-loading program shall establish criteria that need to be verified to
9962 address SAR commitments and regulatory requirements for:

- 9963
- 9964 1. Vacuum drying times and pressures to assure that the short-term fuel
9965 temperature limits are not violated and the cask is adequately dry.
 - 9966
 - 9967 2. Inerting pressure and purity to assure adequate heat transfer and corrosion
9968 control.
 - 9969
 - 9970 3. Leak testing to assure adequate cask integrity and consistency with the offsite
9971 dose analysis.
 - 9972
 - 9973 4. Surface dose rates to assure proper loading and consistency with the offsite
9974 dose analysis.
 - 9975
 - 9976 5. Ambient and pool water temperature to assure adequate subcriticality and
9977 material ductility.
 - 9978
 - 9979 6. SNF pool boron concentration to verify the acceptable subcriticality margin.
 - 9980
 - 9981 7. Clad oxidation thickness for high-burnup fuel in accordance with SRP Chapter 8,
9982 "Materials Evaluation" or other NRC-approved methodology if high-burnup fuel is
9983 included in the contents.
 - 9984

9985 The program shall include compensatory measures and appropriate completion times if the
9986 program requirements are not met.

9987
9988 **9.5.2 Cask Handling and Storage Operations (LOW Priority)**

9989
9990 The reviewer should examine the recommendations associated with procedures necessary to
9991 transfer the cask to the storage location. The reviewer should pay particular attention to
9992 ensuring that all accident events applicable to such transfer are bounded by the design events
9993 analyzed in Chapters 2, "Principle design Criteria", 3, "Structural Evaluation" and 12, "Accident
9994 Analyses Evaluation" of the SAR. This includes procedures to be specified in the SAR for use
9995 after a design-basis accident for testing the effectiveness of the shielding. The structural and
9996 thermal disciplines should coordinate their review to ensure that all conditions for lifting and
9997 handling methods are bounded by the evaluations in their respective Chapters 3 and 4 of the
9998 SAR. There may be technical specifications associated with cask transfer operations such as
9999 restricting lift heights and environmental conditions (e.g., high/low temperatures, etc.) requiring
10000 coordination with the review in Chapter 13, "Technical Specifications and Operating Controls
10001 and Limits Evaluation," of this SRP.

10002
10003 The reviewer should verify that the procedure recommendations discuss the inspection,
10004 surveillance, and maintenance requirements that are applicable during ISFSI storage.
10005 Surveillance and monitoring requirements should also be included in Chapter 13 of the SAR,
10006 and maintenance should be included in Chapter 10 of the SAR. Reviewers should note that if
10007 the confinement vessel closure is bolted, the NRC staff generally requires that the successful
10008 operation of the seals be demonstrated with an initial leak test and a monitoring system and/or a
10009 surveillance program as discussed in Chapter 10, "Acceptance Tests and Maintenance Program
10010 Evaluation," of this SRP.

10011
10012 The shielding and radiation protection reviewers should verify that proposed procedures give
10013 due consideration to maintaining doses ALARA during cask handling and storage operations.

10014
10015 The applicant may request that an ISFSI Operations Program be approved by the NRC.
10016 Requirements for such an administrative program are provided in NUREG-1745. The reviewer
10017 should verify that such a program establishes criteria for:

- 10018
- 10019 • Minimum cask center-to-center spacing.
 - 10020
 - 10021 • Pad parameters (i.e., pad thickness, concrete strength, soil modulus,
10022 reinforcement, etc.) that are consistent with the SAR analysis.
 - 10023
 - 10024 • Maximum lifting heights for the cask system to ensure that the gravity load limits
10025 are met for the design-basis events.
 - 10026

10027 **9.5.3 Cask Unloading (Priority – as indicated)**

10028
10029 (LOW Priority) The reviewer should verify that the SAR adequately describes the necessary
10030 unloading procedure recommendations. The unloading procedure descriptions should present
10031 the activities sequentially in the anticipated order of performance, including those key
10032 prerequisite and preparation tasks that must be accomplished before cask unloading. Where
10033 applicable, the reviewer should verify that the procedure guidance ensures that any fluids, such
10034 as shield or borated water, fill their respective cavities according to design specifications.
10035 Additionally, for casks that require borated water to maintain subcriticality, the reviewer should
10036 ensure that the procedure guidance includes verification that the water to be used for cask
10037 reflood meets the minimum soluble boron content required by the Technical Specifications.

10038
10039 Damaged Fuel (LOW Priority)

10040
10041 The SAR should include appropriate additional measures for the potential presence of damaged
10042 fuel. Procedures should be designed to maximize worker protection from unanticipated
10043 radiation exposures or contaminants due to damaged fuel in accordance with ALARA principles
10044 and, to the maximum extent possible, prevent any uncontrolled releases to the environment.
10045 The following points outline the relevant safety concerns and an acceptable approach to
10046 address damaged fuel contingencies in cask unloading:

- 10047
- 10048 • The procedure descriptions should provide for fuel unloading under normal
10049 conditions.
 - 10050
 - 10051 • The unloading process should ensure that the fuel can be safely unloaded with
10052 regard to structural, criticality, thermal, and radiation protection considerations.
10053 This includes the provision for safe maintenance of the fuel and cask while any
10054 additional measures needed to address suspected damaged fuel are planned
10055 and implemented.
 - 10056
 - 10057 • The unloading process should reflect the potential for damaged fuel and
10058 changing radiological conditions.
 - 10059
 - 10060 • The process should include measures to check for and detect damaged fuel
10061 conditions (such as atmosphere samples) before opening the cask. (Note that

10062 fuel oxidation resulting from exposure to air at temperatures typical for dry
10063 storage is a known form of fuel degradation. Therefore, the presence of air in a
10064 cask designed to maintain an inert atmosphere indicates that the fuel may be
10065 degraded. The detection of fission gases is another indicator that the fuel may
10066 be degraded.)
10067

10068 The process may establish sample result thresholds above which damaged fuel is suspected.
10069 Other technically sound methods may be used to check for potential air leakage paths. Such
10070 methods may include designs that monitor cask internal pressure or seal integrity and alert the
10071 licensee to a problem before oxidation could occur. However, this method may not address
10072 detection of potential fuel degradation resulting from other mechanisms (such as a cask drop
10073 accident).

- 10074 • If the sample indicates normal conditions, the normal unloading process should
10075 be followed.
10076
- 10077 • If damaged fuel is suspected or found, the procedure description should stipulate
10078 that additional measures, appropriate for the specific conditions that include the
10079 canning of the damaged fuel, are to be planned, reviewed, and approved by the
10080 designated approval authority and implemented to minimize exposures to
10081 workers and radiological releases to the environment. These additional
10082 measures may include provision of filters, respiratory protection, and other
10083 methods to control releases and exposures in accordance with ALARA.
10084

10085
10086 Cooling, Venting, and Reflooding (LOW Priority)
10087

10088 The reviewer should verify that the SAR describes applicable operational measures to control
10089 cask cooling, venting, and reflooding (when appropriate). Also, the reviewer should verify that
10090 these measures are consistent with the results of the structural, materials, and thermal
10091 evaluations in the SAR, respectively. Cask cooling, venting, and reflooding should not result in
10092 damage to the fuel. Operational measures may include external cooling of the confinement
10093 cask for initial temperature reduction, restricting reflood flow rates to control and limit internal
10094 cask pressure from steam formation, and limiting cooldown rates.
10095

10096 Special attention should be devoted to reviews in this area since analysis of existing designs
10097 have predicted fuel temperatures during storage and transfer in excess of 533.15°K (500°F) for
10098 design-basis heat loads. Operational controls may be required to address the following
10099 potential effects during a cooldown and reflood evolution:
10100

- 10101 • Cask pressurization may occur as a result of steam formation as reflood water
10102 contacts hot surfaces.
10103
- 10104 • Excessive cooling rates may cause fuel cladding and fuel rod component
10105 damage and release of radioactive material as a result of stress (thermal, internal
10106 pressure, etc.) beyond material strengths (see SRP Section 8.4.17.1, "Cladding
10107 Temperature Limits").
10108
- 10109 • Excessive cooling rates may induce thermal stress that causes gross
10110 deformation of the fuel assembly components and subsequent binding with the
10111 basket.
10112

- 10113 • Cask supply and vent line failures from inadequate design for pressure and
10114 temperature could result in radiological exposures and personnel hazards (e.g.,
10115 steam burns).
10116

10117 Fuel Crud (LOW Priority)
10118

10119 The reviewer should verify that the procedure descriptions include contingencies for protection
10120 from fuel crud particulate material. Appendix E of ANSI/ANS 57.9 provides a short discussion of
10121 crud with respect to dry transfer systems. The unloading procedures should alert cask users to
10122 wait until any loose particles have settled and to slowly move the fuel assemblies to minimize
10123 crud dispersion in the SNF pool. Experience with wet unloading of boiling-water reactor (BWR)
10124 fuel after transportation has involved handling significant amounts of crud. This fine crud, which
10125 includes ⁶⁰Co and ⁵⁵Fe, will remain suspended in water or air for extended periods. The dry
10126 cask reflood process, during unloading of BWR fuel, has the potential to disperse crud into the
10127 fuel transfer pool and the pool area atmosphere, thereby creating airborne exposure and
10128 personnel contamination hazards. By contrast, no significant crud dispersal problems have
10129 been observed in handling pressurized-water reactor (PWR) fuel due to differences in the
10130 characteristics of crud on this type of fuel.
10131

10132 ALARA (LOW Priority)
10133

10134 The reviewer should verify that the procedure descriptions incorporate ALARA principles and
10135 practices. These may include provisions to perform radiological surveys, exposure and
10136 contamination control measures, temporary shielding, and suggested caution statements
10137 related to specific actions that could change radiological conditions. The reviewer should verify
10138 that any recommended surveys incorporate the applicable operating controls and limits
10139 described in Chapter 13, "Technical Specifications and Operation Controls and Limits
10140 Evaluation" of the SAR.
10141

10142 Offsite Release (LOW Priority)
10143

10144 Where applicable, the reviewer should verify that the SAR describes methods such as filtered
10145 ventilation, decontamination, or temporary containments to minimize offsite releases. The
10146 procedures should also provide for minimizing generation of radioactive waste.
10147

10148 Administrative Programs (HIGH Priority)
10149

10150 The applicant may request that the NRC approve an administrative program for cask unloading.
10151 NUREG-1745 provides requirements for such an administrative program. The reviewer should
10152 verify the proposed administrative program meets the requirements summarized in
10153 Section 9.5.1 of this SRP.
10154

10155 **9.6 Evaluation Findings**
10156

10157 The reviewer should examine the 10 CFR Part 72 acceptance criteria and provide a summary
10158 statement for each. These statements should be similar to the following model, as applicable:
10159

- 10160 F9.1 The [cask designation] is compatible with [wet/dry] loading and unloading.
10161 General procedure descriptions for these operations are summarized in
10162 Chapter(s) _____ of the applicant's safety analysis report (SAR). Detailed
10163 procedures will need to be developed and evaluated on a site-specific basis.

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- F9.2 The [welded/bolted lids or other features] of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.
- F9.3 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.
- F9.4 No significant radioactive waste is generated during operations associated with the independent spent fuel storage installation (ISFSI). Contaminated water from the spent fuel pool will be governed by the 10 CFR Part 50 license conditions.
- F9.5 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading will be governed by the 10 CFR Part 50 license conditions.
- F9.6 The content of the general operating procedures described in the SAR are adequate to protect health and minimize damage to life and property. Detailed procedures will need to be developed and approved on a site-specific basis.
- F9.7 The radiation protection chapter of this SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

The reviewer should provide a summary statement similar to the following:

“The staff concludes that the generic 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 procedure descriptions provided in the SAR offers 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.”

10197 **10 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM EVALUATION**

10198

10199 **10.1 Review Objective**

10200

10201 The acceptance tests and maintenance program review ensures that the applicant’s Safety
10202 Analysis Report (SAR) includes the appropriate acceptance tests and maintenance programs
10203 for the system. A clear, specific listing of these commitments will help avoid ambiguities
10204 concerning design, fabrication, and operational testing requirements when the U.S. Nuclear
10205 Regulatory Commission (NRC) staff conducts subsequent inspections. Acceptance tests may
10206 also be described in the applicable chapter of this Standard Review Plan (SRP).

10207

10208 **10.2 Areas of Review**

10209

10210 This chapter of the dry storage system (DSS) SRP provides guidance for use in evaluating the
10211 acceptance tests and maintenance programs outlined in the SAR. The acceptance tests
10212 demonstrate that the cask has been fabricated in accordance with the design criteria and that
10213 the initial operation of the cask complies with regulatory requirements. The maintenance
10214 program describes actions that the licensee needs to implement during the storage period to
10215 ensure that the cask performs its intended functions.

10216

10217 As defined in Section 10.5, “Review Procedures,” a comprehensive evaluation *may* encompass
10218 the following acceptance tests and maintenance programs:

10219

10220 ***Acceptance Tests***

10221

Structural/Pressure Tests

10222

Leak Tests

10223

Visual and Nondestructive Examination Inspections

10224

Shielding Tests

10225

Neutron Absorber Tests

10226

Thermal Tests

10227

Cask Identification

10228

10229 ***Maintenance Program***

10230

Inspection

10231

Tests

10232

Repair, Replacement, and Maintenance

10233

10234 **10.3 Regulatory Requirements**

10235

10236 This section presents a summary matrix of the portions of U.S. Code of Federal Regulations
10237 (CFR), Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel
10238 High-Level Radioactive Waste and Reactor-Related Greater Than Class C Waste,” Title 10,
10239 “Energy” (10 CFR Part 72) that are relevant to the review areas addressed by this chapter. The
10240 NRC staff reviewer should read the exact referenced regulatory language. Table 10-1 matches
10241 the relevant regulatory requirements associated with this chapter to the areas of review
10242 identified in the previous section.

10243

Table 10-1 Relationship of Regulations and Areas of Review

Areas of Review	10 CFR Part 72 Regulations							
	72.82 (d)	72.122 (a), (f)	72.124 (b)	72.162	72.212 (b)(8)	72.232 (b)	72.236 (c)	72.236 (g), (j), (k), (l)
Acceptance Tests	•	•	•	•		•		•
Maintenance Program	•	•						•
Design Verification	•	•			•	•	•	•

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10.4 Acceptance Criteria

In general, the acceptance tests and maintenance programs outlined in the SAR should cite appropriate authoritative codes and standards. The staff has previously accepted the following as the regulatory basis for the design, fabrication, inspection, and testing of DSS components:

System/Component	Acceptable Regulatory Basis*
Confinement System	<ul style="list-style-type: none"> American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel (B&PV) Code," Section III, Division 1, 2007 "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment" (ANSI N14.5)
Confinement Internals (e.g., basket)	<ul style="list-style-type: none"> ASME B&PV Code, Section III, Subsection NG
Metal Cask Overpack	<ul style="list-style-type: none"> ASME B&PV Code, Section VIII
Concrete Cask Overpack	<ul style="list-style-type: none"> American Concrete Institute (ACI), "Code Requirements for Structural Concrete" (ACI-318), "Code Requirements for Nuclear Safety Related Concrete" (ACI-349), as appropriate
Other Metal Structures	<ul style="list-style-type: none"> ASME B&PV Code, Section III, Subsection NF American Institute of Steel Construction (AISC), "Manual of Steel Construction"
* The SAR should clearly identify any exceptions to the listed codes and standards (see SRP Chapter 13, "Technical Specifications and Operating Controls and Limits Evaluation").	

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10.5 Review Procedures

Introduction

Figures 10-1 and 10-2 present an overview of the evaluation process and can be used as a guide to assist in coordinating with the review disciplines.

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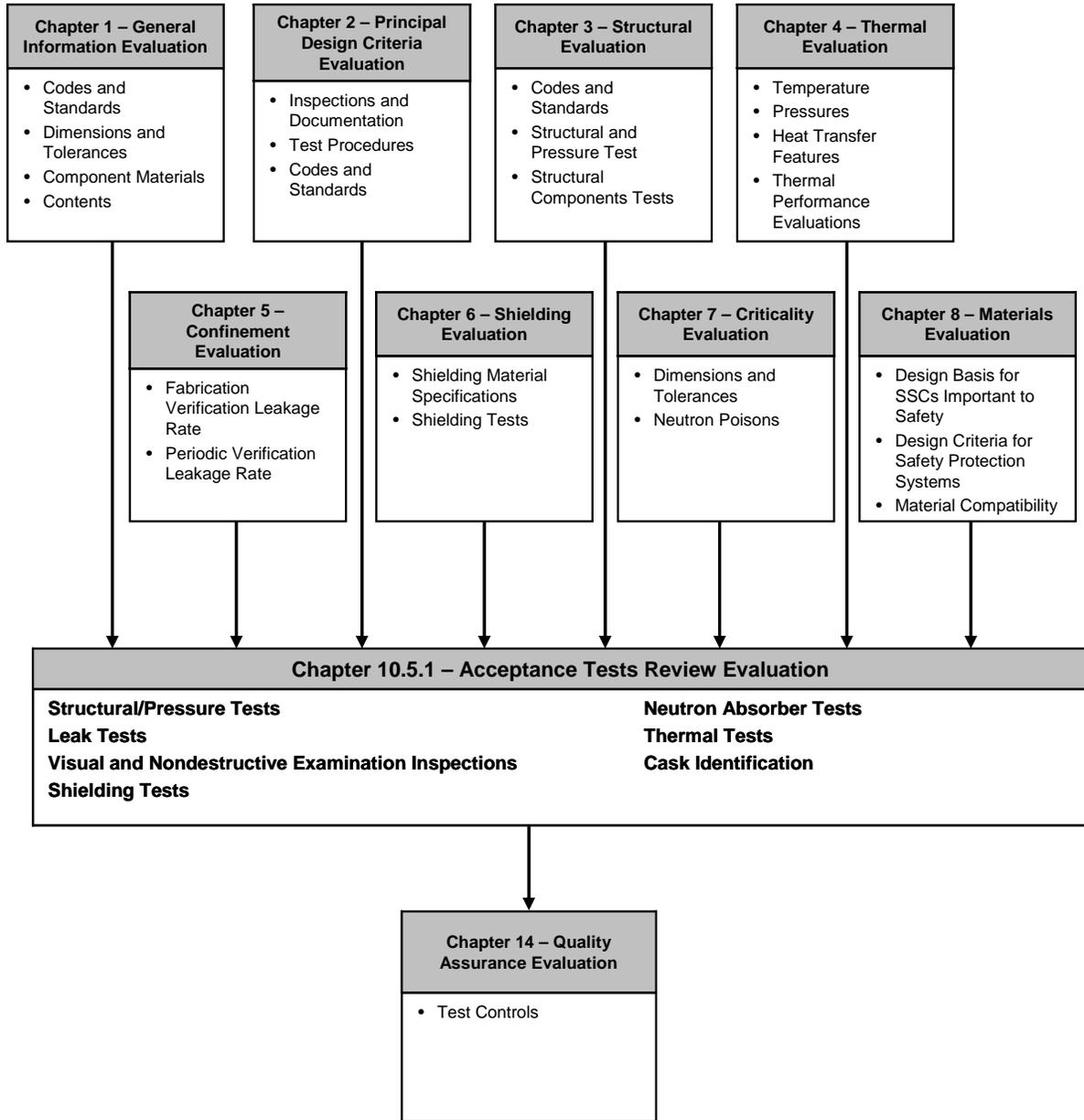


Figure 10-1 Overview of Acceptance Test Review Evaluation

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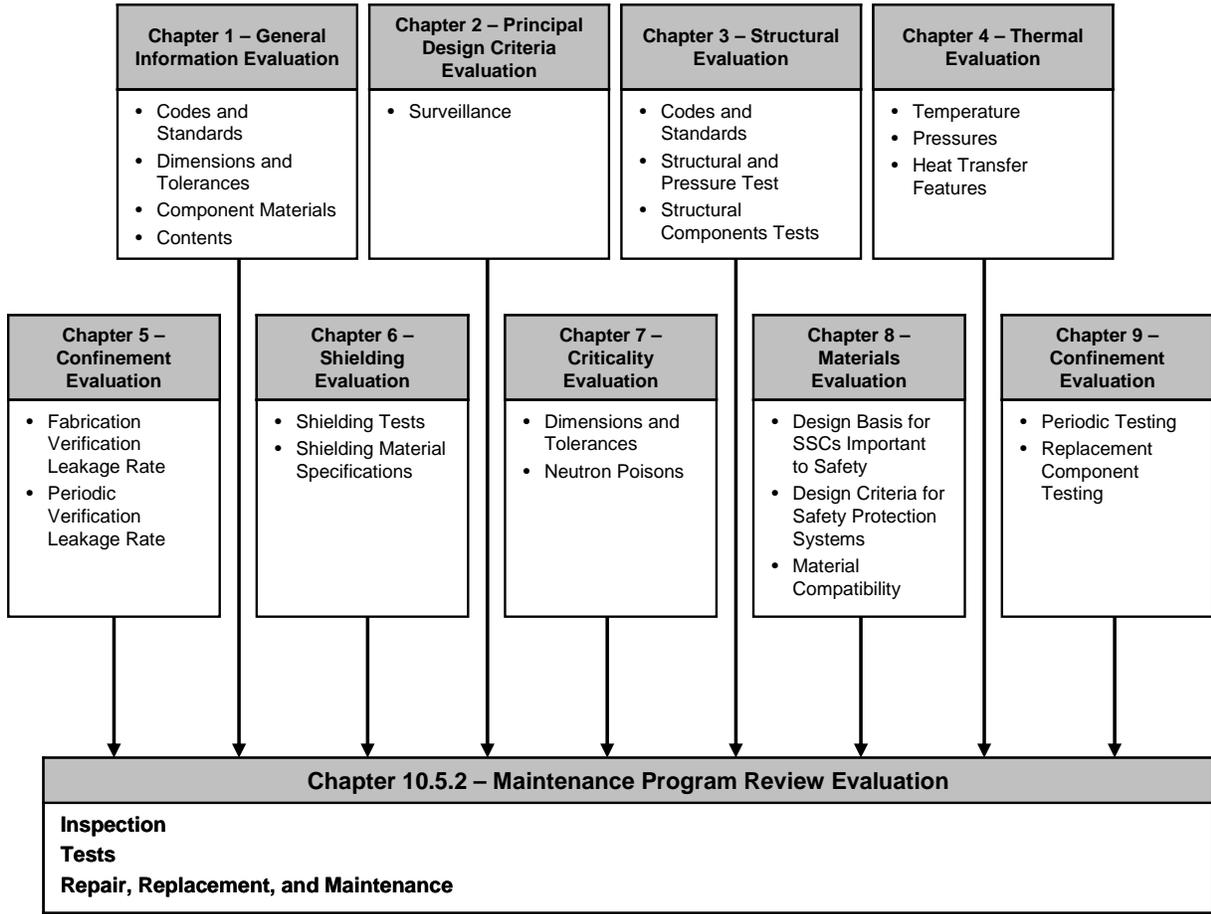


Figure 10-2 Overview of Maintenance Program Review Evaluation

10338 The review procedures described in this section are presented in a format intended to facilitate
10339 a single, independent review. Although one or more individual(s) may be tasked with preparing
10340 the corresponding section of the safety evaluation report (SER) related to the proposed
10341 acceptance tests and maintenance program, all review team members should examine the
10342 related information presented in the SAR. Information in the SAR related to the acceptance
10343 tests may be located in the chapters related to specific disciplines (i.e., SAR Chapter 4,
10344 "Thermal Evaluation") and/or in SAR Chapter 10, "Acceptance Tests and Maintenance
10345 Program"). Reviewers should devote special attention to those tests (or the lack of tests) that
10346 affect their functional area of review. If the descriptions included in the SAR are not sufficiently
10347 detailed to allow a complete evaluation concerning fulfillment of the acceptance criteria,
10348 reviewers should request additional information from the applicant.

10349
10350 In general, applicants commit to design, construct, and test the system under review to the
10351 codes and standards identified in SAR Chapter 2, "Principal Design Criteria." The NRC does
10352 not generally review specific test and maintenance procedures as part of the licensing process;
10353 however, the applicant is expected to describe (in the SAR) certain elements of the proposed
10354 test and maintenance programs. The staff may inspect selected portions of test procedures as
10355 part of its onsite activities.

10356
10357 The following subsections provide *representative examples* of test and maintenance program
10358 elements that should be subject to licensing review. If included in the SAR, each of these tests
10359 and maintenance elements should be reviewed to ensure that the applicant has identified the
10360 purpose of the test, explained the proposed test method (including any applicable standard to
10361 which the test will be performed), defined the acceptance criteria and bases for the test, and
10362 described the actions to be taken if the acceptance criteria are not satisfied.

10364 **10.5.1 Acceptance Tests (Priority – as indicated)**

10365
10366 The following guidance is presented on the basis of tests deemed acceptable by the staff in
10367 previous SAR reviews. Alternative tests and criteria may be used if the SAR provides
10368 appropriate explanation and adequate justification.

10370 10.5.1.1 Structural/Pressure Tests

10371
10372 (MEDIUM Priority) Lifting trunnions should be fabricated and tested in accordance with ANSI
10373 N14.6, "American National Standard for Radioactive Materials-Special Lifting Devices for
10374 Shipping Containers Weighing 10,000 pounds (4,500 Kilograms) or More." Site-specific details
10375 of the pool and lifting procedures may enable the cask to be considered a non-critical load, as
10376 defined by this standard. Generally, however, the cask is considered a critical load during its
10377 handling in the pool. Consequently, trunnion testing should be performed at a minimum of
10378 150 percent of the maximum service load, if redundant lifting is employed or 300 percent of the
10379 service load if non-redundant lifting applies. These load tests should be performed to ensure
10380 that the trunnions and cask are conservatively constructed and provide an adequate margin of
10381 safety when filled with SNF. Trunnion load testing should also be performed annually for the
10382 transfer cask and at least one year before use for the storage cask. Load testing of integral
10383 trunnions is not required once the loaded storage cask has been placed on the pad.
10384 Restrictions on cask lifting resulting from these tests should be included in Chapter 13,
10385 "Technical Specifications and Operating Controls and Limits Evaluation," of the SAR and the
10386 related SER prepared by the NRC staff. SAR Chapter 10, "Acceptance Tests and Maintenance
10387 Program Evaluation" should explicitly state the testing values.

10388

10389 (MEDIUM Priority) The entire confinement boundary should be pressure tested hydrostatically
10390 or pneumatically to 125 or 110 percent of the design pressure, respectively. The pressure test
10391 should be performed in accordance with governing code associated with the confinement
10392 boundary, which typically has been ASME B&PV Code, Section III, Division 1, Subsection NB or
10393 NC. The test pressure should be maintained for a minimum of 10 minutes, after which a visual
10394 inspection should be performed to detect any leakage. SAR Chapter 10, "Acceptance tests and
10395 Maintenance Program Evaluation" should clearly specify the hydrostatic and pneumatic test
10396 pressures. The helium leakage test, per ANSI 14.5 is not considered as a substitute for the
10397 Code required pressure test, and conversely, the Code pressure test is not a substitute for the
10398 helium leakage test. If a shop pressure test isn't performed and only a field pressure test is
10399 performed after the first closure weld is made, the staff has accepted the shop helium leakage
10400 test as meeting the pressure test acceptance criteria of no leakage for the shell welds since they
10401 are generally inaccessible in the field.

10402
10403 (LOW Priority) Some casks contain a neutron shielding material that may off-gas at higher
10404 temperatures. Such material is usually contained inside a thin steel shell to prevent loss of
10405 mass and provide protection from minor accidents and natural phenomenon events. Rupture
10406 disks or relief valves are generally provided to prevent catastrophic failure of this shell. The
10407 shell should be tested to 125 percent of the rupture disk burst pressure, which is usually
10408 equivalent to 125 percent of the shell design pressure. The SAR should clearly specify the
10409 burst pressure for the rupture disk, along with its coincident burst temperature and tolerance on
10410 burst pressure.

10411
10412 (HIGH Priority) Some cask designs use ferritic steels that are subject to brittle fracture failures at
10413 low temperature. ASME B&PV Code, Section II, Part A, contains procedures for testing ferritic
10414 steel used in low temperature applications. On the basis of guidance in NUREG/CR-1815,
10415 "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping
10416 Containers Up to Four Inches Thick," Section 5.1.1, the NRC established two methods for
10417 identifying suitable materials:

- 10418
10419 • The nil-ductility transition (NDT) temperature must be determined by either direct
10420 measurement, (American Society for Testing and Materials' (ASTM) "Method of
10421 Conducting Drop Weight Test to Determine Nil-ductility Transition Temperature
10422 for Ferritic Steel" [ASTM E-208]) or indirect measurement ("Dynamic Tear
10423 Testing of Metallic Materials" [ASTM E-604]), and the minimum operating
10424 temperature of the steel must be specified as 28°C (50°F) higher than the NDT.
- 10425
10426 • The NRC staff accepts ASME Charpy testing procedures for verification of the
10427 material's minimum absorbed energy. Acceptable energy absorption values and
10428 test temperatures of Charpy, V-Notch impact tests are listed in the ASME B&PV
10429 Code, Section II, SA-20, "Specifications for General Requirements for Steel
10430 Plates for Pressure Vessels" Table A1.15. Coordinate with the thermal review
10431 (Chapter 4 of this SRP) to ensure that the applicant selected the correct
10432 temperatures for the tests and that the SAR specifies the method of testing.

10433
10434 For casks with ferritic steel walls thicker than 102 mm (4 in.), the guidance provided in
10435 NUREG/CR-3826, "Recommendations for Protecting Against Failure by Brittle Fracture in
10436 Ferritic Steel Shipping Containers Greater than Four Inches Thick," should be followed.
10437

10438 10.5.1.2 Leak Tests (LOW Priority)

10439

10440 The licensee should perform leak tests on all confinement boundaries except as excluded in
10441 Chapter 8, "Materials Evaluation" - Section 8.9.2, which only applies to the closure welds
10442 typically made in the field. For all-welded cask confinements, the NRC staff has, with adequate
10443 justification, considered it acceptable for licensees to omit leak testing of the second cask
10444 closure weld and the seal welds for the closure plates of the purge and vent valves (if not
10445 potentially pressurized at the time of welding). For such cases, leak testing must show that the
10446 inner closure weld meets the leakage limits. A fabrication leak test should be performed on
10447 every canister in the shop to ensure that the tested leakage rate is compatible with the
10448 regulatory dose limits at the controlled area boundary, 10 CFR 72.236(d), (i), and (j).

10449

10450 Leakage criteria in units of Pa.m³/s or reference cm³/s must be at least as restrictive as those
10451 specified in the principal design criteria (in SAR Chapter 2). The SAR should also indicate the
10452 general testing methods (e.g., pressure increase, mass spectrometer) and required sensitivities.
10453 If cask closure depends on more than one seal (e.g., lid, vent port, drain port), the leakage
10454 criteria should ensure that the total leakage is within the design requirements. Leak testing
10455 should be conducted in accordance with ANSI N14.5.

10456

10457 10.5.1.3 Visual and Nondestructive Examination Inspections

10458

10459 (HIGH Priority) Reviewers should verify the applicant's commitment to fabricate and examine
10460 cask components in accordance with an accepted design standard such as ASME B&PV Code,
10461 Section III or VIII. These sections define the examination requirements mentioned in Section II,
10462 "Materials Specifications and Properties"; Section V, "NDE Specifications and Procedures"; and
10463 Section IX, "Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and
10464 Welding and Brazing Operators." The following guidance assumes that the ASME B&PV Code
10465 is applicable to the cask being reviewed.

10466

10467 (HIGH Priority) The nondestructive examination (NDE) of weldments must be well-characterized
10468 on drawings, using standard NDE symbols and/or notations (see American Welding Society's
10469 (AWS) "Standard Symbols for Welding, Brazing, and Nondestructive Examination" [AWS A2.4]).
10470 Each fabricator should be required to establish and document a detailed, written weld inspection
10471 plan in accordance with an approved quality assurance (QA) program that complies with
10472 10 CFR Part 72, Subpart G. The inspection plan should include visual (VT), dye penetrant (PT),
10473 magnetic particle (MT), ultrasonic (UT), and radiographic (RT) examinations, as applicable. The
10474 inspection plan should identify welds to be examined, the examination sequence, type of
10475 examination, and the appropriate acceptance criteria as defined by either the ASME B&PV
10476 Code, or an alternative approach proposed and justified by the applicant. Inspection personnel
10477 should be qualified, in accordance with the current revision of the American Society for
10478 Nondestructive Testing's (SNT) "Personnel Qualification and Certification in Nondestructive
10479 Testing" (SNT-TC-1A), as specified by the ASME B&PV Code. All weld-related NDE should be
10480 performed in accordance with written and approved procedures. Fabrication controls and
10481 specifications should be in-place and field tested to prevent post-welding operations (such as
10482 grinding) from compromising the design requirements (such as wall thickness).

10483

10484 (HIGH Priority) Confinement boundary non-closure welds should meet the requirements of
10485 ASME B&PV Code, Section III, Division 1, Subsections NB or NC, Article NB/NC-5200,
10486 "Required Examination of Welds for Fabrication and Preservice Baseline." This section requires
10487 volumetric examination and either PT or MT for all Category A and most Category B or
10488 Category C welded joints in vessels, and longitudinal or full penetration welded joints in other

10489 components. The ASME-approved specifications for RT, UT, PT, and MT are detailed in ASME
10490 B&PV Code, Section V, Articles 2, 4, 6, and 7, respectively.

10491
10492 (HIGH Priority) Acceptance standards for nondestructive testing should be in accordance with
10493 ASME B&PV Code, Section III, Division 1, Subsection NB or NC -5300. Testers should reject
10494 unacceptable imperfections (such as a crack, a zone of incomplete fusion or penetration,
10495 elongated indications with lengths greater than specified limits, and rounded indications in
10496 excess of the limits in ASME B&PV Code, Section III, Division 1, Appendix VI). Repaired welds
10497 should be reexamined in accordance with the original examination method and associated
10498 acceptance criteria.

10499
10500 (HIGH Priority) For confinement welds that cannot be volumetrically examined using RT, the
10501 licensee may use 100 percent UT. The ASME-approved UT specifications are detailed in
10502 ASME B&PV Code, Section V, Article 4. Acceptance criteria should be defined in accordance
10503 with ASME B&PV Code, Section III, Division 1, Subsection NB or NC-5330, "Ultrasonic
10504 Acceptance Standards." Cracks, lack of fusion, or incomplete penetration are unacceptable,
10505 regardless of length.

10506
10507 (HIGH Priority) The NRC has accepted multiple surface examinations of welds, combined with
10508 helium leak tests for inspecting the final redundant seal welded closures.

10509
10510 (HIGH Priority) For confinement internals, the licensee should perform all NDE testing in
10511 accordance with ASME B&PV Code, Section III, Division 1, Subsection NG.

10512
10513 (LOW Priority) Nonconfinement welds (which exclude welds of confinement internals) should
10514 meet the requirements of ASME B&PV Code, Section III, Subsection NF, or Section VIII,
10515 Division 1, as applicable. The required volumetric examination of welds is either RT or UT, as
10516 discussed in ASME B&PV Code, Section III, NF-5200, and Section VIII, UW-11. The
10517 appropriate specifications from ASME B&PV Code, Section V, are invoked in Article 2 for RT
10518 and in Article 5 for UT. Acceptance standards for RT are detailed in ASME B&PV Code,
10519 Section III, Subsection NF, NF-5320, "Radiographic Acceptance Standards," and for UT in
10520 NF-5330, "Ultrasonic Acceptance Standards." For Section VIII weldments, RT acceptance
10521 criteria should be in accordance with ASME B&PV Code, Section VIII, Division 1, UW-51, and
10522 the repair of unacceptable defects should be in accordance with UW-38. Repaired welds
10523 should be reexamined in accordance with the original acceptance criteria.

10524
10525 (LOW Priority) Nonconfinement welds that cannot be examined using RT should undergo UT in
10526 accordance with ASME B&PV Code, Section V, Article 4. Acceptance criteria should be in
10527 accordance with ASME B&PV Code, Section VIII, Division 1, UW-53 and Appendix 12, and the
10528 repair of unacceptable defects should be in accordance with UW-38. Repaired welds should be
10529 reexamined in accordance with the original examination methods and associated acceptance
10530 criteria. If applicable, the SAR should also justify the rationale for not requiring RT examination
10531 of these welds.

10532
10533 (LOW Priority) Nonconfinement welds for cask system components that are designed and
10534 fabricated in accordance with ASME B&PV Code, Section III, that cannot be examined using RT
10535 or UT should undergo PT or MT examination in accordance with ASME B&PV Code, Section V,
10536 Articles 6 and 7, respectively. Acceptance criteria should be in accordance with Articles
10537 NF-5350 and NF-5340, respectively. Repaired welds should be reexamined in accordance with
10538 the original acceptance criteria. If applicable, the SAR should also justify the rationale for not
10539 requiring volumetric inspection techniques (RT or UT) for these welds.

10540
10541 (LOW Priority) Finished surfaces of the cask should be visually examined in accordance with
10542 the ASME B&PV Code Section V, Article 9. For welds examined using VT, the acceptance
10543 criteria should be in accordance with ASME B&PV Code, Section VIII, Division 1, UW-35 and
10544 UW-36, or NF-5360, "Acceptance Standards for Visual Examination of Welds."
10545

10546 (HIGH for confinement/LOW for non-confinement) The licensee should use PT to detect
10547 discontinuities (such as cracks, seams, laps, laminations, and porosity) that open to the surface
10548 of nonporous metals. PT should be performed in accordance with ASME B&PV Code,
10549 Section V, Article 6. Acceptance criteria for PT examination of confinement welds should be in
10550 accordance with ASME B&PV Code, Section III, Subsection NB/NC, Article NB/NC-5350.
10551 Repair procedures should be in accordance with NB/NC-4450 of the ASME B&PV Code,
10552 Section III. Acceptance criteria for PT examination of nonconfinement welds should be in
10553 accordance with ASME B&PV Code, Section VIII, Division 1, Appendix 8, or NF-5350, "Liquid
10554 Penetrant Acceptance Standards." Repair procedures should be in accordance with ASME
10555 B&PV Code, Section III or NF-2500, "Examination and Repair of Material," and NF-4450,
10556 "Repair of Weld Material Defects."
10557

10558 10.5.1.4 Shielding Tests (LOW Priority)

10559

10560 The materials that comprise the DSS should sufficiently maintain their physical and mechanical
10561 properties during all conditions of operations. DSS gamma shielding materials (e.g., lead)
10562 should not experience slumping or loss of shielding effectiveness to an extent that compromises
10563 safety. The shield should perform its intended function throughout the licensed service period.
10564

10565 DSS materials used for neutron absorption should be designed to perform their safety function
10566 without degradation, gas release, or physical alteration for the full term of the license. Tests are
10567 required to ensure these conditions are met.
10568

10569 Tests of the effectiveness of both the gamma and neutron shielding may be required if, for
10570 example, the cask contains a poured lead shield or a special neutron absorbing material. In
10571 such instances, the SAR should describe any scanning or probing with an auxiliary source for
10572 the purpose of characterizing the shielding. This shield testing should be done for every cask
10573 that uses poured shielding material, to demonstrate proper fabrication in accordance with the
10574 design drawings. The suggested shield test applies equally to both storage and transfer casks.
10575

10576 In addition to the above tests, the licensee should perform dose rate measurements after the
10577 SNF is loaded to establish that the stated design criteria have been satisfied.
10578

10579 10.5.1.5 Neutron Absorber Tests (HIGH Priority)

10580

10581 Neutron absorber materials require both qualification and acceptance testing to provide
10582 assurance that the control of criticality by absorbing thermal neutrons will be assured in systems
10583 designed for nuclear fuel storage, transportation or both. Both qualification and acceptance
10584 testing are in general as described in ASTM Designation C1671, "Standard Practice for
10585 Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality
10586 Control for Dry Storage Systems and Transportation Packaging."
10587

10588 Acceptance tests are used to ensure that material properties for plates and other shapes
10589 produced in a given production run are in compliance with the materials requirements of the
10590 application. In one sense, acceptance tests verify that the material of a given production run

10591 has yielded products that have been shown to be like the products that were used in the
10592 qualification testing. Acceptance tests are used to ensure that the production process is
10593 operating in a satisfactory manner, and they use statistical data for selected measurable
10594 parameters. For all boron-containing absorber materials, acceptance tests should (a) verify ¹⁰B
10595 content and uniformity, (b) require visual examinations to establish only acceptable levels of
10596 defects are present from cracks, porosity, blisters, or foreign inclusions, and (c) make
10597 dimensional (e.g., plate thickness which is important to the areal density).
10598

10599 Some materials may obtain 100 percent credit for the amount of ¹⁰B that is shown to be present
10600 in the absorber materials. This level of credit is sometimes called 90 percent credit because the
10601 credit level refers to a manner in which K-effective calculations are conducted and in these
10602 calculations, any absorber is given a 10 percent penalty before being used in the calculation.
10603 Likewise other materials that are given only 82 percent credit are called materials with
10604 75 percent credit. For purposes of obtaining high levels (100 percent) of credit, the amount of
10605 ¹⁰B, which is the absorber species, is assessed in boron-containing absorber materials using
10606 neutron attenuation testing.
10607

10608 Neutron attenuation tests are calibrated using appropriate standards such as those based on
10609 (coated with) zirconium diboride (ZrB₂) plates to ensure the accuracy of the measured values.
10610 Approved substitutes may be used for the attenuation tests. These include tests such as
10611 chemical analysis, provided that (1) both the neutron attenuation tests and the alternative tests
10612 have at least the sensitivity of tests specified in C-1671 and (2) the alternate form of testing is
10613 regularly bench marked against calibrated neutron attenuation tests. Chemical analyses should
10614 also include spectrochemical analysis for material impurity levels and ¹⁰B content. Uniformity is
10615 assessed using statistical sampling techniques that ensure that the entire plate of material and
10616 all plates in a lot meet a 95/95 criterion, which means that a test result has a 95 percent
10617 likelihood of containing the minimum required amount of ¹⁰B and that this is known at the 95
10618 percent confidence level.
10619

10620 Acceptance tests are used to ensure that material properties for plates and other shapes
10621 produced in a given production run are in compliance with the materials requirements of the
10622 application. In one sense, acceptance tests verify that the material of a given production run
10623 has yielded products that have been shown to be like the products that were used in the
10624 qualification testing. Acceptance tests are used to ensure that the production process is
10625 operating in a satisfactory manner, and they use statistical data for selected measurable
10626 parameters. For all boron-containing absorber materials, acceptance tests should (a) verify ¹⁰B
10627 content and uniformity, (b) require visual examinations to establish only acceptable levels of
10628 defects are present from cracks, porosity, blisters, or foreign inclusions, and (c) make
10629 dimensional (e.g., plate thickness which is important to the areal density).
10630

10631 The reviewer should confirm that the calculation of minimum poison content (e.g., poison areal
10632 density) conservatively accounts for tolerance limits on material density, poison concentration,
10633 and component dimensions. It is noted that thickness tolerances on rolled plates, sheets or
10634 shape are typically on the order of ± 10 percent. The acceptance testing should provide a
10635 representative sampling of coupons for plates or sheets from a given lot. Statistical sampling
10636 can be used to the extent practical, using test locations on a coupon that will account for local
10637 variations or anomalies within the coupon and hence within the plates represented by the
10638 coupon. Adequate numbers of samples should be taken to ensure the confidence level required
10639 for the application.
10640

10641 Acceptance Testing of Fabricated Materials for 75-Percent Boron Credit

10642

10643 For multi-phase absorber materials analyzed with 75-percent poison credit (or less) the reviewer
10644 should confirm that acceptance testing is consistent with the following:

10645

10646 • The effective ^{10}B content should be verified from plate coupons by either
10647 (a) neutron attenuation testing, or (b) chemical assay to determine boron content
10648 with mass spectrometric analysis for isotopic composition.

10649

10650 • Sufficient coupons should be taken for acceptance testing to justify the level of
10651 credit given. Rejection of a coupon should result in rejection of the plate from
10652 which it is taken. Sampling may be reduced to lesser percentages of the
10653 coupons taken (e.g., to 50 percent of all coupons) after acceptance of all
10654 coupons in the first 25 percent of the lot. A rejection during reduced inspection
10655 should invoke a 100 percent inspection for coupons from that lot.

10656

10657 • A visual examination of all plates for defects should be conducted.

10658

10659 Acceptance Testing for Greater Than 75 Percent Boron Credit

10660

10661 For acceptance testing of borated absorbers at levels of poison credit beyond 75 percent, the
10662 extent of the acceptance testing and inspection is enhanced. Some of the data helpful in
10663 meeting the guidance in C-1671 Sec 5.3.4 are as follows:

10664

10665 • The effective ^{10}B content is verified by neutron attenuation testing of coupons.
10666 An adequate number of coupons should be acceptance tested for each lot of
10667 materials to statistically demonstrate that the 95/95 criterion is satisfied for the
10668 minimum required ^{10}B content. The minimum areal density is specified in the
10669 SAR. Note that if the coupon from a plate fails the single neutron attenuation
10670 measurement, the associated plate is rejected unless acceptable alternative
10671 testing is done with acceptable results.

10672

10673 • Sufficient coupons should be taken to satisfy the 95/95 criterion. For example,
10674 coupons are taken from at least every other plate unless justification for fewer is
10675 given. Measurements are made on samples taken from 100 percent of all
10676 coupons. Rejection of a coupon should result in rejection of the plate. Sampling
10677 may be reduced to 50 percent of all coupons after acceptance of all coupons in
10678 the first 25 percent of the lot. A rejection during reduced inspection should
10679 invoke a return to 100 percent inspection for that lot.

10680

10681 • A statistical analysis of the neutron attenuation results should be performed by
10682 the applicant for all plates in a lot. This analysis shall show that the lot meets the
10683 95/95 criterion. That is, using a one-sided tolerance limit factor for a normal
10684 distribution with at least 95 percent probability, the areal density is greater than or
10685 equal to the specified minimum value with 95 percent confidence level. Failure to
10686 meet this acceptance criterion of this statistical analysis shall result in rejection of
10687 the entire lot for use at the 100 percent (90 percent credit in K-effective
10688 calculations). Applicants may choose to convert all areal densities determined by
10689 neutron attenuation to a volume density by dividing by the thickness of the
10690 coupon. The one side tolerance limit of volume density with 95 percent
10691 probability and 95 percent confidence may then be determined. The minimum

10692 specified value of the areal density may be divided by the 95/95 lower tolerance
10693 limit of ¹⁰B volume density to arrive at the minimum plate thickness. Hence, all
10694 plates which have any locations thinner than this minimum shall be rejected, and
10695 those equal to or thicker may be accepted.

- 10696
- 10697 • A visual examination of all plates for defects should be verified.
- 10698

10699 The reviewer should refer to Section 8.4.13.2 of this SRP regarding how to compute per level of
10700 credit.

10701

10702 10.5.1.6 Thermal Tests (LOW Priority)

10703

10704 Depending on the details of the cask design and the ability to determine its heat removal
10705 capability through thermal analysis, testing may be required to verify cask performance. The
10706 applicant should establish acceptance criteria on the basis of the conditions of the test (e.g., test
10707 heat loading, ambient conditions). SAR Chapter 4, "Thermal Evaluation," should discuss the
10708 correlation between test performance and actual loading conditions to avoid ambiguous or
10709 unreviewed analysis after the test data are obtained.

10710

10711 10.5.1.7 Cask Identification (LOW Priority)

10712

10713 The vendor/licensee must mark the cask with a model number, unique identification number,
10714 and empty weight. Generally this information will appear on a data plate, which should be
10715 detailed in one of the drawings included in SAR Chapter 1, "General Description." In addition,
10716 the vendor/licensee should mark the exterior of shielding casks or other structures that may hold
10717 the confinement cask while it is in storage. This marking should provide a unique, permanent,
10718 and visible number to permit identification of the cask stored therein.

10719

10720 **10.5.2 Maintenance Program (LOW Priority)**

10721

10722 Storage casks are typically designed as passive units requiring minimal maintenance. The SAR
10723 should address the following areas, as applicable:

10724

10725 10.5.2.1 Inspection

10726

10727 Usually, the cask has at least one monitoring system (e.g., pressure, temperature, dosimetry).
10728 The SAR should discuss how such systems will be used to provide information regarding
10729 possible off-normal events and what surveillance actions may be necessary to ensure that these
10730 systems function properly. Detailed procedures will be developed and implemented by the
10731 licensee at the site.

10732

10733 The SAR should describe routine periodic visual surface and weld inspections, which should be
10734 limited to the readily accessible surfaces (i.e., the exterior surface of the storage cask and all
10735 surfaces of empty transfer casks). In addition, the SAR should discuss inspection of lifting and
10736 rotating trunnion load-bearing surfaces.

10737

10738 10.5.2.2 Tests

10739

10740 The SAR should describe any periodic tests of cask components or calibration of monitoring
10741 instrumentation, as well as periodic tests to verify shielding and thermal capabilities. The SAR
10742 should also describe procedures for any applicable periodic testing of neutron poison

10743 effectiveness. As an alternative to the licensee's periodic testing of neutron poison
10744 effectiveness, the applicant may show continued poison effectiveness in the manner described
10745 in Section 7.5.3.2 of this SRP. The qualification tests of the poison material, discussed in
10746 Section 8.4.13.3 of this SRP, may also be useful in showing the material's continued
10747 effectiveness.
10748

10749 In addition, the SAR should discuss any routine testing of support systems (e.g., vacuum drying,
10750 helium backfill, and leak testing equipment).
10751

10752 10.5.2.3 Repair, Replacement, and Maintenance

10753 The SAR should discuss the repair and replacement of cask components, as may be required
10754 during the lifetime of the storage and transfer casks. This discussion should include methods of
10755 repair or replacement, testing procedures, and acceptance criteria. The SAR should also
10756 describe procedures for routine maintenance (such as lubrication and re-application of corrosion
10757 inhibiting materials in the event of scratches) through the expiration of the service life of the
10758 equipment. Such information is also often included in SAR Chapter 12, "Accident Analyses,"
10759 which describes actions to be taken following an off-normal event or accident-level condition.
10760
10761

10762 10.6 Evaluation Findings

10763 The 10 CFR Part 72 acceptance criteria should be reviewed with a summary statement
10764 provided for each. These statements should be similar to the following model, as applicable:
10765
10766

10767 F10.1 Chapter(s) _____ of the SAR describe(s) the applicant's proposed program for
10768 preoperational testing and initial operations of the (cask designation).
10769 Chapter(s) _____ discuss the proposed maintenance program.
10770

10771 F10.2 Structures, systems, and components (SSCs) important to safety will be
10772 designed, fabricated, erected, tested, and maintained to quality standards
10773 commensurate with the importance to safety of the function they are intended to
10774 perform. Chapter ____ of the SAR identifies the safety importance of SSCs, and
10775 Chapter(s) _____ present(s) the applicable standards for their design,
10776 fabrication, and testing.
10777

10778 F10.3 The applicant/licensee will examine and/or test the (cask designation) to ensure
10779 that it does not exhibit any defects that could significantly reduce its confinement
10780 effectiveness. Chapter(s) _____ of the SAR describe(s) this inspection and
10781 testing.
10782

10783 F10.4 The applicant/licensee will mark the cask with a data plate indicating its model
10784 number, unique identification number, and empty weight. Drawing _____ in
10785 SAR Chapter ____ illustrates and/or describes this data plate.
10786

10787 The reviewer should provide a summary statement similar to the following:
10788

10789 "The staff concludes that the acceptance tests and maintenance program for the (cask
10790 designation) are in compliance with 10 CFR Part 72 and that the applicable acceptance
10791 criteria have been satisfied. The evaluation of the acceptance tests and maintenance
10792 program provides reasonable assurance that the cask will allow safe storage of
10793 throughout its licensed or certified term. This finding is reached on the basis of a review

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that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.”

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11 RADIATION PROTECTION EVALUATION

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11.1 Review Objective

10802 This chapter describes the radiation protection evaluation requirements and considerations of
10803 the proposed dry storage system (DSS). As used here, radiation protection refers to
10804 organizational, design, and operational elements that are primarily intended to limit radiation
10805 exposures from normal operations and anticipated occurrences. The evaluation of the
10806 radiological consequences for accidents is addressed in Chapter 12, "Accident Analyses
10807 Evaluation" of this SRP.

10809 The primary objectives of the radiation protection evaluation are to determine whether the
10810 design features and proposed operations meet the following criteria:

- 10812 • the proposed DSS radiation protection features meet the U.S. Nuclear
10813 Regulatory Commission (NRC) design criteria for direct radiation;
- 10814
- 10815 • the applicant has proposed engineering features and operating procedures for
10816 the DSS that will ensure occupational exposures will remain ALARA; and
- 10817
- 10818 • the radiation doses to the general public will meet regulatory standards during
10819 both normal conditions and anticipated occurrences.

10820

10821 In independent spent fuel storage installation (ISFSI) operations, the major mode of radiation
10822 exposure associated with spent nuclear fuel (SNF) storage cask handling is from direct
10823 radiation. Because of the cask design requirements, radionuclides are not expected to be
10824 released from the cask during either normal operations or design-basis accidents (DBAs).

10825

11.2 Areas of Review

10826

10827

10828 This chapter of the DSS Standard Review Plan (SRP) provides guidance for use in evaluating
10829 the radiation protection capabilities of the proposed cask system. The following outline shows
10830 the areas of review addressed in Section 11.4, "Acceptance Criteria," and Section 11.5, "Review
10831 Procedures," that may be encompassed in a comprehensive radiation protection review:

10832

10833 ***Radiation Protection Design Criteria and Features***

10834

10835 ***Occupational Exposures***

10836

10837 ***Exposures at or Beyond the Controlled Area Boundary***

10838 Normal Conditions

10839 Accident Conditions and Natural Phenomenon Events

10840

10841 ***ALARA***

10842 Design Considerations

10843 Engineering Controls and Procedures

10844

10845 **11.3 Regulatory Requirements**

10846

10847 This section presents a summary matrix of the portions of U.S. Code of Federal Regulations
 10848 (CFR) Parts 20 and 72 that are relevant to the review areas addressed by this chapter. The
 10849 NRC staff reviewer should read the exact referenced regulatory language. Virtually the entire
 10850 contents of 10 CFR 20 “Standards for Protection Against Radiation” are also applicable to this
 10851 review. Tables 11-1 and 11-2 match the relevant regulatory requirements associated with this
 10852 chapter to the areas of review identified in the previous section.
 10853

Table 11-1 Relationship of 10 CFR Part 20 Regulations and Areas of Review

Areas of Review	10 CFR Part 20 Regulations									
	20.1101	20.1201 (a)	20.1207	20.1208	20.1301 (a), (b), (d)	20.1302 (a)	20.1406	20.1501 (a)(1)	20.1701	20.1702
Radiation Protection Design Criteria and Features	•						•	•	•	•
Occupational Exposures	•	•	•	•				•		•
Exposures at or Beyond the Controlled Area Boundary	•				•	•		•		
ALARA	•						•	•		•

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Table 11-2 Relationship of 10 CFR Part 72 Regulations and Areas of Review

Areas of Review	10 CFR Part 72 Regulations			
	72.104(a)	72.104(b)	72.126(a)	72.236(d)
Radiation Protection Design Criteria and Features			•	•
Occupational Exposures				
Exposures at or Beyond the Controlled Area Boundary	•			•
ALARA		•	•	•

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10858 **11.4 Acceptance Criteria**

10859
10860 This section describes the acceptance criteria used for review of radiation protection features of
10861 and programs proposed for use with a DSS. These criteria are organized according to the
10862 areas of review specified in Section 11.2 of this chapter. The reviewer should note that some
10863 overlap exists between acceptance criteria for radiation protection and those related to Chapter
10864 5, "Confinement Evaluation," and Chapter 6, "Shielding Evaluation," of this SRP; therefore, the
10865 reviews of the chapters should be coordinated.

10866
10867 **11.4.1 Radiation Protection Design Criteria and Features**

10868
10869 Limitations on dose rates associated with direct radiation from the cask are established on the
10870 basis of the shielding and confinement evaluations to satisfy the regulatory requirements for
10871 dose limits to individuals located beyond the controlled area boundary (10 CFR 72.104).

10872
10873 **11.4.2 Occupational Exposures**

10874
10875 Estimated dose rates should be provided in Chapter 6, "Shielding Evaluation," of the Safety
10876 Analysis Report (SAR) for representative points within the restricted areas as well as at or
10877 beyond the perimeter of the controlled area. The radiation protection review includes a dose
10878 assessment that incorporates findings of the shielding review, as applicable. Individual and
10879 collective doses should be calculated.

10880
10881 All individual doses to workers should be well below the dose limits specified in
10882 10 CFR 20.1201. Collective doses should be consistent with the objectives contained in a well-
10883 structured ALARA program. The information provided by the applicant should allow for the
10884 determination of compliance with these criteria. To assess the applicant's occupational dose
10885 calculations, the reviewer should check such things as the number of workers specified for a
10886 task and the time specified for performing the task being reasonable.

10887
10888 **11.4.3 Exposures at or Beyond the Controlled Area Boundary**

10889 a. Normal Conditions:

10890
10891 For normal operations and anticipated occurrences, the estimated dose to any real
10892 individual located at or beyond the controlled area boundary may not exceed the
10893 following values specified in 10 CFR 72.104(a):
10894
10895

Whole body	0.25 mSv/yr (25 mrem/yr)
Thyroid	0.75 mSv/yr (75 mrem/yr)
Other organ	0.25 mSv/yr (25 mrem/yr)

10896
10897 For purposes of the DSS review, the calculated doses must include both direct radiation
10898 and any planned discharges of radioactive material.
10899

10900 b. Accident and Natural Phenomenon Events:

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10902
10903
10904
10905

Radiation shielding and confinement features should be provided sufficient to meet the requirements of 10 CFR 72.106(b). Any individual located on or beyond the nearest boundary of the controlled area may not receive the following dose from any DBA:

The more limiting of	
TEDE or Sum of the DDE and the CDE to any individual organ or tissue (other than the lens of the eye)	0.05 Sv (5 rem) 0.5 Sv (50 rem)
Lens of the eye	0.15 Sv (15 rem)
Shallow Dose Equivalent (SDE) to skin or any extremity	0.5 Sv (50 rem)

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11.4.4 ALARA

10910 For any new design or design change, the ALARA discussion should demonstrate how the design or design change

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10913

- accounted for radiation protection, technological, and economic considerations; and

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- to the extent practicable, employed engineering controls and procedures that were founded upon sound radiation protection principles.

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11.5 Review Procedures

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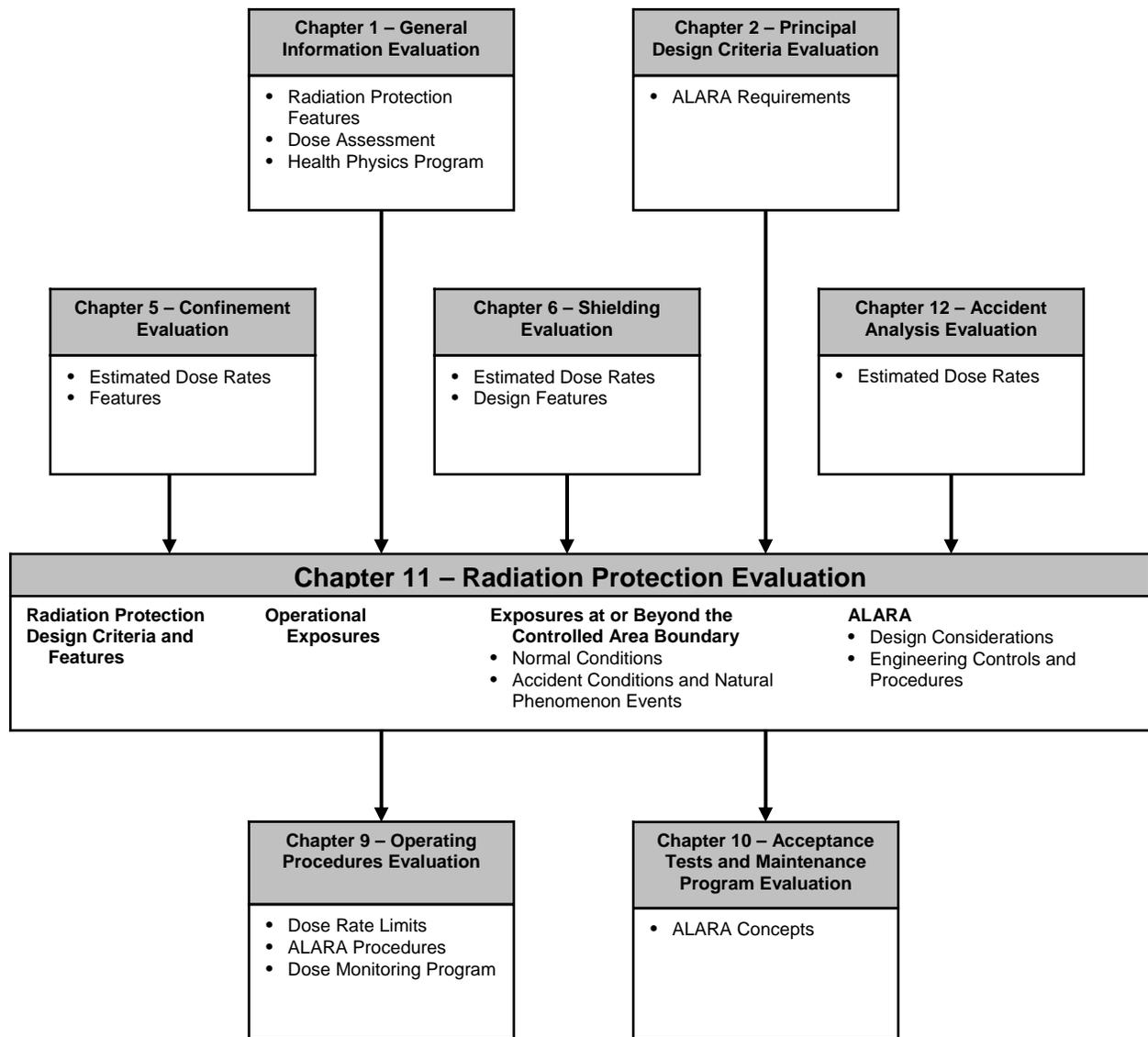
The interrelationship of the radiation protection review with other disciplines is shown in Figure 11-1.

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11.5.1 Radiation Protection Design Criteria and Features for the Transfer Cask and Storage Cask (MEDIUM Priority)

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The reviewer should read the general description and functional features of the cask presented in Chapter 1, "General Description," of the SAR. In addition, Chapter 2, "Principal Design Criteria," of the applicant's SAR should be reviewed as well as any additional detail regarding radiation protection provided in the Shielding and Confinement chapters of the SAR. If not previously discussed, the following additional criteria should be presented in Chapter 11, Radiation Protection, of the SAR.



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Figure 11-1 Overview of the Radiation Protection Evaluation

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- The cask system design should satisfy ALARA and other occupational exposure requirements of 10 CFR Part 20, and
- The sum of the doses from direct radiation and from release of radioactive materials to the atmosphere should satisfy the requirements of 10 CFR 72.104(a) and 72.106(b). Because of the stringent design requirements for SNF cask systems, the release of radionuclides into the atmosphere is expected to be insignificant under both normal and accident conditions. Direct radiation is the major mode of exposure.

11.5.2 Occupational Exposures (MEDIUM Priority)

The reviewer should analyze Chapter 9, "Operating Procedures," of the SAR and direct radiation dose calculations in Chapter 6, "Shielding Evaluation" of the SAR. These data should be used in Chapter 11, "Radiation Protection" of the SAR to estimate the doses received by occupational personnel, including minors, during cask loading and transfer to the ISFSI. Any significant differences from these doses that may occur during cask retrieval and unloading should be identified. In addition, the reviewer should verify that the applicant presents similar dose estimates for periodic or routine maintenance as well as surveillance activities. These estimates may require additional assumptions concerning adjacent casks for a typical storage configuration.

The reviewer should verify that the applicant presents the rationale used to justify the bases for various exposure times, personnel locations relative to the casks (including hot spots), number of personnel required, and appropriate gamma and neutron dose rates. In addition, the reviewer should verify that the calculated doses are consistent with these estimates. The actual operations will be performed under an active dose-monitoring program that further ensures compliance with the requirements of 10 CFR Part 20. Regulatory Guide (RG) 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," which was developed to implement revisions to 10 CFR Part 20, can be used to determine the acceptability of the applicant's occupational exposure evaluation and monitoring recommendations. Exposure to the embryo/fetus of a declared pregnant worker can be determined using the methodology in RG 8.36, "Radiation Dose to the Embryo/Fetus."

11.5.3 Exposures at or Beyond the Controlled Area Boundary (MEDIUM Priority)

As required by 10 CFR 72.236(d), the application must (1) demonstrate that the shielding and confinement features of the cask are sufficient to meet the requirements for real individuals in 10 CFR 72.104, and for DBA conditions in 10 CFR 72.106, and (2) facilitate future site-specific evaluations for each general ISFSI licensee. The real individual is an individual at or beyond the controlled area. Dose to any real individual must not exceed the limits specified in 10 CFR 72.104 from both the storage facility and other surrounding fuel cycle activities. For example, a real individual may be anyone living, working, or recreating close to the facility for a significant portion of the year.

However, for approval of a cask design, the reviewer should ensure that the applicant evaluates the shielding and confinement features of a single cask and a theoretical array of casks, assuming design-basis source terms and full-time occupancy. Supplemental shielding that may be used at an ISFSI to meet the exposure requirements to a real individual should also be appropriately evaluated. The reviewer should coordinate the review of supplemental shielding

10990 with the Chapter 13, "Technical Specifications and Operating Controls and Limits Evaluation," of
10991 this SRP review.

10992
10993 11.5.3.1 Normal Conditions

10994
10995 The single-cask analysis should identify the minimum distance that is required to meet the dose
10996 rates in 10 CFR 72.104. Past applications have shown this distance to be typically within 200m
10997 of the cask. A dose rate versus distance curve for a single cask should be included to facilitate
10998 site-specific evaluations for general ISFSI licensees. To satisfy section 10 CFR 72.106(b), dose
10999 evaluations should be determined at a minimum of 100m (328 ft) distance to the closest
11000 boundary of the controlled area. However, the applicant may use a longer distance provided
11001 that the longer distance is made a condition of use.

11002
11003 The reviewer should verify that the applicant includes a dose rate versus distance curve for a
11004 theoretical cask array. The theoretical cask array should consist of at least 20 storage casks
11005 (2x10 array) and may include the effect of shielding among casks in the array.

11006
11007 It is important to note that the general ISFSI licensee is permitted to use additional engineering
11008 features (supplemental shielding) such as berms to mitigate doses to real individuals near the
11009 site. If such features are used in the cask SAR to show compliance with the regulations, they
11010 should be included in the cask conditions of use. In addition, the SAR should determine the
11011 degree to which the normal condition dose rates could change for the identified off-normal
11012 conditions.

11013
11014 Since both distance and shielding can be used to limit doses, an applicant may choose to
11015 evaluate both options. If both longer distance to the controlled area boundary and supplemental
11016 shielding are evaluated in the SAR to demonstrate that either may be used to show compliance
11017 with the regulations, the condition of use (in the CoC or technical specifications) should be
11018 specified to allow the user to choose which to implement.

11019
11020 As required by 72.212(b)(2)(i)(C), a general licensee must perform a written evaluation to
11021 demonstrate that the requirements of 72.104 are met. An evaluation similar to that for a site-
11022 specific ISFSI should be performed. The licensee may use information provided in the cask
11023 SAR as well as site-specific information to perform the evaluation. Evaluations performed by
11024 the general ISFSI licensee are not submitted to NRC for approval; however, they are subject to
11025 NRC inspection and should be recorded and maintained by the general licensee.

11026
11027 The general licensee should establish measures in the radiological protection program,
11028 environmental monitoring program, and/or operating procedures to identify and re-evaluate
11029 potential increases in exposure to the real individuals. Compliance with the dose limits in
11030 10 CFR 72.104 will be verified by the environmental monitoring program with direct radiation
11031 measurements and/or effluent measurements, as appropriate.

11032
11033 11.5.3.2 Accident Conditions and Natural Phenomenon Events

11034
11035 The direct dose rate associated with accident conditions at the boundary of the controlled area
11036 should be reviewed as discussed in Chapter 6, "Shielding Evaluation," of this SRP. Also, the
11037 dose rate resulting from accidental release of radionuclides, as presented in Chapter 5,
11038 "Confinement Evaluation," of this SRP, should be reviewed. The accident-related radionuclide
11039 release dose should account for both air and liquid pathways as appropriate. In addition, the
11040 reviewer should verify that the applicant has evaluated the source terms for both SNF fission

11041 product and cask surface contamination. The sum of these should satisfy the requirements of
11042 10 CFR 72.106(b). For purposes of demonstrating compliance with 10 CFR 72.106(b) and
11043 evaluation against the Environmental Protection Agency Protective Action Guides in the *Manual*
11044 *of Protective Action Guides and Protective Actions for Nuclear Incidents* (EPA 410R-92-001),
11045 the skin, extremities, and the lens of the eye may be considered separately from other organs.
11046

11047 As noted in Chapter 6, "Shielding Evaluation," of this SRP, the time-integrated dose at the
11048 boundary of the controlled area may be small. Consequently, the reviewer should verify that the
11049 applicant estimates the doses at 100m (328 ft.) from the storage location to the nearest
11050 boundary of the controlled area unless the SAR specifies a greater minimum distance that is
11051 also made a condition of use for the proposed DSS. Alternatively, applicants may depict dose
11052 estimation using a curve showing dose versus distance from an assumed array of casks.
11053

11054 **11.5.4 ALARA (MEDIUM Priority)**

11055
11056 Further information on ALARA can be found in RG 8.8, "Information Relevant to Ensuring that
11057 Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As is Reasonably
11058 Achievable," and RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation
11059 Exposures As Low As is Reasonably Achievable."
11060

11061 11.5.4.1 Design Considerations

11062
11063 The cask design features should be reviewed to ensure that the features for which credit is
11064 taken in radiation protection analyses are clearly identified on the drawings. Also, the reviewer
11065 should ensure the application includes commitments to implement those features that have
11066 been credited in analyses to show compliance with regulatory requirements or ALARA goals.
11067 The reviewer should coordinate with the reviewers of SRP Chapters 5, "Confinement
11068 Evaluation" and 6, "Shielding Evaluation."
11069

11070 11.5.4.2 Procedures and Engineering Controls

11071
11072 The reviewer should determine that the descriptions of proposed DSS operations adequately
11073 demonstrate that ALARA principles have been incorporated into operational procedures and
11074 engineering controls. The reviewer should ensure that plans and procedures have been
11075 developed in accordance with applicable requirements and guidance.
11076

11077 **11.6 Evaluation Findings**

11078
11079 Evaluation findings are prepared by the reviewer upon determination that the regulatory
11080 requirements related to radiation protection as identified in Section 11.3 of this chapter have
11081 been satisfied. Some of these determinations can be made only after evaluating the results of
11082 reviews performed under other chapters of this SRP. If the documentation submitted with the
11083 application fully supports positive findings for each of the regulatory requirements, the
11084 statements of findings should be similar to the following:
11085

11086 F11.1 The [cask designation] provides radiation shielding and confinement features that
11087 are sufficient to meet the requirements of 10 CFR 72.104 and 72.106.
11088

11089 F11.2 The design and operating procedures of the [cask designation] provide
11090 acceptable means for controlling and limiting occupational radiation exposures

11091 within the limits given in 10 CFR 20 and for meeting the objective of maintaining
11092 exposures ALARA.

11093

11094 A summary statement similar to the following should be made:

11095

11096 “The staff concludes that the design of the radiation protection system of the [cask
11097 designation] is in compliance with 10 CFR Part 72 and that the applicable design and
11098 acceptance criteria have been satisfied. The evaluation of the radiation protection
11099 system design provides reasonable assurance that the [cask designation] will allow safe
11100 storage of SNF. This finding is reached on the basis of a review that considered the
11101 regulation itself, appropriate regulatory guides, applicable codes and standards, and
11102 accepted health physics practices.”

12 ACCIDENT ANALYSES EVALUATION

12.1 Review Objective

In this portion of the dry storage system (DSS) review, the U.S. Nuclear Regulatory Commission (NRC) evaluates the applicant's identification and analysis of hazards as well as the summary analysis of system responses to both off-normal and accident or design-basis events.

Accident events are considered to occur infrequently, if ever, during the lifetime of the facility. ANSI/ANS 57.9-92 subdivides this class of accidents into two categories – Design Events III and IV. Design Event III is a set of infrequent events that could be expected to occur during the lifetime of a DSS, and Design Event IV is a set of events that establishes a conservative design basis for structures, systems, and components (SSC) important to safety. The effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches are considered to be accident events in addition to manmade events.

This review ensures that the applicant has conducted thorough accident analyses as reflected by the following factors:

- Identified all credible accidents.
- Provided complete information in the safety analysis report (SAR).
- Analyzed the safety performance of the cask system in each review area.
- Fulfilled all applicable regulatory requirements.

12.2 Areas of Review

This portion of the DSS review evaluates the applicant's identification and analysis of hazards with particular emphasis on the safety performance of the cask system under off-normal events and conditions, and accident or design-basis events. Consequently, this chapter of the DSS Standard Review Plan (SRP) provides guidance for use in reviewing the applicant's identification and analysis of hazards as well as the summary analysis of system responses. A comprehensive accident analysis evaluation may encompass the following areas of review:

Cause of the Event

Detection of the Event

Summary of Event Consequences and Regulatory Compliance

Corrective Course of Action

12.3 Regulatory Requirements

This section presents a summary matrix of the portions of U.S. Code of Federal Regulations (CFR), Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, "Energy" (10 CFR Part 72) that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should read the exact referenced regulatory language. Table 12-1 matches the relevant regulatory requirements associated with this chapter to the areas of review identified in the previous section.

Table 12-1 Relationship of Regulations and Areas of Review

Areas of Review	10 CFR Part 72 Regulations				
	72.104 (a)	72.106 (b)	72.122(b)(1),(3), (d), (g), (h)(4), (i), (l)	72.124(a)	72.236(c), (d), (l)
Cause of the Event			•		
Detection of the Event			•	•	
Summary of Event Consequences and Regulatory Compliance	•	•	•	•	•
Corrective Course of Action			•		

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12.4 Acceptance Criteria

Accidents and natural phenomena events may share common regulatory and design limits. Consequently, the following sections sometimes refer to these scenarios collectively as accident conditions.

By contrast, off-normal conditions (anticipated occurrences) are distinguished, in part, from accidents or natural phenomena by the appropriate regulatory guidance and design criteria. For example, the radiation dose from an off-normal event must not exceed the limits specified in 10 CFR Part 20, “Standards for Protection Against Radiation,” and 10 CFR 72.104(a), whereas the radiation dose from an accident or natural phenomenon must not exceed the specifications of 10 CFR 72.106(b). Accident conditions may also have different allowable structural criteria.

In general, this portion of the DSS review seeks to ensure that the DSS design and the applicant’s hazard identification and analyses of related system responses fulfill the following acceptance criteria:

12.4.1 Dose Limits for Off-Normal Events

During normal operations and off-normal conditions, the requirements specified in 10 CFR Part 20 must be met. In addition, the annual dose equivalent to any individual located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other organ as a result of exposure to the following sources (10 CFR 72.104):

- Planned discharges to the general environment of radioactive materials (with the exception of radon and its decay products).
- Direct radiation from operations of the ISFSI.

- 11181
11182 • Any other cumulative radiation from uranium fuel cycle operations (i.e., nuclear
11183 power plant) in the affected area.
11184

11185 **12.4.2 Dose Limit for Design-Basis Accidents**
11186

11187 The dose from any credible design basis accident to any individual located on or beyond the
11188 nearest boundary of the controlled area may not exceed the limits specified in 10 CFR 72.106.
11189 Specifically, these are: the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem),
11190 or the sum of the deep dose equivalent to and the committed dose equivalent to any individual
11191 organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem); a lens dose equivalent of
11192 0.15 SV (15 rem); and a shallow dose equivalent to skin or any extremity of 0.5 Sv (50 rem).
11193

11194 **12.4.3 Criticality**
11195

11196 The spent nuclear fuel (SNF) must be maintained in a subcritical condition under credible
11197 conditions (i.e., k_{eff} , including all biases and uncertainties, equal to or less than 0.95). At least
11198 two unlikely, independent, and concurrent or sequential changes in the conditions essential to
11199 nuclear criticality safety should occur before a nuclear criticality accident is deemed to be
11200 possible (double contingency).
11201

11202 **12.4.4 Confinement**
11203

11204 The cask and its systems important to safety must be evaluated using appropriate tests or by
11205 other means acceptable to the NRC to demonstrate that they will reasonably maintain
11206 confinement of radioactive material under credible accident conditions.
11207

11208 **12.4.5 Recovery and Retrievability**
11209

11210 Recovery is the capability to return the stored radioactive material to a safe condition without
11211 endangering public health and safety. This generally means ensuring that any potential release
11212 of radioactive materials to the environment or radiation exposures is not in excess of the limits in
11213 10 CFR Part 20 or 10 CFR 72.122(h)(5).
11214

11215 Retrievability, on the other hand, is the capability to remove the SNF, high-level radioactive
11216 waste, or reactor-related GTCC waste from a storage system for further processing or disposal
11217 without endangering public health and safety. The DSS must be designed to allow ready
11218 retrieval of the stored SNF for compliance with 10 CFR 72.122(l), which applies to normal and
11219 off-normal design conditions and not to accident-level conditions.
11220

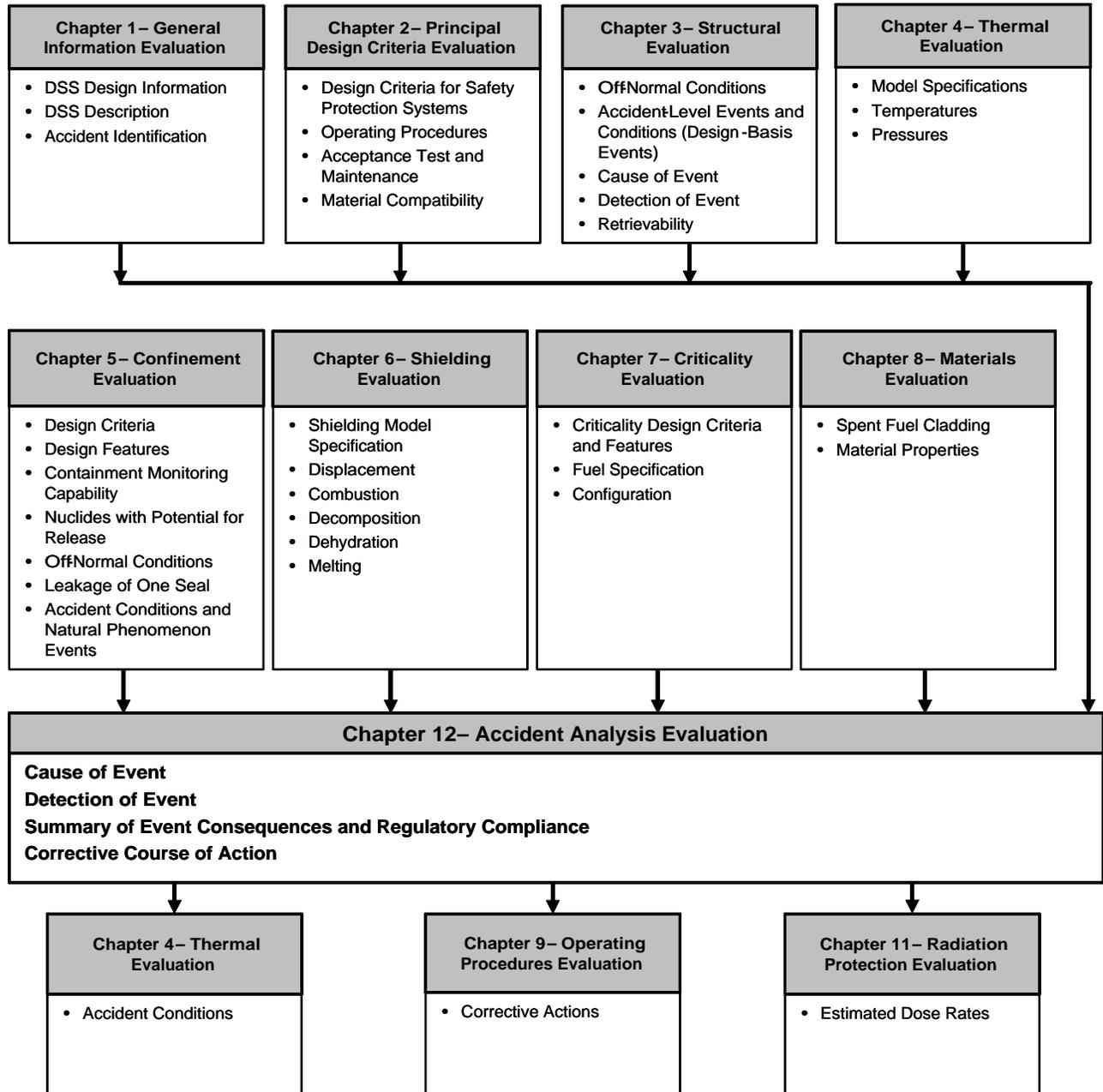
11221 **12.4.6 Instrumentation**
11222

11223 The SAR must identify all instruments and control systems that must remain operational under
11224 accident conditions.
11225

11226 **12.5 Review Procedures**

11227 Introduction

11228 Figure 12-1 presents an overview of the evaluation process and can be used as a guide to
 11229 assist in coordinating between the review disciplines.
 11230
 11231



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 11233
 11234

Figure 12-1 Overview of Accident Analysis Evaluation

11235
11236 The review procedures presented here describe general procedures for reviewing a DSS
11237 submittal. The review procedures in Chapter 15 of NUREG-1567, "Standard Review Plan for
11238 Spent Fuel Dry Storage Facilities," provide more detailed procedures and, where applicable,
11239 may be used as a guide to supplement the review procedures presented herein.

11240
11241 The off-normal conditions, accidents, and natural phenomena events identified in SAR
11242 Chapter 2, "Principle Design Criteria" should be reviewed by all disciplines, especially those
11243 accidents with potential consequences resulting in the failure of the confinement boundary. Off-
11244 normal conditions should be evaluated against the requirements of 10 CFR 72.104. Accidents
11245 and natural phenomena events should be evaluated against the requirements of 10 CFR 72.106
11246 and 72.122(b). Recovery methods or the need for overpacks or dry transfer systems to
11247 maintain safe storage conditions would then not be considered and evaluated as part of the
11248 NRC approval process. For each type of event, this discussion should include the applicant's
11249 evaluation of the following areas, as applicable.

11250
11251 **12.5.1 Cause of the Event (MEDIUM Priority)**
11252

11253 The cause of the accident should be described. The description should include the chain of
11254 events that leads to the credible accident condition and any bounding conditions.

11255
11256 **12.5.2 Detection of the Event (MEDIUM Priority)**
11257

11258 The licensee may detect an event through surveillance programs or monitoring instrumentation
11259 and alarms. Surveillance programs and monitoring instrumentation and alarms should have
11260 reasonable flexibility to allow for the identification of an accident condition or noncompliance
11261 situation that has not been previously considered in the SAR. The method of detection will be
11262 intuitively obvious for some events, whereas other events (e.g., fuel rod rupture) may remain
11263 undetected for a significant period of time.

11264
11265 DSS monitoring equipment (such as a pressure-monitoring system) are classified as important
11266 to safety in accordance with NUREG/CR-6407, "Classification of Transportation Packaging and
11267 Dry Spent Fuel Storage System Components According to Importance to Safety." Reviewers
11268 should refer to Chapter 5, "Confinement Evaluation," of this SRP.

11269
11270 **12.5.3 Summary of Event Consequences and Regulatory Compliance**
11271 **(MEDIUM PRIORITY)**
11272

11273 The applicant should address event consequences in each functional area corresponding to
11274 earlier chapters of the SAR (i.e., structural, thermal, shielding, criticality, confinement, materials,
11275 and radiation protection). This discussion should refer back to each SAR chapter in which the
11276 individual consequences are evaluated in detail. The applicant should provide a summary of
11277 the accident dose calculations and show that the consequences comply with the applicable
11278 regulatory criteria. For off-normal conditions, the applicant should demonstrate compliance with
11279 Part 20 as well as Part 72.

11280
11281 **12.5.4 Corrective Course of Action (MEDIUM Priority)**
11282

11283 The applicant should identify what action(s), if any, would be necessary to recover from the
11284 event. If various courses of action are possible, the applicant should present a discussion
11285 concerning the selection of the most appropriate action. Because the fuel must be readily

11286 retrievable, returning the cask to the fuel handling building and reloading the SNF into a new
11287 cask is a viable option. If corrective courses of action are to be included in operating
11288 procedures or administrative programs, then the applicable sections of SAR Chapter 9,
11289 "Operating Procedures," should be referenced.

11290
11291 **12.6 Evaluation Findings**

11292
11293 Review the 10 CFR Part 72 acceptance criteria and provide a summary statement for each.
11294 These statements should be similar to the following model:

- 11295
11296 F12.1 Structures, systems, and components of the [cask designation] are adequate to
11297 prevent accidents and to mitigate the consequences of accidents and natural
11298 phenomena events that do occur.
- 11299
11300 F12.2 The spacing of casks, discussed in Chapter _____ of the safety evaluation
11301 report (SER) and included as an operating limit in Chapter 13, "Technical
11302 Specifications and Operation Controls and Limits Evaluation" of the SAR will
11303 ensure accessibility of the equipment and services required for emergency
11304 response.
- 11305
11306 F12.3 Table _____ of the SER lists the Technical Specifications for the [cask system
11307 designation]. These Technical Specifications are further discussed in
11308 Chapter _____ of the SER.
- 11309
11310 F12.4 The applicant has evaluated the [cask designation] to demonstrate that it will
11311 reasonably maintain confinement of radioactive material under credible accident
11312 conditions.
- 11313
11314 F12.5 An accident or natural phenomena event will not preclude the ready retrieval of
11315 SNF for further processing or disposal.
- 11316
11317 F12.6 The SNF will be maintained in a subcritical condition under accident conditions.
- 11318
11319 F12.7 Neither off-normal nor accident conditions will result in a dose to an individual
11320 outside the controlled area that exceeds the limits of 10 CFR 72.104(a) or
11321 72.106(b), respectively.
- 11322
11323 F12.8 No instruments or control systems are required to remain operational under
11324 accident conditions [as applicable].

11325
11326 The reviewer should provide a summary statement similar to the following:

11327
11328 "The staff concludes that the accident design criteria for the [DSS designation] are in
11329 compliance with 10 CFR Part 72, and the accident design and acceptance criteria have
11330 been satisfied. The applicant's accident evaluation of the cask adequately demonstrates
11331 that it will provide for safe storage of SNF during credible accident situations. This
11332 finding is reached on the basis of a review that considered independent confirmatory
11333 calculations, the regulation itself, appropriate regulatory guides, applicable codes and
11334 standards, and accepted engineering practices."

11335

11336 **13 TECHNICAL SPECIFICATIONS AND OPERATING CONTROLS AND LIMITS**
11337 **EVALUATION**

11338
11339 **13.1 Review Objective**
11340

11341 The technical specifications and operating controls and limits review ensures that the operating
11342 controls and limits or the technical specifications, including their bases and justification, meet
11343 the requirements of the U.S. Code of Federal Regulations (CFR), Part 72, "Licensing
11344 Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive
11345 Waste and Reactor-Related Greater Than Class C Waste," Title 10, "Energy" (10 CFR Part 72).
11346 This evaluation is based on information that the applicant presents in Safety Analysis Report
11347 (SAR) Chapter 13, "Technical Specifications and Operation Controls and Limits Evaluation" as
11348 well as accepted practices and the applicant's commitments discussed in other chapters of the
11349 SAR or in correspondence subsequent to submission of the application. The NRC staff should
11350 also describe in the Safety Evaluation Report (SER) any additional operating controls and limits
11351 that the staff deems necessary and has added them, as appropriate, to the cask system's
11352 Technical Specifications.

11353
11354 For simplicity in defining the acceptance criteria and review procedures, the term "technical
11355 specifications" may be considered synonymous with "operating controls and limits." The
11356 technical specifications define the conditions that are deemed necessary for safe dry storage
11357 system (DSS) use. Specifically, they define operating limits and controls, monitoring
11358 instruments and control settings, surveillance requirements, design features, and administrative
11359 controls that ensure safe operation of the DSS. As such, these technical specifications are
11360 included in a DSS Certificate of Compliance (CoC). Each specification should be clearly
11361 documented and justified in the technical review sections of the SAR and the associated SER
11362 as necessary for safe DSS operation.

11363
11364 ONLY THE TERMS AND CONDITIONS OF THE COC, INCLUDING THE ATTACHED
11365 TECHNICAL SPECIFICATIONS AND DRAWINGS, ARE LEGALLY ENFORCEABLE. IF A
11366 REVIEWER DEEMS AN ITEM SO IMPORTANT THAT IT SHOULD NOT BE CHANGED
11367 WITHOUT NRC STAFF APPROVAL, THE ITEM SHOULD EITHER BE INCLUDED DIRECTLY
11368 IN THE COC TERMS, CONDITIONS OR TECHNICAL SPECIFICATIONS.
11369

11370 **13.2 Areas of Review**
11371

11372 This chapter of the DSS Standard Review Plan (SRP) provides guidance for use in evaluating
11373 the technical specifications that the applicant deems necessary for safe use of the proposed
11374 DSS system. As defined in Section 13.5, "Review Procedures," a comprehensive review of the
11375 proposed technical specifications would assess the applicant's compliance with the regulations
11376 to provide a level of control commensurate with that specified by 10 CFR 72.234 and 72.236.
11377 These requirements represent the following areas of review:

11378 ***Functional/Operating Limits, Monitoring Instruments, and Limiting Control***
11379 ***Settings***

11380
11381 ***Limiting Conditions***

11382
11383 ***Surveillance Requirements***

11384
11385 ***Design Features***
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Administrative Controls

13.3 Regulatory Requirements

This section presents a summary matrix of the portions of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. The U.S. Nuclear Regulatory Commission (NRC) staff reviewer should read the exact referenced regulatory language. Table 13-1 matches the relevant regulatory requirements associated with this chapter to the areas of review identified in the previous section.

Table 13-1 Relationship of Regulations and Areas of Review										
Areas of Review	10 CFR Part 72 Requirements									
	72.234 (a)	72.236								
		(a)	(b)	(c)	(d)	(e), (f), (h)	(g)	(i)	(j)	(l)
Functional/Operating Limits, Monitoring Instruments, and Limiting Control Settings	•	•		•	•					•
Limiting Conditions	•	•		•	•					•
Surveillance Requirements	•				•		•		•	
Design Features	•		•		•	•	•	•		•
Administrative Controls	•	•			•			•		•

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13.4 Acceptance Criteria

The reviewer should verify that the applicant identifies proposed technical specifications necessary to maintain subcriticality, confinement, shielding, heat removal, and structural integrity under normal, off-normal, and accident-level conditions. In addition, the reviewer should ensure that the applicant identifies the basis for each of the proposed technical specifications by reference to the analysis in the SAR. The NRC staff can use NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance," as appropriate guidance in the review of the technical specifications.

13.4.1 Functional/Operating Limits, Monitoring Instruments, and Limiting Control Settings

Acceptance criteria for functional and operating limits, monitoring instruments, and limiting control settings include limits placed on fuel, waste handling, and storage conditions to protect the integrity of the fuel and container, to protect the employees against occupational exposures, and to guard against the uncontrolled release of radioactive materials.

11417 **13.4.2 Limiting Conditions**

11418
11419 Acceptance criteria for functional and operating limits, monitoring instruments, and limiting
11420 control settings include limits placed on fuel, waste handling, and storage conditions to protect
11421 the integrity of the fuel and container, to protect the employees against occupational exposures,
11422 and to guard against the uncontrolled release of radioactive materials. Acceptance criteria for
11423 limiting conditions are the lowest levels required for safe operation.

11424
11425 **13.4.3 Surveillance Requirements**

11426
11427 Acceptance criteria for establishing surveillance requirements include the frequency and scope
11428 of surveillance requirements to verify performance and availability of structures, systems, and
11429 components (SSCs) important to safety, and the verification of the bases for the proposed
11430 limiting conditions.

11431
11432 **13.4.4 Design Features**

11433
11434 Acceptance criteria for design features include commitments to specified codes. The condition
11435 or technical specification should also describe a process to address deviations from the
11436 applicable codes that may be necessary. In such cases, the licensee should request an
11437 alternative to the requirements of the applicable code from the NRC. If the staff finds that the
11438 deviation does not adversely impact safety, it may authorize the requested alternative in writing.

11439
11440 Currently, there is an existing code for the design and construction of metallic nuclear fuel
11441 storage casks and the document is identified as Subsection WC of Division 3 of Section III of
11442 the ASME Boiler and Pressure Vessel Code. This was first issued as the 2005 addenda to the
11443 2004 Code. The current Code edition is 2007. As of February 2008, NRC staff had not taken a
11444 position regarding the acceptability of this document. In the past, Division 1 of the ASME B&PV
11445 Code had been used by NRC staff allowing alternatives to some provisions of that document
11446 which were judged to not be applicable to spent nuclear fuel storage casks. Early SNF dry
11447 storage licenses and certificates of compliance were issued without documenting which specific
11448 alternatives to ASME B&PV Code, Section III, were approved. Poor quality assurance practices
11449 during design and fabrication sometimes led to significant deviations from the Code without
11450 appropriate certificate holder design review or NRC review and approval. Therefore, the
11451 applicant should document commitments to ASME B&PV Code, Section III, with proposed
11452 alternatives in the application.

11453
11454 Likewise the NRC should document these commitments in the 10 CFR Part 72 licenses,
11455 certificates of compliance, or technical specifications and its approval of the proposed
11456 alternatives in the SER. Also, the NRC should include a statement (in the CoC or technical
11457 specifications) that refers the reader to the SAR and applicable SERs for any alternatives to the
11458 codes. In addition, to ensure that similar problems do not exist in other areas, all other codes
11459 and standards applied to components important to safety should be identified in the SAR and
11460 should be included in the CoC or technical specification. Figure 13-1 presents an example of a
11461 provision for allowing alternatives to applicable codes.

Codes and Standards

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1992 Edition with Addenda through 1994 is the governing Code for the storage system.

Design Alternatives to Codes, Standards, and Criteria

Table #-# lists all approved alternatives for the design of the DSS.

Construction/Fabrication Alternatives to Codes, Standards, and Criteria

Proposed alternatives to ASME B&PV Code Section III, 1992 Edition with Addenda through 1994, including alternatives referenced in Section 4.3.1, may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee.

The proposal to the NRC must demonstrate that the alternatives would provide an acceptable level of quality and safety, or that compliance with the specified requirements of ASME B&PV Code, Section III, 1992 Edition with Addenda through 1994 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Figure 13-1 Provision Example

In addition, acceptance criteria for design features include specifications important to criticality safety. Where criticality analyses rely upon the condition that the assemblies' active fuel length remains within the cask region containing the solid neutron absorbers, the applicant should commit to ensuring the cask features fulfill this analysis assumption. One common method is the installation of fuel spacers, upper and/or lower spacers as needed, to maintain the assemblies' position under all credible conditions. The minimum Boron-10 content of the solid neutron absorbers is another important design feature specification together with the qualification and acceptance testing method for ensuring the neutron absorbers meet the required minimum Boron-10 content throughout the absorber material. The proximity of fuel assemblies to each other also affects the cask's reactivity, generally with reactivity increasing as the assemblies are brought closer together; therefore, a minimum dimension(s) between adjacent assembly locations is specified. This dimension may be a minimum flux trap width or a minimum fuel cell pitch. These design parameters and commitments should also be included in the license, certificate of compliance, or technical specifications.

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13.4.5 Administrative Control

Acceptance criteria for administrative controls include organizational and management procedures, recordkeeping, review and audit systems, and reporting necessary to ensure that the DSS is managed in a safe and reliable manner. Administrative action that must be taken in the event of noncompliance with a limit or condition should be specified.

13.5 Review Procedures (HIGH Priority)

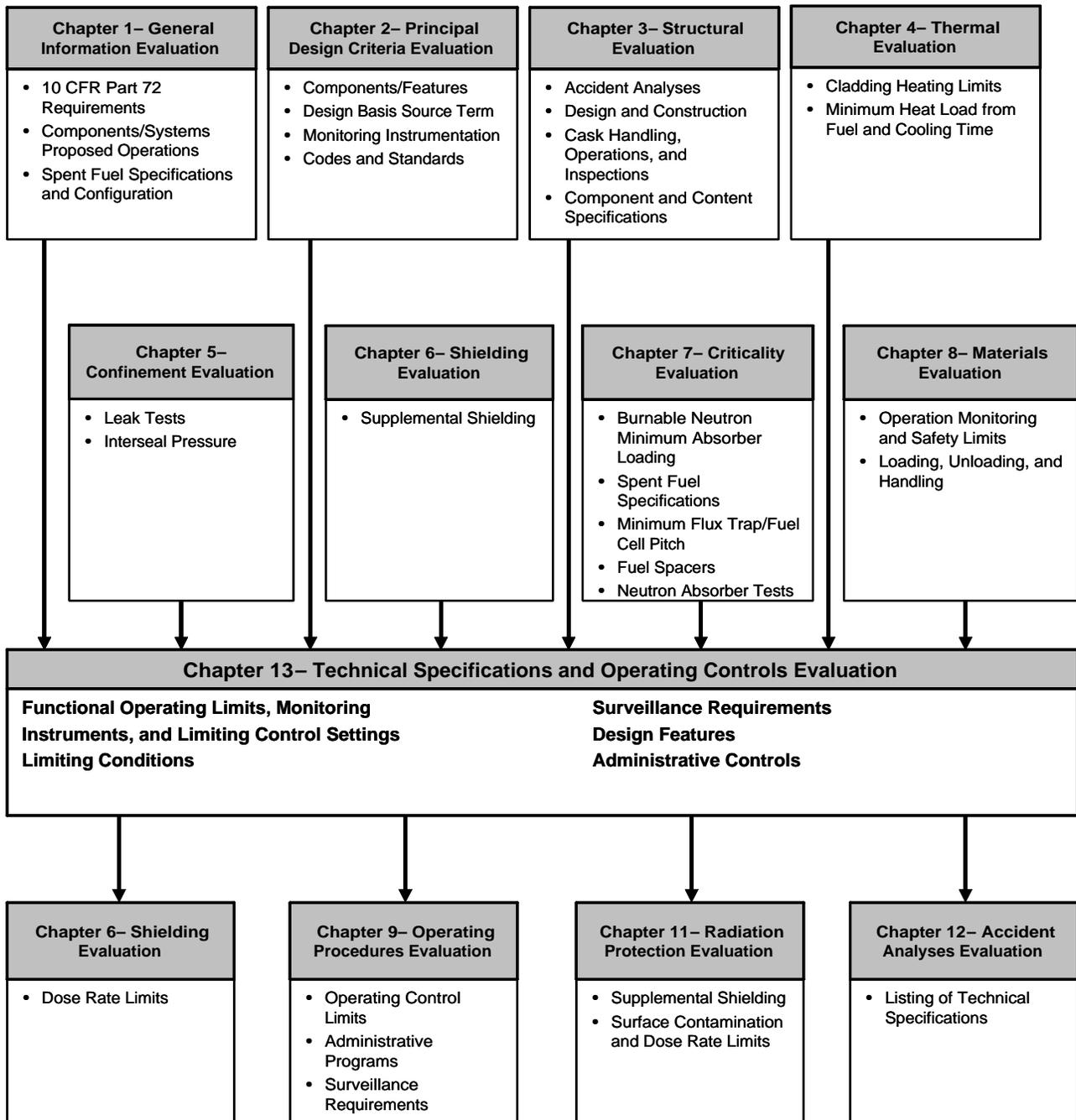
Figure 13-2 presents an overview of the evaluation process and can be used as a guide to assist in coordinating between review disciplines.

Reviewers should evaluate each chapter of the SAR with the goal of establishing the technical specifications. The variability of designs and operations makes it impossible to define each instance for which a technical specification is necessary. For this reason, it is important that the NRC staff conduct a coordinated, detailed, and thorough evaluation of each technical section of the SAR. Reviewers should note all instances in which the SAR either makes an assumption or imposes a condition that should be identified as a technical specification. Reviewers should also note any instances in which the SAR requests alternatives or exemptions from regulatory requirements, or other conditions that the reviewer identifies as an operational limit or condition. Such limits and exemptions should be clearly identified and documented in SAR Chapter 13. "Technical Specifications and Operation Controls and Limits Evaluation".

The various technical disciplines should review the results of their specific evaluations and compare their list of technical specifications to those identified by the applicant. The NRC staff should ensure that the conditions for use, as evaluated and approved by the technical reviewers, complement one another and are not contradictory. In addition, the staff will coordinate the resolution of any disputed condition, limit, or specification. The staff 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.

All reviewers should be familiar with the technical specifications of similar cask designs previously approved by the NRC staff. For example, the staff has previously approved cask designs and issued technical specifications regarding a variety of items including, but not limited to, the following examples:

- General requirements and conditions regarding site-specific parameters, operating procedures, quality assurance, heavy loads, training, etc.
- A preoperational training exercise and demonstration of most cask operations including loading, sealing, and drying (using mockups as appropriate); placement in storage; and return of fuel to the SNF pool.



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Figure 13-2 Overview of Technical Specifications and Operating Controls Evaluation

- 11529 • Specifications for the SNF to be stored in the cask, including, but not limited to,
11530 the type of SNF (i.e., boiling water reactor [BWR], pressurized water reactor
11531 [PWR], or both), the minimum and maximum allowable enrichments of the fuel
11532 before irradiation, burnup (i.e., megawatt-days/MTU), the minimum acceptable
11533 cooling time of the SNF before storage in the cask, the maximum heat designed
11534 to be dissipated, the maximum SNF loading limit, the maximum neutron and
11535 gamma source terms, condition of the SNF (i.e., intact assembly or consolidated
11536 fuel rods, allowable cladding condition), associated non-fuel hardware, and
11537 physical parameters (e.g., length, width, depth, weight, etc.). The reviewer
11538 should be aware that additional SNF specifications regarding operational history
11539 parameters (e.g., average moderator temperature, average in-core soluble boron
11540 concentrations, and operations under control rod banks or with control rod
11541 insertion) will need to be included in the technical specifications for cask systems
11542 relying on burnup credit.
11543
- 11544 • Criticality controls such as cask water boron concentrations, minimum flux
11545 trap/fuel cell pitch, use of fuel spacers, minimum neutron absorber loading, and
11546 neutron absorber tests.
11547
- 11548 • The inerting atmosphere requirements such as vacuum drying and helium backfill
11549 parameters.
11550
- 11551 • Cask handling restrictions such as lift height limits and ambient temperature
11552 (high/low) conditions.
11553
- 11554 • Confinement barrier requirements such as leak rate limits.
11555
- 11556 • Thermal performance parameters such as maximum temperatures or delta-
11557 temperatures.
11558
- 11559 • Radiological controls such as radiation dose rates and contamination limits.
11560
- 11561 • Cask array and/or spacing limits for thermal performance and radiological
11562 considerations.
11563

11564 All disciplines should coordinate their review of the proposed technical specifications to assure
11565 the operational limitations are measurable and inspectible. Other topics may include:

- 11566 • Frequency and scope proposed for the surveillance requirements.
11567
- 11568 • Administrative controls that include organization and administrative systems and
11569 procedures, record-keeping, review, and audit systems required to ensure that
11570 the DSS is managed in a safe and reliable manner.
11571
- 11572 • Administrative action that must be taken in the event of noncompliance with a
11573 limit or condition.
11574

11575 The reviewer should verify that the applicant includes a written description in a condition to the
11576 CoC or technical specification that documents the codes to which the applicant has committed.
11577 In addition, the condition or technical specification should describe a process to address any
11578

11579 deviations from the ASME B&PV Code or other codes that may be needed. Likewise, the
11580 reviewer should verify that these commitments are documented in the 10 CFR Part 72 CoC or
11581 technical specifications. A list of proposed alternatives to code requirements should also be
11582 provided in the SAR. This list should be revised as necessary to reflect all NRC-authorized
11583 alternatives.

11584
11585 NUREG-1745 provides a recommended format for use by applicants in presenting technical
11586 specifications. However, this format may not be applicable to all controls. Since the basis for
11587 the control may be extensively discussed in earlier chapters of the SAR, the applicant may use
11588 an abbreviated format in SAR Chapter 12.

11589
11590 Reviewers should ensure that all necessary technical specifications are explicitly delineated in
11591 SER Chapter 13, "Technical Specifications and Operating Controls and Limits Evaluation," and
11592 in the CoC. These delineations typically restate the technical specifications defined in the SAR
11593 but may be modified or supplemented as the staff deems appropriate. Reviewers should also
11594 ensure that limits and exemptions requested by the applicant are clearly identified and
11595 documented in the SER. The staff may prepare a separate table or appendix for SER
11596 Chapter 13 to explicitly designate the technical specifications that are applicable to the cask.
11597 Applicable drawings from the SAR should be identified by number and revision.

11598 11599 **13.6 Evaluation Findings**

11600
11601 NRC staff reviewers prepare evaluation findings regarding satisfaction of the regulatory
11602 requirements related to technical specifications. Evaluation findings developed or included in all
11603 SER sections relating to technical specifications are also listed in this section. These
11604 statements should be similar to the following model:

11605
11606 F13.1 The staff concludes that the conditions for use for [DSS name] identify necessary
11607 technical specifications to satisfy 10 CFR Part 72 and that the applicable
11608 acceptance criteria have been satisfied. The proposed technical specifications
11609 provide reasonable assurance that the DSS will allow safe storage of SNF. This
11610 finding is based on the regulation itself, appropriate regulatory guides, applicable
11611 codes and standards, and accepted practices. The technical specifications
11612 identified by the applicant include the following: [Reviewer to specify].

11613
11614 The reviewer should provide a summary statement similar to the following:

11615
11616 "The proposed technical specifications provide reasonable assurance that the cask will
11617 allow safe storage of spent fuel. This finding is reached on the basis of a review that
11618 considered the regulation itself, appropriate regulatory guides, applicable codes and
11619 standards, and accepted practices."

11620
11621 **14 QUALITY ASSURANCE EVALUATION**
11622

11623 **14.1 Review Objective**
11624

11625 The objective of the review is to determine whether the applicant for a dry storage system (DSS)
11626 certificate has submitted a quality assurance (QA) program description (QAPD) that
11627 demonstrates that the applicant's QA program complies with the requirements of 10 CFR Part
11628 72, Subpart G (Part 72), "Quality Assurance."
11629

11630 The basis for that determination is developed from an evaluation of the applicant's high level
11631 QAPD against the criteria provided in Section 14.4, Review Procedures below, Part 72, and any
11632 associated information found in the Federal Register since the last rulemaking has been
11633 completed, as applicable. (Note: The scope of review does not include actual procedures and
11634 instructions that implement the QA program, but may be described in the QAPD).
11635

11636 Determination of compliance for the applicant's QA program occurs during NRC inspection
11637 activities where implementation of the QA plan is evaluated. (Note: The scope of an inspection
11638 does include actual procedures and instructions that implement the QA program).
11639

11640 **14.2 Areas of Review**
11641

11642 This SRP provides guidance for use by a reviewer to perform an evaluation of a QAPD in terms
11643 of the 18 criteria defined in 10 CFR Part 72, Subpart G and Section 14.4, "Review Procedures"
11644 below, and the Federal Register, as applicable.
11645

11646 **14.3 Regulatory Requirements**
11647

11648 This section identifies the reviewer's need to review the exact regulatory language found in
11649 Part 72 relevant to quality assurance as applied to a DSS. Refer to Subpart G -Quality
11650 Assurance of 10 CFR Part 72.
11651

11652 **14.4 Acceptance Criteria**
11653

11654 The acceptance criteria below reflect the 18 quality criteria of Part 72, Subpart G. These criteria
11655 are presented in the form of descriptions of information to be included in the applicant's QAPD.
11656 For each criterion shown in Sections 14.5.1 through 14.5.18 of this SRP, examples of measures
11657 have been provided which may assist the reviewer in determining if the QAPD indicates that it
11658 meets the applicable criterion. For each of the activities and items identified as important to
11659 safety, the applicant should identify the applicable QA programmatic elements and include, as
11660 applicable, provisions for meeting each of the following quality criteria itemized in Section 14.5.
11661

11662 **14.5 Review Procedures (All items in this section are HIGH Priority)**
11663

11664 The purpose of the review is to obtain reasonable assurance that the applicant has developed
11665 and described a QA program for design, fabrication, construction, testing, operations,
11666 modification, and decommissioning activities associated with important-to-safety DSS systems,
11667 structures and components (SSCs).
11668

11669 It is important that the applicant's QAPD and associated portions of the safety analysis report
11670 (SAR) provide sufficient detail to enable the reviewer to assess that the applicant has committed

11671 to comply with the program and the QA program complies with the applicable requirements of
11672 10 CFR Part 72, Subpart G. If the reviewer determines that sufficient detail does not exist in the
11673 QAPD, the reviewer should refer to Section 14.6, Evaluation Findings for further direction. If the
11674 QAPD indicates commitment to follow certain standards, codes, etc., then the reviewer should
11675 consider the commitments as an integral part of the QA program.
11676

11677 The reviewer should recognize that application for QA program approval may either be separate
11678 from the SAR or may exist as a section in the applicant's SAR. Since it is possible that some
11679 aspects of the QA program are described in various portions of the application (the SAR or a
11680 submittal separate from the SAR) the reviewer should consider these aspects when evaluating
11681 the program against the acceptance criteria of Section 14.4. Therefore, if possible, the QAPD
11682 evaluation should be coordinated with other aspects of the DSS review. Such coordination will
11683 allow reviewers to derive a more accurate and complete assessment of the applicant's level of
11684 commitment to the overall QA program, the selection of quality criteria and quality levels, and
11685 the proposed implementation methods.
11686

11687 The applicant's QA program may be structured to apply QA measures and controls to all
11688 activities and items in proportion to their importance to safety, commonly referred to as a graded
11689 approach. A graded approach for the application of QA should be described in the QAPD by
11690 adequately assigning appropriate grading classifications and providing an associated
11691 justification. However, an applicant may choose to apply the highest level of QA and control to
11692 all activities and items. The QA program should identify the activities and items that are
11693 important to safety and the degree of their importance. For application of a graded approach,
11694 the highly important-to-safety activities and items must have a high level of control, while those
11695 less important may have a lower level of control. If the QA program is graded, the staff should
11696 be able to conclude that the structure of the graded program is acceptable and that the highest
11697 levels of QA are applied to those SSCs that are most important to safety. In making
11698 determinations about the application of QA to those SSCs that are listed in the description as
11699 important to safety, the reviewer of the QA program description should coordinate with the
11700 appropriate NRC project manager and associated technical staff to compare those SSCs
11701 described in other portions of the applicant's submittal.
11702

11703 If after review, the reviewer finds the QAPD acceptable, the acceptance of the evaluation should
11704 be documented in the Safety Evaluation Report (SER) for QAPDs submitted as part of a SAR.
11705 If the applicant's QAPD was submitted prior to the applicant's SAR submittal, the acceptance of
11706 the evaluation should be documented in a letter to the applicant and if possible included in the
11707 SER at a later time. In either case, the documentation of the review should include the basis for
11708 acceptance as noted in the example in Section 14.6 Evaluation Findings. Any
11709 recommendations for modifications in the application that are required before the application
11710 can be accepted should be addressed by referring to Section 14.6 for initiation of a request for
11711 additional information (RAI).
11712

11713 Figure 14-1 presents an overview of the evaluation process and can be used as a guide to
11714 assist in coordinating with other review disciplines.
11715

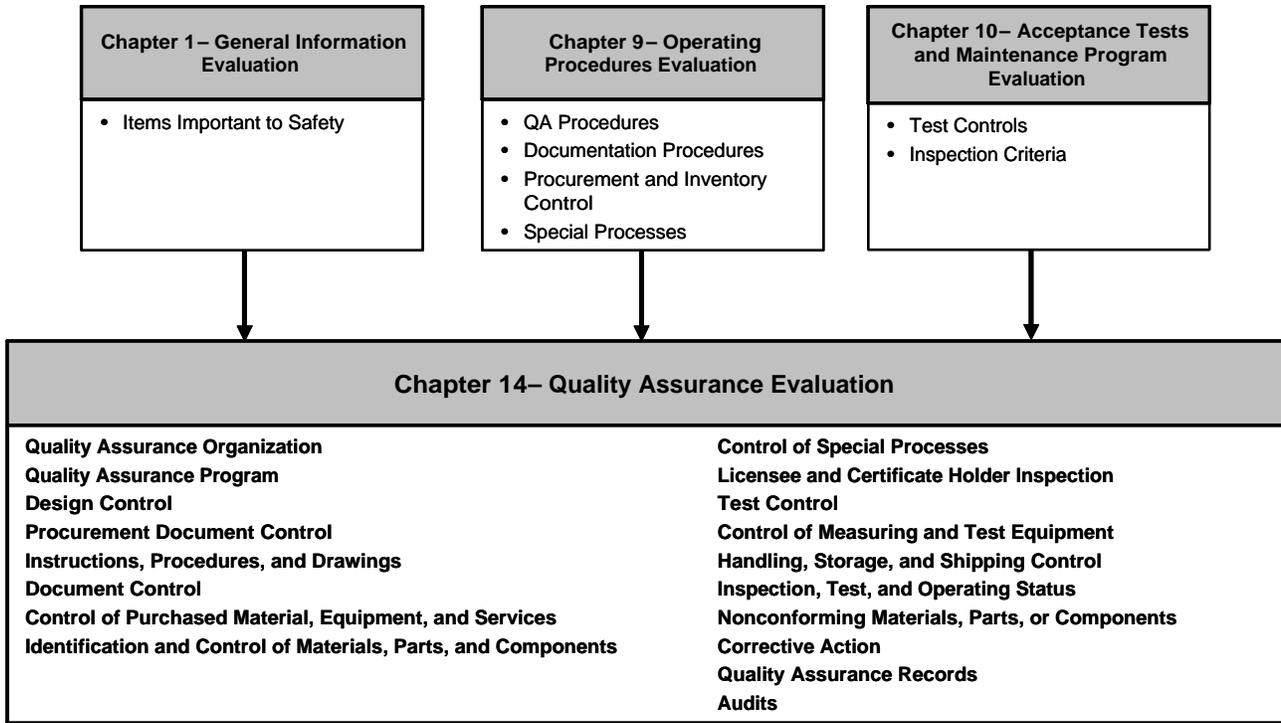


Figure 14-1 Quality Assurance Evaluation

14.5.1 Quality Assurance Organization

The QAPD should describe the structure, interrelationships, and areas of functional responsibility and authority for all organizational elements that will perform activities related to quality and safety. The following are examples of areas/items that may be addressed to support implementation of the quality criteria:

- a. Measures to retain and exercise responsibility for the QA program. The assignment of responsibility for the overall QA program in no degree relieves line management of their responsibility for the achievement of quality.
- b. Measures to identify and describe the QA functions performed by the applicant's QA organization or delegated to other organizations that will provide controls to ensure implementation of the applicable elements of the QA criteria.
- c. Measures to provide clear management controls and effective lines of communication should exist between the applicant's QA organizations and suppliers to ensure proper direction of the QA program and resolution of QA problems.
- d. Measures to identify onsite and offsite organizational elements that will function under the purview of the QA program and the lines of responsibility.
- e. Measures to ensure that high-level management is responsible for documenting and promulgating the applicant's QA policies, goals, and objectives, and this management level should maintain a continuing involvement in QA matters. The application should

- 11744 also describe the lines of communication between intermediate levels of management
 11745 and between this position and the Manager (or Director) of QA.
 11746
 11747 f. Measures to designate a position that retains overall authority and responsibility for the
 11748 QA program.
 11749
 11750 g. Measures to provide authority and independence of the individual responsible for
 11751 managing the QA program should be such that he or she can direct and control the
 11752 organization's QA program, effectively ensure conformance to quality requirements, and
 11753 remain sufficiently independent of undue influences and responsibilities of schedules
 11754 and costs. In addition, measures to have this individual report to at least the same
 11755 organizational level as the highest line manager directly responsible for performing
 11756 activities affecting quality.
 11757
 11758 h. Measures for individuals or groups responsible for defining and controlling the content of
 11759 the QA program and related manuals to have appropriate organizational position and
 11760 authority, as should the management level responsible for final review and approval.
 11761
 11762 i. Measures describing the qualification requirements for the principal QA management
 11763 positions so as to demonstrate management and technical competence commensurate
 11764 with the responsibilities of these positions.
 11765
 11766 j. Measures to ensure conformance to established requirements be verified by individuals
 11767 or groups who do not have direct responsibility for performing the work being verified.
 11768 The quality control function may be part of the line organization provided that the QA
 11769 organization performs periodic surveillance to confirm sufficient independence from the
 11770 individuals who performed the activities.
 11771
 11772 k. Persons and organizations performing QA functions should have direct access to
 11773 management levels that will ensure accomplishment of quality-affecting activities. These
 11774 individuals should have sufficient authority and organizational freedom to perform their
 11775 QA functions effectively and without reservation. In addition, they should be able to
 11776 identify quality problems; initiate, recommend, or provide solutions through designated
 11777 channels; and verify implementation of solutions.
 11778
 11779 l. Designated QA individuals or organizations should have the responsibility and authority,
 11780 delineated in writing, to stop unsatisfactory work and control further processing, delivery,
 11781 or installation of nonconforming material. In addition, the application should describe
 11782 how stop-work requests will be initiated and completed.
 11783
 11784 m. Measures to determine the extent of QA controls to be determined by the QA staff in
 11785 combination with the line staff and to depend upon the specific activity or item complexity
 11786 and level of importance to safety.
 11787

11788 **14.5.2 Quality Assurance Program**
 11789

11790 The QAPD should provide acceptable evidence that the applicant's proposed QA program will
 11791 be well-documented, planned, implemented, and maintained to provide the appropriate level of
 11792 control over activities and SSCs consistent with their relative importance to safety. The
 11793 following are examples of areas/items that may be addressed to support implementation of the
 11794 quality criteria:

- 11795
11796 a. Measures used to ensure that the QA program meets applicable acceptance criteria.
11797
11798 b. Measures for management to regularly assess the effectiveness of the QA program. In
11799 addition, measures for management (above and beyond the QA organization) to
11800 regularly assess the scope, status, adequacy, and compliance of the QA program to the
11801 requirements of 10 CFR Part 72. Measures to provide for management's frequent
11802 contact with program status through reports, meetings, and audits as well as
11803 performance of a periodic assessment that is planned and documented with corrective
11804 action identified and tracked.
11805
11806 c. Measures used to ensure that trained, qualified personnel within the organization will be
11807 assigned to determine that functions delegated to contractors are properly
11808 accomplished.
11809
11810 d. Summarizations of the corporate QA policies, goals, and objectives and establishment of
11811 a meaningful channel for transmittal of these policies, goals, and objectives down
11812 through the levels of management.
11813
11814 e. Measures to designate responsibilities for implementing the major activities addressed in
11815 the QA manuals.
11816
11817 f. Measures to control the distribution of the QA manuals and revisions.
11818
11819 g. Measures for communicating to all responsible organizations and individuals that
11820 policies, QA manuals, and procedures are mandatory requirements.
11821
11822 h. Measures to provide a comprehensive listing of QA procedures, plus a matrix of these
11823 procedures cross-referenced to each of the QA criteria, to demonstrate that the QA
11824 program will be fully implemented by documented procedures.
11825
11826 i. Identification of the structures, systems, and components (SSCs) that are important to
11827 safety and how they will be controlled by the QA program.
11828
11829 j. Measures for review and documents to show agreement with the QA program provisions
11830 of its suppliers to ensure implementation of a program meeting the QA criteria.
11831
11832 k. Measures for the resolution of disputes involving quality arising from a difference of
11833 opinion between QA/Quality Control (QC) personnel and personnel from other
11834 departments (engineering, procurement, manufacturing, etc.).
11835
11836 l. Measures for indoctrination, training, and qualification programs that fulfill the following
11837 criteria:
11838
11839 • Personnel responsible for performing activities affecting quality should be
11840 instructed as to the purpose, scope, and implementation of the quality-related
11841 manuals, instructions, and procedures.
11842
11843 • Personnel performing activities affecting quality should be trained and qualified in
11844 the principles and techniques of the activities being performed.
11845

- 11846 • Maintenance of the proficiency of personnel performing quality-affecting activities
11847 by retraining, reexamining, and re-certifying.
- 11848
- 11849 • Preparation and maintenance of documentation of completed training and
11850 qualification.
- 11851
- 11852 • Qualification of personnel in accordance with accepted codes and standards.
- 11853

11854 **14.5.3 Design Control**

11855
11856 The QAPD should describe the approach that the applicant will use to define, control, and verify
11857 the design and development of the DSS. The following are examples of areas/items that may
11858 be addressed to support implementation of the quality criteria:

- 11859
- 11860 a. Measures to carry out design activities in a planned, controlled, and orderly manner.
- 11861
- 11862 b. Measures to correctly translate the applicable regulatory requirements and design bases
11863 into specifications, drawings, written procedures, and instructions.
- 11864
- 11865 c. Measures to describe how the applicant will specify quality standards in the design
11866 documents and control deviations and changes from these quality standards.
- 11867
- 11868 d. Measures to describe how the applicant will review designs to ensure that design
11869 characteristics can be controlled, inspected, and tested and that inspection and test
11870 criteria are identified.
- 11871
- 11872 e. Measures to describe how the applicant will establish both internal and external design
11873 interface controls. These controls should include review, approval, release, distribution,
11874 and revision of documents involving design interfaces with participating design
11875 organizations.
- 11876
- 11877 f. Measures to describe how they will properly select and perform design verification
11878 processes such as design reviews, alternative calculations, or qualification testing.
11879 When a test program is to be used to verify the adequacy of a design, the measures
11880 should be developed to describe how they will use a qualification test of a prototype unit
11881 under adverse design conditions.
- 11882
- 11883 g. Design verification constitutes confirmation that the design of the SSC is suitable for its
11884 intended purpose. Measures to ensure design verifications are completed by an
11885 individual with a level of skill at least equal to that of the original designer, recognizing
11886 design checking can be performed by a less experienced person. (As an example,
11887 design checking, which should also be performed, includes confirmation of the numerical
11888 accuracy of computations and the accuracy of data input to computer codes.
11889 Confirmation that the correct computer code has been used is part of design
11890 verification.) Measures to describe how design verification will be performed by persons
11891 other than those performing design checking. In addition, measures to include how
11892 individuals or groups responsible for design verification will not include the original
11893 designer and normally not include the designer's immediate supervisor.
- 11894

- 11895 h. Measures to ensure design and specification changes are subject to the same design
11896 controls and the same or equivalent approvals that were applicable to the original
11897 design.
11898
- 11899 i. Measures to ensure the documentation of all errors and deficiencies in the design or the
11900 design process that could adversely affect SSCs important to safety. In addition, the
11901 applicant should provide measures for adequate corrective action, including root cause
11902 evaluation of significant errors and deficiencies, to preclude repetition.
11903
- 11904 j. Before selecting materials, parts, and equipment that are standard, commercial (off-the-
11905 shelf), or have been previously approved for a different application, measures should be
11906 provided to review the suitability of any materials, parts, and equipment for the intended
11907 application.
11908
- 11909 k. Measures to provide written procedures to identify and control the authority and
11910 responsibilities of all individuals or groups responsible for design reviews and other
11911 design verification activities.
11912
- 11913 l. Measures that include the use of valid industry standards and specifications for the
11914 selection of suitable materials, parts, equipment, and processes for SSCs that are
11915 important to safety.
11916

11917 **14.5.4 Procurement Document Control**

11918 Documents used to procure SSCs or services should include or reference applicable design
11919 bases and other requirements necessary to ensure adequate quality. The following are
11920 examples of areas/items that may be addressed to support implementation of the quality
11921 criteria:
11922

- 11923 a. Measures to establish procedures that clearly delineate the sequence of actions to be
11924 accomplished in the preparation, review, approval, and control of procurement
11925 documents.
11926
- 11927 b. Measures to ensure that qualified personnel review and concur with the adequacy of
11928 quality requirements stated in procurement documents. These measures should also
11929 ensure that the quality requirements are correctly stated, inspectible, and controllable;
11930 there are adequate acceptance and rejection criteria; and the procurement document
11931 has been prepared, reviewed, and approved in accordance with QA program
11932 requirements.
11933
- 11934 c. Measures to document the review and approval of procurement documents before they
11935 are released, and the documentation should be available for verification.
11936
- 11937 d. Procurement documents should identify the applicable QA requirements that should be
11938 compiled and described in the supplier's QA program. In addition, the applicant should
11939 review and concur with the supplier's QA program.
11940
- 11941 e. Measures to ensure procurement documents contain or reference the regulatory
11942 requirements, design bases, and other technical requirements.
11943
11944

- 11945 f. Measures to ensure procurement documents identify the documentation (e.g., drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, and chemical and physical test results of material) to be prepared, maintained, and submitted to the purchaser for review and approval.
- 11946
- 11947
- 11948
- 11949
- 11950
- 11951 g. Measures to ensure procurement documents identify records to be retained, controlled, and maintained by the supplier and those records to be delivered to the purchaser before use or installation of the hardware.
- 11952
- 11953
- 11954
- 11955 h. Measures to ensure procurement documents specify the procuring agency's right of access to the supplier's facilities and records for source inspection and audit.
- 11956
- 11957
- 11958 i. Measures to ensure that changes and revisions to procurement documents are subject to the same or equivalent review and approval as the original documents.
- 11959
- 11960

11961 **14.5.5 Instructions, Procedures, and Drawings**

11962

11963 The QAPD should define the applicant's proposed procedures for ensuring that activities affecting quality will be prescribed by, and performed in accordance with, documented instructions, procedures, or drawings of a type appropriate for the circumstances. The following are examples of areas/items that may be addressed to support implementation of the quality criteria:

- 11968 a. Measures to ensure activities affecting quality are prescribed and accomplished in accordance with documented instructions, procedures, or drawings.
- 11969
- 11970
- 11971 b. Measures to establish provisions that clearly delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of instructions, procedures, and drawings.
- 11972
- 11973
- 11974
- 11975 c. Measures to ensure instructions, procedures, and drawings specify the methods for complying with each of the applicable QA criteria.
- 11976
- 11977
- 11978
- 11979 d. Measures to ensure instructions, procedures, and drawings include quantitative acceptance criteria (such as dimensions, tolerances, and operating limits) as well as qualitative acceptance criteria (such as workmanship samples) as verification that activities important to safety have been satisfactorily accomplished.
- 11980
- 11981
- 11982
- 11983 e. Measures to ensure the QA organization reviews and concurs with the procedures, drawings, and specifications related to inspection plans, tests, calibrations, and special processes as well as any subsequent changes to these documents.
- 11984
- 11985
- 11986
- 11987

11988 **14.5.6 Document Control**

11989

11990 The QAPD should define the applicant's proposed procedures for preparing, issuing, and revising documents that specify quality requirements or prescribe activities affecting quality. The following are examples of areas/items that may be addressed to support implementation of the quality criteria:

11991

11992

11993

11994

- 11995 a. The QAPD should identify all documents to be controlled under this subsection. As a
 11996 minimum, this should include design specifications; design and fabrication drawings;
 11997 procurement documents; QA manuals; design criteria documents; fabrication,
 11998 inspection, and testing instructions; and test procedures.
 11999
- 12000 b. Measures to ensure establishment of procedures to control the review, approval, and
 12001 issuance of documents and changes thereto before release to ensure that the
 12002 documents are adequate and applicable quality requirements are stated.
 12003
- 12004 c. Measures to ensure establishment of provisions to identify individuals or groups
 12005 responsible for reviewing, approving, and issuing documents and revisions thereto.
 12006
- 12007 d. Measures to ensure document revisions receive review and approval by the same
 12008 organizations that performed the original review and approval or by other qualified
 12009 responsible organizations designated by the applicant.
 12010
- 12010 e. Measures to ensure that approved changes be included in instructions, procedures,
 12011 drawings, and other documents before the change is implemented.
 12012
- 12013 f. Measures to ensure the control of obsolete or superseded documents to prevent
 12014 inadvertent use.
 12015
- 12016 g. Measures to ensure documents are available at the location where the activity is
 12017 performed.
 12018
- 12019 h. Measures to ensure establishment of a master list (or equivalent) to identify the current
 12020 revision number of instructions, procedures, specifications, drawings, and procurement
 12021 documents. In addition, measures to ensure updating of the list and distribution of it to
 12022 predetermined, responsible personnel to preclude use of superseded documents.
 12023

12024 **14.5.7 Control of Purchased Material, Equipment, and Services**
 12025

12026 The QAPD should define the applicant's proposed procedures for controlling purchased
 12027 material, equipment, and services to ensure conformance with specified requirements. The
 12028 following are examples of areas/items that may be addressed to support implementation of the
 12029 quality criteria:
 12030

- 12031 a. Measures to ensure qualified personnel evaluate the supplier's capability to provide
 12032 services and products of acceptable quality before the award of the procurement order
 12033 or contract. In addition, measures to ensure QA and engineering groups participate in
 12034 the evaluation of those suppliers providing critical items and services important to safety,
 12035 and the applicant should define the responsibilities for each group's participation.
 12036
- 12037 b. Measures to ensure evaluation of suppliers on the basis of one or more of the following
 12038 criteria:
 12039
- 12040 • The supplier's capability to comply with the elements of the QA criteria that are
 12041 applicable to the type of material, equipment, or service being procured.
 12042
 - 12043 • Review of previous records and performance of suppliers who have provided
 12044 similar articles or services of the type being procured.
 12045

- 12046 • A survey of the supplier's facilities and QA program to assess the capability to
- 12047 supply a product that meets applicable design, manufacturing, and quality
- 12048 requirements.
- 12049
- 12050 c. Measures to ensure documentation and filing of the results of supplier evaluations.
- 12051
- 12052 d. Measures to ensure planning and performing adequate surveillance of suppliers during
- 12053 fabrication, inspection, testing, and shipment of materials, equipment, and components
- 12054 in accordance with written procedures to ensure conformance to the purchase order
- 12055 requirements. In addition the measures should ensure that the procedures provide the
- 12056 following information:
- 12057
- 12058 • Instructions that specify the characteristics or processes to be witnessed,
- 12059 inspected or verified, and accepted; the method of surveillance and the extent of
- 12060 documentation required; and those responsible for implementing these
- 12061 instructions.
- 12062
- 12063 • Procedures for audits and surveillance to ensure that the supplier complies with
- 12064 the quality requirements (surveillance should be performed for SSCs for which
- 12065 verification of procurement requirements cannot be determined upon receipt).
- 12066
- 12067 e. Measures to ensure the supplier furnish the following records to the purchaser:
- 12068
- 12069 • Documentation that identifies the purchased material or equipment and the
- 12070 specific procurement requirements (e.g., codes, standards, and specifications)
- 12071 met by the items.
- 12072
- 12073 • Documentation that identifies any procurement requirements that have not been
- 12074 met and a description of any nonconformances designated "accept as is" or
- 12075 "repair."
- 12076
- 12077 f. Measures to describe the proposed procedures for reviewing and accepting these
- 12078 documents and, as a minimum, to ensure that this review and acceptance will be
- 12079 undertaken by a responsible QA individual.
- 12080
- 12081 g. Measures to ensure the conduct periodic audits, independent inspections, or tests to
- 12082 ensure the validity of the suppliers' certificates of conformance.
- 12083
- 12084 h. Measures to ensure the performance of a receiving inspection of the supplier-furnished
- 12085 material, equipment, and services to ensure fulfillment of the following criteria:
- 12086
- 12087 • The material, component, or equipment should be properly identified in a manner
- 12088 that corresponds with the identification on the purchasing and receiving
- 12089 documentation.
- 12090
- 12091 • Material, components, equipment, and acceptance records should be inspected
- 12092 and judged acceptable in accordance with predetermined inspection instructions
- 12093 before installation or use.
- 12094

- 12095 • Inspection records or certificates of conformance attesting to the acceptance of
- 12096 material, components, and equipment should be available before installation or
- 12097 use.
- 12098
- 12099 • Items accepted and released should be identified as to their inspection status
- 12100 before they are forwarded to a controlled storage area or released for installation
- 12101 or further work.
- 12102
- 12103 i. Measures to assess the effectiveness of suppliers' quality controls at intervals consistent
- 12104 with the importance to safety, complexity, and quantity of the SSCs procured.
- 12105

12106 **14.5.8 Identification and Control of Materials, Parts, and Components**

12107

12108 The QAPD should define the applicant's proposed provisions for identifying and controlling

12109 materials, parts, and components to ensure that incorrect or defective SSCs are not used. The

12110 following are examples of areas/items that may be addressed to support implementation of the

12111 quality criteria:

- 12112
- 12113 a. Measures to establish procedures to identify and control materials, parts, and
- 12114 components (including partially fabricated subassemblies).
- 12115
- 12116 b. Measures to determine identification requirements during generation of specifications
- 12117 and design drawings.
- 12118
- 12119 c. Measures to ensure that identification will be maintained either on the item or on records
- 12120 traceable to the item to preclude use of incorrect or defective items.
- 12121
- 12122 d. Measures to ensure Identification of materials and parts of important-to-safety items are
- 12123 traceable to the appropriate documentation (such as drawings, specifications, purchase
- 12124 orders, manufacturing and inspection documents, deviation reports, and physical and
- 12125 chemical mill test reports).
- 12126
- 12127 e. Measures to ensure the location and method of identification does not affect the fit,
- 12128 function, or quality of the item being identified.
- 12129
- 12130 f. Measures to verify and document the correct identification of all materials, parts, and
- 12131 components before releasing them for fabrication, assembly, shipping, and installation.
- 12132

12133 **14.5.9 Control of Special Processes**

12134

12135 The QAPD should describe the controls that the applicant will establish to ensure the

12136 acceptability of special processes (such as welding, heat treatment, nondestructive testing, and

12137 chemical cleaning) and that the proposed controls are performed by qualified personnel using

12138 qualified procedures and equipment. The following are examples of areas/items that may be

12139 addressed to support implementation of the quality criteria:

- 12140
- 12141 a. Measures to establish procedures to control special processes (such as welding, heat
- 12142 treating, nondestructive testing, and cleaning) for which direct inspection is generally
- 12143 impossible or disadvantageous. In addition, the applicant should provide a listing of
- 12144 these special processes.
- 12145

- 12146 b. Measures to qualify procedures, equipment, and personnel connected with special
12147 processes in accordance with applicable codes, standards, and specifications.
12148
12149 c. Measures to ensure qualified personnel perform special processes in accordance with
12150 written process sheets (or the equivalent) with recorded evidence of verification.
12151
12152 d. Measures to establish, file, and keep current qualification records of procedures,
12153 equipment, and personnel associated with special processes.
12154

12155 **14.5.10 Licensee Inspection**
12156

12157 The QAPD should define the applicant's proposed provisions for inspection of activities affecting
12158 quality to verify conformance with instructions, procedures, and drawings. The following are
12159 examples of areas/items that may be addressed to support implementation of the quality
12160 criteria:
12161

- 12162 a. Measures to establish, document, and conduct an inspection program that effectively
12163 verifies conformance of quality-affecting activities with requirements in accordance with
12164 written, controlled procedures.
12165
12166 b. Measures to ensure inspection personnel are sufficiently independent from the
12167 individuals performing the activities being inspected.
12168
12169 c. Measures to ensure inspection procedures, instructions, and check lists provide the
12170 following details:
12171
12172 • Identification of characteristics and activities to be inspected.
12173
12174 • Identification of the individuals or groups responsible for performing the
12175 inspection operation.
12176
12177 • Acceptance and rejection criteria.
12178
12179 • A description of the method of inspection.
12180
12181 • Procedures for recording evidence of completing and verifying a manufacturing,
12182 inspection, or test operation.
12183
12184 • Identification of the recording inspector or data recorder and the results of the
12185 inspection operation.
12186
12187 d. Measures to ensure the use of inspection procedures or instructions with the necessary
12188 drawings and specifications when performing inspection operations.
12189
12190 e. Measures to qualify inspectors in accordance with applicable codes, standards, and
12191 company training programs and in addition keeping inspector's qualifications and
12192 certifications current.
12193
12194 f. Measures to inspect modifications, repairs, and replacements in accordance with the
12195 original design and inspection requirements or acceptable alternatives.
12196

- 12197 g. Measures to establish provisions that identify mandatory inspection hold points for
12198 witnessing by a designated inspector.
- 12199
- 12200 h. Measures to identify the individuals or groups who will perform receiving and process
12201 verification inspections, and should demonstrate that these individuals or groups have
12202 sufficient independence and qualifications.
- 12203
- 12204 l. Measures to establish provisions for indirect control by monitoring processing methods,
12205 equipment, and personnel if direct inspection is not possible.
- 12206

12207 **14.5.11 Test Control**

12208

12209 The QAPD should define the applicant's proposed provisions for tests to verify that SSCs
12210 conform to specified requirements and will perform satisfactorily in service. The following are
12211 examples of areas/items that may be addressed to support implementation of the quality
12212 criteria:

- 12213
- 12214 a. Measures to establish, document, and conduct a test program to demonstrate that the
12215 item will perform satisfactorily in service in accordance with written, controlled
12216 procedures.
- 12217
- 12218 b. Measures to ensure written test procedures incorporate or reference the following
12219 information:
- 12220
- 12221 • Requirements and acceptance limits contained in applicable design and
12222 procurement documents.
- 12223
- 12224 • Instructions for performing the test.
- 12225
- 12226 • Test prerequisites.
- 12227
- 12228 • Mandatory inspection hold points.
- 12229
- 12230 • Acceptance and rejection criteria.
- 12231
- 12232 • Methods of documenting or recording test data results.
- 12233
- 12234 c. Measures to ensure a qualified, responsible individual or group document test results
12235 and evaluate their acceptability. When practicable, the measures should ensure testing
12236 of the SSC occurs under conditions that will be present during normal and anticipated
12237 off-normal operations.
- 12238

12239 **14.5.12 Control of Measuring and Test Equipment**

12240

12241 The QAPD should define the applicant's proposed provisions to ensure that tools, gauges,
12242 instruments, and other measuring and testing devices are properly identified, controlled,
12243 calibrated, and adjusted at specified intervals. The following are examples of areas/items that
12244 may be addressed to support implementation of the quality criteria:

- 12245
- 12246 a. Measures to ensure documented procedures describe the calibration technique and
12247 frequency, maintenance, and control of all measuring and test equipment (instruments,

- 12248 tools, gauges, fixtures, reference and transfer standards, and nondestructive test
- 12249 equipment) that will be used in the measurement, inspection, and monitoring of SSCs
- 12250 that are important to safety.
- 12251
- 12252 b. Measures to ensure measuring and test equipment are identified and traceable to the
- 12253 calibration test data.
- 12254
- 12255 c. Measures to ensure the use of labels, tags, or documents for measuring and test
- 12256 equipment to indicate the date of the next scheduled calibration and to provide
- 12257 traceability to calibration test data.
- 12258
- 12259 d. Measures to calibrate measuring and test instruments at specified intervals on the basis
- 12260 of the required accuracy, precision, purpose, degree of usage, stability characteristics,
- 12261 and other conditions that could affect the accuracy of the measurements.
- 12262
- 12263 e. Measures to assess the validity of previous inspections when measuring and test
- 12264 equipment is found to be out of calibration. In addition, measures should also be
- 12265 provided to document the assessment and take control of the out of calibration
- 12266 equipment.
- 12267
- 12268 f. Measures to document and maintain the complete status of all items under the
- 12269 calibration system.
- 12270
- 12271 g. Measures to ensure reference and transfer standards are traceable to nationally
- 12272 recognized standards; where national standards do not exist, the applicant should
- 12273 establish provisions to document the basis for calibration.
- 12274

14.5.13 Handling, Storage, and Shipping Control

The QAPD should define the applicant's proposed provisions to control the handling, storage, shipping, cleaning, and preservation of SSCs in accordance with work and inspection instructions to prevent damage, loss, and deterioration. The following are examples of areas/items that may be addressed to support implementation of the quality criteria:

- 12281
- 12282 a. Measures to establish and accomplish special handling, preservation, storage, cleaning,
- 12283 packaging, and shipping requirements in accordance with predetermined work and
- 12284 inspection instructions.
- 12285
- 12286 b. Measures to control the cleaning, handling, storage, packaging, shipping, and
- 12287 preservation of materials, components, and systems in accordance with design and
- 12288 specification requirements to preclude damage, loss, or deterioration by environmental
- 12289 conditions (such as temperature or humidity).
- 12290

14.5.14 Inspection, Test, and Operating Status

The QAPD should define the applicant's proposed provisions to control the inspection, test, and operating status of SSCs to prevent inadvertent use or bypassing of inspections and tests. The following are examples of areas/items that may be addressed to support implementation of the quality criteria:

- 12291
- 12292
- 12293
- 12294
- 12295
- 12296
- 12297
- 12298 a. Measures to know the inspection and test status of items throughout fabrication.

- 12299
12300 b. Measures to establish procedures to control the application and removal of inspection
12301 and welding stamps and operating status indicators (such as tags, markings, labels, and
12302 stamps).
- 12303
12304 c. Measures to ensure procedures under the cognizance of the QA organization controls
12305 the bypassing of required inspections, tests, and other critical operations.
- 12306
12307 d. Measures to specify the organization responsible for documenting the status of
12308 nonconforming, inoperative, or malfunctioning SSCs and identifying the item to prevent
12309 inadvertent use.

12310
12311 **14.5.15 Nonconforming Materials, Parts, or Components**

12312
12313 The QAPD should define the applicant's proposed provisions to control the use or disposition of
12314 nonconforming materials, parts, or components. The following are examples of areas/items that
12315 may be addressed to support implementation of the quality criteria:

- 12316
12317 a. Measures to establish procedures to control the identification, documentation, tracking,
12318 segregation, review, disposition, and notification of affected organizations regarding
12319 nonconforming materials, parts, components, services, or activities.
- 12320
12321 b. Measures to provide for adequate documentation to identify nonconforming items and
12322 describe the nonconformance, its disposition, and the related inspection requirements.
12323 The measures should also provide for adequate documentation and include signature
12324 approval of the disposition.
- 12325
12326 c. Measures to establish provisions to identify those individuals or groups with the
12327 responsibility and authority for the disposition and closeout of nonconformances.
- 12328
12329 d. Measures to ensure nonconforming items are segregated from acceptable items and
12330 identified as discrepant until properly dispositioned and closed out.
- 12331
12332 e. Measures to verify the acceptability of reworked or repaired materials, parts, and SSCs
12333 by re-inspecting and retesting the item as originally inspected and tested or by using a
12334 method that is at least equal to the original inspection and testing method. In addition,
12335 the measures should provide for documentation of the relevant inspection, testing,
12336 rework, and repair procedures.
- 12337
12338 f. Measures to ensure nonconformance reports designated "accept as is" or "repair" are
12339 made part of the inspection records and forwarded with the hardware to the customer for
12340 review and assessment.
- 12341
12342 g. Measures to periodically analyze nonconformance reports to show quality trends and
12343 help identify root causes of nonconformances. Significant results should be reported to
12344 responsible management for review and assessment.

12345
12346 **14.5.16 Corrective Action**

12347
12348 The QAPD should define the applicant's proposed provisions to ensure that conditions adverse
12349 to quality are promptly identified and corrected, and that measures are taken to preclude

12350 recurrence. The following are examples of areas/items that may be addressed to support
12351 implementation of the quality criteria:

- 12352
- 12353 a. Measures to evaluate conditions adverse to quality (such as nonconformances, failures,
12354 malfunctions, deficiencies, deviations, and defective material and equipment) in
12355 accordance with established procedures to assess the need for corrective action.
- 12356
- 12357 b. Measures to initiate corrective action to preclude recurrence of a condition identified as
12358 adverse to quality.
- 12359
- 12360 c. Measures to conduct follow-up activities to verify proper implementation of corrective
12361 actions and close out the corrective action documentation in a timely manner.
- 12362
- 12363 d. Measures to document significant conditions adverse to quality, as well as the root
12364 causes of the conditions, and the corrective actions taken to remedy the and preclude
12365 recurrence of the conditions. In addition, this information should be reported to
12366 cognizant levels of management for review and assessment.

12367

12368 **14.5.17 Quality Assurance Records**

12369

12370 The SAR should define the applicant's proposed provisions for identifying, retaining, retrieving,
12371 and maintaining records that document evidence of the control of quality for activities and SSCs
12372 important to safety. The following are examples of areas/items that may be addressed to
12373 support implementation of the quality criteria:

- 12374
- 12375 a. Measures to define the scope of the records program such that sufficient records will be
12376 maintained to provide documentary evidence of the quality of items and activities
12377 affecting quality. To minimize the retention of unnecessary records, the records program
12378 should list records to be retained by "type of data" rather than by record title.
- 12379
- 12380 b. Measures to ensure that QA records include operating logs; results of reviews,
12381 inspections, tests, audits, and material analyses; monitoring of work performance;
12382 qualification of personnel, procedures, and equipment; and other documentation such as
12383 drawings, specifications, procurement documents, calibration procedures and reports,
12384 design review and peer review reports, nonconformance reports, and corrective action
12385 reports.
- 12386
- 12387 c. Measures to ensure records are identified and retrievable.
- 12388
- 12389 d. Measures to ensure requirements and responsibilities for record creation, transmittal,
12390 retention (such as duration, location, fire protection, and assigned responsibilities), and
12391 maintenance subsequent to completion of work are consistent with applicable codes,
12392 standards, and procurement documents.
- 12393
- 12394 e. Measures to ensure inspection and test records contain the following information, where
12395 applicable:
- 12396
- 12397 • A description of the type of observation.
 - 12398 • The date and results of the inspection or test.
 - 12399 • Information related to conditions adverse to quality.
 - 12400 • Identification of the inspector or data recorder.

- 12401 • Evidence as to the acceptability of the results.
- 12402 • Action taken to resolve any noted discrepancies.
- 12403
- 12404 f. Measures to ensure record storage facilities are constructed, located, and secured to
- 12405 prevent destruction of the records by fire, flood, theft, and deterioration by environmental
- 12406 conditions (such as temperature or humidity). In addition, the facilities are to be
- 12407 maintained by, or under the control of, the licensee throughout the life of the DSS or the
- 12408 individual product.
- 12409

14.5.18 Audits

The QAPD should define the applicant's proposed provisions for planning and scheduling audits to verify compliance with all aspects of the QA program, and to determine the effectiveness of the overall program. The following are examples of areas/items that may be addressed to support implementation of the quality criteria:

- 12416
- 12417 a. Measures to perform audits in accordance with written procedures or checklists;
- 12418 qualified personnel tasked with performing these audits should not have direct
- 12419 responsibility for the achievement of quality in the areas being audited.
- 12420
- 12421 b. Measures to ensure audit results are documented and reviewed with management
- 12422 having responsibility in the area audited.
- 12423
- 12424 c. Measures to establish provisions for responsible management to undertake appropriate
- 12425 corrective action as a follow-up to audit reports. In addition, the measures should
- 12426 ensure auditing organizations schedule and conduct appropriate follow-up to ensure that
- 12427 the corrective action is effectively accomplished.
- 12428
- 12429 d. Measures to perform both technical and QA programmatic audits to achieve the
- 12430 following objectives:
- 12431
- 12432 • Provide a comprehensive independent verification and evaluation of procedures
- 12433 and activities affecting quality.
- 12434
- 12435 • Verify and evaluate suppliers' QA programs, procedures, and activities.
- 12436
- 12437 e. Measures to ensure audits are led by appropriately qualified and certified audit
- 12438 personnel from the QA organization. The measures should also ensure that the audit
- 12439 team membership include personnel (not necessarily QA organization personnel) having
- 12440 technical expertise in the areas being audited.
- 12441
- 12442 f. Measures to schedule regular audits on the basis of the status and importance to safety
- 12443 of the activities being audited. The measures should also address that audits be
- 12444 initiated early enough to ensure effective QA during design, procurement, and
- 12445 contracting activities.
- 12446
- 12447 g. Measures to analyze and trend audit deficiency data as well as ensuring resultant
- 12448 reports, indicating quality trends and the effectiveness of the QA program, should be
- 12449 given to management for review, assessment, corrective action, and follow-up.
- 12450

- 12451 h. Measures to ensure that audits objectively assess the effectiveness and proper
12452 implementation of the QA program and should address the technical adequacy of the
12453 activities being conducted.
12454
12455 I. Measures to establish provisions requiring the performance of audits in all areas to
12456 which the requirements of the QA program apply.
12457

12458 **14.6 Evaluation Findings**

12459

12460 If the reviewer determines that the applicant's QAPD does not adequately address the Part 72
12461 requirements, a request for additional information (RAI) must be prepared and submitted to the
12462 Project Manager to be forwarded to the applicant for resolution and response to the NRC. If the
12463 reviewer concludes that information provided with the application, along with additional
12464 information provided in response to NRC RAI(s), shows that the QA program description meets
12465 the acceptance requirements referenced in Section 14.4, findings of the following type should
12466 be included in the staff's SER or in a letter to the applicant, if the applicant's QA program
12467 description was submitted separate from a SAR.
12468

12469 (finding numbering is for convenience in referencing within the FSRP and SER):
12470

12471 F14.1 Based upon a review and evaluation of the QA program description contained in the
12472 Safety Analysis Report or applicant's submittal (identified by date and any other pertinent
12473 identifiers) for a DSS, the staff concludes that:
12474

- 12475 • The licensee's description of the QA program indicates requirements,
12476 procedures, and controls that, when properly implemented, should comply with
12477 the requirements of 10 CFR 72, Subpart G.
12478
- 12479 • The licensee's description of the QA program covers activities affecting SSCs
12480 important to safety as identified in the Safety Analysis Report.
12481
- 12482 • The licensee's description of the QA program describes organizations and
12483 persons performing QA functions indicating that sufficient independence and
12484 authority should exist to perform their functions without undue influence from
12485 those directly responsible for costs and schedules.
12486
- 12487 • The licensee's description of the QA program is in compliance with applicable
12488 NRC regulations and industry standards, and the acceptance of the QA program
12489 description by NRC allows implementation of the associated QA program for the
12490 (specify: design, fabrication and construction, operation, decommissioning)
12491 phases of the installation's life cycle.

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APPENDIX A CONSOLIDATED REFERENCES

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13091 **B.2 Scope, Approach and Process Description**

13092
13093 **B.2.1 Scope**

13094
13095 The scope of the SRP risk-informing effort includes all SRP sections related to technical
13096 disciplines (e.g., criticality). Within each of these sections, only the review procedures were
13097 prioritized. The requirements and their acceptance criteria contained in each section were not
13098 prioritized, since these need to be met regardless of the priority of its corresponding review
13099 procedure.

13100
13101 **B.2.2 Approach**

13102
13103 The approach used in developing the prioritization process is a graded approach that combines
13104 risk insights with deterministic considerations and operating experience. It is directed to assess
13105 the relative value of performing each review procedure and results in a qualitative prioritization
13106 considering:

- 13107
- 13108 1) The likelihood of the applicant's non-compliance with a review procedure in the SRP.
 - 13109 2) The perceived "value added" provided by the NRC review of a given SRP procedural
13110 step.
 - 13111 3) The potential consequence if the non-compliance were to remain undetected and
13112 uncorrected.
 - 13113 4) The impact on defense-in-depth if the non-compliance remains undetected, assuming
13114 the review procedure being prioritized was related to a defense-in-depth item.
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13119 The risk insights are those associated with risk to workers as well as risk to the public.
13120 The prioritization was done on a generic basis (i.e., no specific dry cask design being
13121 considered) using the SRP review procedures identified for prioritization. However, it is always
13122 possible that a design being reviewed will have such unique features (e.g., new material, new
13123 configurations) that the prioritization needs to be revisited. This can be done on a case-by-case
13124 basis by reapplying this process on an actual application.

13125
13126 Finally, in developing the prioritization approach and process, certain assumptions were
13127 developed . These assumptions included:

- 13128
- 13129 • The cost of correcting a non-compliance was not a factor included in the process.
 - 13130 • The time and resources required to perform a review procedure were not factors
13131 included in the process.
 - 13132 • Dose thresholds used in this process were consistent with thresholds established in
13133 10 CFR 20 and 10 CFR 72.104.
 - 13134 • The "value added" by the review was consistent with the current review level of effort
13135 and staff experience.
 - 13136 • Items to be prioritized were chosen such that overlap between them is minimized.
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- All other requirements, except those included in the specific SRP review procedure being prioritized, were assumed to be satisfied.

13145 **B.2.3 Process Description**

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13147 The process was applied to each technical discipline area in the SRP. The process was
13148 implemented by the NRC staff reviewers responsible for that discipline (i.e., multiple reviewers
13149 participated in the prioritization of each review procedure, and the final priority was developed
13150 based upon a consensus among the reviewers). The process involved looking at each SRP
13151 review procedure paragraph (or group of paragraphs) in each technical discipline area, and
13152 asking a structured set of questions. These questions addressed:

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- What is the likelihood of the applicant not meeting the requirement(s) contained in the SRP review procedure being prioritized (need for staff review)?
 - What is perceived value added by the staff review (i.e. likelihood of identifying a non-compliance for a given review procedure.
 - What is the potential consequence to public and/or worker radiological safety if the requirement(s) remain unmet?
 - What is the impact on defense-in-depth, if any, if the review procedure remains unmet?

13165 The answers to the above questions were based upon the judgment of the NRC staff reviewers
13166 who participated in the prioritization process. This judgment reflected the reviewer's experience
13167 with current and previous applications and their views regarding potential future problems.

13168
13169 NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a
13170 Nuclear Power Plant" was previously developed to assess the risk to the public of a specific dry
13171 storage system at a boiling water reactor site to postulated events. The PRA information was
13172 not explicitly used in this SRP prioritization because it was limited in scope and assumed that
13173 the cask was properly designed, constructed and tested. Furthermore, the PRA did not address
13174 the factors listed in Table B-1 and B-2. It only assessed the risk during cask use from external
13175 hazards (e.g., fire) and operational errors (e.g., cask drop). Some of these accident sequences
13176 were also outside the scope of regulatory accidents typically evaluated under Part 72 for
13177 certified cask systems. In summary, the prioritized review procedures in the SRP address cask
13178 design, construction and testing, operations, and performance under normal and accident
13179 conditions to verify compliance with 10 CFR Part 72.

13180
13181 The steps the reviewers took in prioritizing each SRP review procedure were the following.
13182 First, the answers to the first two questions were qualitatively determined using a 5 tier
13183 qualitative ranking. Second, the answer to the third question was qualitatively determined using
13184 a 3-tier qualitative ranking system. The ranking systems are defined in Tables B-1, B-2 and B-
13185 3. The quantitative values used in Tables B-1, B-2 and B-3 are intended to serve as guidance
13186 in the selection of the appropriate qualitative ranking and reflect conservative estimates so as to
13187 provide a margin to account for uncertainties. The qualitative rankings resulting from Tables B-
13188 1, B-2 and B-3 were then assigned point values as shown in Table B-4. The point values
13189 corresponding to the qualitative rankings from Tables B-1, B-2 and B-3 were added together
13190 and, using the guidance described in Table B-4, an overall qualitative risk component of the

13191 prioritization (High, Medium or Low) was determined. The reason the scores from Tables B-1,
 13192 B-2 and B-3 were added is that each is a reflection of the importance of the NRC staff
 13193 performing the review procedure being prioritized. Finally, the answer to the last question
 13194 (defense-in-depth) was qualitatively determined using a 3-tier scale (High, Medium or Low)
 13195 following the guidance contained in Table B-5 and Attachment 2 and the reviewer's expert
 13196 opinion.

13197
 13198 The result was a risk-informed prioritization and, if applicable, a defense-in-depth prioritization
 13199 ranking. The final prioritization for the SRP review procedure was the overall risk ranking and, if
 13200 also related to defense-in-depth, a weighed combination of these two, with the weights
 13201 determined by the NRC staff. These weights were determined for each review procedure
 13202 prioritized and used only for that respective item (i.e., the importance of risk versus defense-in-
 13203 depth may vary from item to item). Attachment 1 to this appendix lists the detailed steps
 13204 associated with implementing the prioritization process that was used in assessing the priority of
 13205 each SRP review procedure. Attachment 2 provides a more detailed discussion on defense-in-
 13206 depth. Attachment 3 provides an example of the documentation and major considerations
 13207 associated with implementation of the process for one specific review procedure.

13208
 13209 **B.3 SRP Priority Designation and Implications**

13210
 13211 Upon completion of the prioritization process, the priority (HIGH, MEDIUM or LOW) associated
 13212 with each review procedure has been indicated in the SRP at the beginning of each paragraph
 13213 in the review procedures.

13214
 13215
 13216 The risk-informed procedures are intended to ensure that reviews are adequately focused on
 13217 areas that have the most significant impact on safety and compliance with regulatory limits. It is
 13218 important to remember that the priority designations were developed on a generic basis and
 13219 may need to be adjusted depending upon the characteristics of specific applications. It is the
 13220 responsibility of the individual reviewer to assess the design and determine the ultimate rigor
 13221 needed to make a safety determination, with reasonable assurance, in each review area.

13222
 13223 Finally it should be noted that a low or medium priority review procedure does not mean an
 13224 application is exempted from any associated regulatory requirement, design requirement, or
 13225 safety analyses that is expected within the review objectives and acceptance criteria.

13226
 13227 **Table B-1 Likelihood of Applicant's Non-Compliance with the SRP Review Procedure**
 13228

Likelihood of Not Meeting the Requirements	Description
Very High	Qualitative: Likely to occur. Quantitative: $P > 0.5$
High	Qualitative: Probably will occur. Quantitative: $0.1 < P < 0.5$
Medium	Qualitative: May occur. Quantitative: $0.03 < P < 0.1$
Low	Qualitative: Unlikely to occur. Quantitative: $0.01 < P < 0.03$

Very Low

Qualitative: Occurrence improbable.
Quantitative: $P < 0.01$

13229 **P = Probability**

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13232 **Table B-2 Potential "Valve Added" through the NRC Review Process**

13233

Likelihood that the NRC Review of a Specific Review Procedure Step Will Identify a Non-Compliance

Description

Very High

Qualitative: Likely to occur.
Quantitative: $P > 0.5$

High

Qualitative: Probably will occur.
Quantitative: $0.1 < P < 0.5$

Medium

Qualitative: May occur.
Quantitative: $0.03 < P < 0.1$

Low

Qualitative: Unlikely to occur.
Quantitative: $0.01 < P < 0.03$

Very Low

Qualitative: Not probable.
Quantitative: $P < 0.01$

13234 **P = Probability**

13235

13236

13237 **Table B-3 Potential Impact if the Non-Compliance were to remain uncorrected**

13238

Increase in Risk (Likelihood and / or Consequence) if Requirements Remain Unmet

Description

High

Qualitative: Likely to occur or significant consequences.
Quantitative: $>10^{-3}/\text{yr}^*$ or >25 rem to worker or > 1 rem to public.

Medium

Qualitative: May occur or moderate consequences.
Quantitative: $<10^{-3}/\text{yr}$ but $>10^{-5}/\text{yr}^{**}$ or 5 -25 rem to worker or 0.1 rem - 1 rem to public.

Low

Qualitative: Occurrence improbable or minimal consequences.
Quantitative: $< 10^{-5}/\text{yr}$ or less than 10 CFR 20 dose limits for workers and the public.

13239

13240 * $10^{-3}/\text{yr}$ corresponds to the likelihood of an event that could occur in one or more casks over a 20 year life of 50 casks.

13241

13242

13243 **** 10⁻⁵/yr corresponds to the likelihood of an event that could occur in one or more casks**
13244 **over a 20 year life of 5000 casks (i.e., 50 at each of 100 operating reactors).**

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Table B-4 Overall Risk Ranking

Numerical values for each qualitative risk designation for Tables B-1, B-2 and B-3 are assigned as follows (note that Table B-3 only assigns values of 1 through 3):

Very High	4
High	3
Medium	2
Low	1
Very Low	0

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For each SRP review procedure, the qualitative scores from Tables B-1, B-2 and B-3 are added and a combined qualitative score is determined as follows:

High	9 - 11
Medium	6 - 8
Low	1 - 5

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Table B-5 Defense-in-Depth Ranking

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Defense-in-depth has long been a key element of the NRC's safety philosophy. It is intended to ensure that the accomplishment of key safety functions is not dependent upon a single element of design, construction, maintenance or operation. In effect, defense-in-depth is used to provide one or more additional measures to back up the front line safety measures, to provide additional assurance that key safety functions will be accomplished. Traditional defense-in-depth measures for reactors have included items such as confinement, containment, redundant and diverse means of decay heat removal and emergency evacuation plans. For DSS, examples of measures associated with defense-in-depth are discussed in Attachment B-2. Defense-in-depth measures are generally decided upon using deterministic considerations (i.e., engineering judgment) regarding the importance of the safety function and the potential uncertainties that could affect its performance.

With respect to prioritizing the review procedures in this SRP, a review procedure can be considered associated with defense-in-depth if it is related to providing a backup to the front line of defense (e.g., confinement is generally considered a defense-in-depth measure since it provides a backup to cladding integrity).

Defense-in-depth measures are not intended to detract from the importance of front line safety measures. Defense-in-depth measures are intended to provide additional assurance so the safety function can be accomplished. It is not the intent of defense-in-depth to reduce the importance of the front line safety measures since, if their importance were reduced, the importance of the NRC staff review associated with those measures could also be reduced, which could affect the reliability or performance of the front line safety measures. This could leave the defense-in-depth measures as the primary means of performing the safety functions, instead of being the backup.

If failure to perform the review procedure could impact defense-in-depth (assuming the front line safety measure has failed) and has:

- a low likelihood and/or consequence, then the paragraph should be prioritized as "LOW."
- a medium likelihood and/or consequence, then the paragraph should be prioritized as "MEDIUM."
- a high likelihood and/or consequence, then the paragraph should be prioritized "HIGH."

Likelihood and consequence are defined in Table B-3.

13300 **Attachment B-1**

13301 **Process Steps to Prioritize SRP Review Procedures**

13302 The following steps should be followed in prioritizing each review procedure. Multiple staff
13303 reviewers in each technical area should participate in the prioritization so as to arrive at a
13304 consensus on the priority. The checklist at the end of this attachment can be used to document
13305 each step.
13306

- 13307
- 13308 1. Identify the SRP review procedures to be prioritized, with a focus on the requirements
13309 that the review procedure is checking. This will result in individual paragraphs (or
13310 groups of paragraphs) being prioritized as separate items.
13311
 - 13312 2. Estimate the likelihood that the requirement related to the SRP review procedure will not
13313 be met by the applicant by choosing the appropriate likelihood range from Table B-1
13314 (Likelihood of Applicant's Non-Compliance with the SRP Review Procedure). This
13315 estimate can be affected by several factors, including the experience of the applicant,
13316 the novelty of the technology used in the application, the difficulty level of meeting the
13317 requirement, the applicant's quality assurance program, etc.
13318

13319 The rankings listed in Table B-1 are arranged to provide more staff review effort where it
13320 is determined that the applicant is less likely to meet the review procedure. Conversely,
13321 where it is felt that the applicant will meet the review procedure, less staff effort would be
13322 required.
13323

- 13324
- 13325 3. Estimate the likelihood that if the requirement is not met, this fact will be discovered by
13326 performing the SRP review procedure. This is done by choosing the appropriate
13327 likelihood range from Table B-2 (Likelihood that the NRC Review Would Identify the
13328 Non-Compliance, Given that it Exists). This factor may be relatively high, however, there
13329 may be review procedures that have varying degrees of implementation.
13330

13331 The rankings listed in Table B-2 are arranged to continue to provide a high level of staff
13332 effort in areas where the staff review has typically identified problems. Conversely,
13333 where historical staff review efforts have not identified problems, that level of staff effort
13334 is minimized.
13335

- 13336 4. Estimate the potential radiological risk to public and worker safety if the requirement
13337 were to remain unmet. It is recognized that this is not a trivial task and that no complete
13338 probabilistic risk assessment (PRA) is available for dry casks or ISFSIs. The following
13339 was intended to aid the prioritizer with this assessment:
13340

- 13341 • Consider potential event sequences or a set of event sequences, such that the
13342 dose to the most exposed person from these sequences include the bulk of the
13343 dose from all possible sequences. The premise here is that every possible
13344 sequence of events has some likelihood of occurring and results in some dose to
13345 workers and the public. Some sequences are very likely and result in very little
13346 dose, others are very unlikely and result in very large dose, etc. The prioritizer
13347 should use experience in considering the sequence(s) that have the highest risk
13348 to the most exposed person. This is equivalent to answering the following
13349 questions:
13350

- 13351 - What can happen? (i.e., what can go wrong?)
13352 - How likely is it that that will happen?
13353 - If it does happen, what are the consequences?
13354
13355 • Using Table B-3 Potential Impact if a Non-Compliance is not identified,
13356 determine the corresponding range of increased likelihood or dose. This range
13357 corresponds with the likelihoods and / or consequences for the dominant
13358 sequences.

13359
13360 The rankings listed in Table B-3 are weighted to devote more staff resources to the
13361 review procedures that are viewed to be more risk significant and less staff resources to
13362 those that are viewed to be less risk significant.

13363
13364 5. The prioritizer now has three qualitative rankings corresponding to:

- 13365
13366 • Likelihood of the applicant not meeting the requirements.
13367 • Likelihood that the NRC Review would find the discrepancy, given that it exists.
13368 • Potential consequences if the requirements remain unmet.
13369

13370 Using these three rankings, determine the overall qualitative risk-ranking (High, Medium
13371 or Low) for this review procedure by adding the numerical values assigned to each
13372 qualitative ranking and the guidance in Table B-4.

13373
13374 6. Using Table B-5, assess the applicability and impact on defense-in-depth, if any, if the
13375 SRP review procedure is not met. Defense-in-depth consists of a number of elements
13376 as discussed in Attachment 2 and will not be applicable to all review procedures. If
13377 applicable, this step results in a High / Medium / Low qualitative ranking.

13378
13379 7. There is now a qualitative ranking and, if applicable, a qualitative defense-in-depth
13380 ranking. The method of combining these scores reflects the relative importance given to
13381 risk versus defense-in-depth. Judgment must be used to integrate these two rankings
13382 into a single ranking applicable to the SRP review procedure. This integration is done by
13383 weighing the two rankings using weights determined by the NRC reviewers. The
13384 weights are determined for each review procedure being prioritized and used for that
13385 procedure only.

13386
13387 8. A prioritization process checklist is to be filled out for each paragraph (or group of
13388 paragraphs) prioritized, so as to document the basis for the priorities assigned to each
13389 review procedure. This checklist is shown on the following page and Attachment B-3
13390 provides an example of a completed checklist for a specific review procedure.
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Prioritization Process Checklist

Chapter:

Paragraph Number:

STEP	SCORE	COMMENTS
1. Identify the SRP procedure to be prioritized.	N/A	
2. Likelihood that requirement will not be met (Table B-1).		
3. Likelihood that staff reviews will find discrepancy (Table B-2).		
4. Risk if requirement is not met (Table B-3).		
5. Determine combined risk value (Table B-4).		
6. Determine defense-in-depth value (Table B-5), if applicable.		
7. Determine relative weight of risk and defense-in-depth values determined in (steps 5 and 6 above).		
8. Overall priority (Combine risk and defense-in-depth values).		

13396
13397

13398 **Attachment B-2**

13399
13400 **Defense-in-Depth (DID)**

13401
13402 Defense-in-depth has long been a key element of NRC's safety philosophy. It is intended to
13403 ensure that the accomplishment of key safety functions is not dependent upon a single element
13404 of design, construction, maintenance or operation. In effect, defense-in-depth is used to
13405 compensate for uncertainties by employing one or more additional measures to back up the
13406 front line safety measures, thus providing additional assurance that key safety functions will be
13407 performed. Traditional defense-in-depth measures for reactors have included items such as
13408 confinement, containment, redundant and diverse means of decay heat removal and emergency
13409 evacuation plans. Defense-in-depth measures are generally decided upon using deterministic
13410 considerations (i.e., engineering judgment) regarding the importance of the safety function and
13411 the potential uncertainties that could affect its performance.

13412
13413 In the dry cask SRP prioritization, each paragraph (or group of paragraphs) to be prioritized,
13414 would be examined individually from a DID perspective to determine if that paragraph (or group
13415 of paragraphs) is related to defense-in-depth. If so, and if the paragraph is not met, a
13416 determination would then be made as to whether or not a defense-in-depth measure could be
13417 compromised and the risk significance.

13418
13419 To determine if a defense-in-depth measure could be compromised, it is first necessary to
13420 decide what are defense-in-depth measures? To help make this decision, the following
13421 guidance was used.

- 13422
- 13423 • A defense-in-depth measure is any design feature or action that is required by the SRP
13424 as a backup measure to the front line safety measures. This ensures that, if the front
13425 line safety measure is lost, the backup measure is present to perform that safety
13426 function.

13427
13428 DSS defense-in-depth measures may include:

- 13429
- 13430 • Confinement (to back up fuel clad integrity);
 - 13431
 - 13432 • Independent NRC analysis (to back up the applicant's analysis); and
 - 13433
 - 13434 • Safety margin (to provide additional assurance beyond normal design conditions).

13435
13436 SRP review procedures that relate to items that can be considered defense-in-depth should
13437 receive a DID ranking.

13438
13439 If the SRP paragraph (or group of paragraphs) being prioritized is related to a measure that
13440 meets the above guidance, then it would be evaluated as a defense-in-depth measure and
13441 prioritized as follows:

- 13442
- 13443 • If the failure of the front line and DID measures *relative to the issue identified in the SRP*
13444 *review procedure* would result in a low likelihood and / or consequence, then the
13445 paragraph should be prioritized as "LOW."
- 13446

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- If the failure of the front line and DID measures *relative to the issue identified in the SRP review procedure* would result in a medium likelihood and / or consequence, then the paragraph should be prioritized as "MEDIUM."

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- If the failure of the front line and DID measures *relative to the issue identified in the SRP review procedure* would result in a high likelihood and / or consequence, then the paragraph should be prioritized "HIGH."

13455 Risk and consequence are defined in Table B-3.

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It should be noted that defense-in-depth measures are not intended to detract from the importance of front line safety measures. Defense-in-depth measures are intended to provide additional assurance so the safety function can be accomplished.

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Attachment B-3

This attachment provides an example of a completed prioritization checklist to illustrate the level of documentation and major considerations associated with the prioritization of each specific review procedure. The review procedure used in the example is Section 4.5.4.7 "Confirmatory Analysis" in Chapter 4 "Thermal Evaluation" of NUREG-1536. A total of three staff reviewers participated in the prioritization of Chapter 4 and the prioritization input and outcome reflects a consensus among the reviewers.

Prioritization Process Checklist

Chapter: 4 - "Thermal Evaluation"

Paragraph Number: 4.5.4.7

STEP	SCORE	COMMENTS
1. Identify the SRP procedure to be prioritized.	<i>N/A</i>	<i>Done by reviewers.</i>
2. Likelihood that requirement will not be met (Table B-1).	<i>L</i>	<i>Applicant provides calculations using generally accepted analytical tools.</i>
3. Likelihood that staff reviews will find discrepancy (Table B-2).	<i>H</i>	<i>Staff provides a thorough review.</i>
4. Risk if requirement is not met (Table B-3).	<i>H</i>	<i>Fuel cladding (i.e., first line-of-defense for fission product retention) could fail if thermal analysis is incorrect.</i>
5. Determine combined risk value (Table B-4).	<i>M</i>	<i>L (1) + H (3) + H (3) = 7 (MEDIUM)</i>
6. Determine defense-in-depth value (Table B-5), if applicable.	<i>H</i>	<i>Provides independent check (i.e., second line-of-defense) as backup to front line staff review of applicant's submittal.</i>
7. Determine relative weight of risk and defense-in-depth values determined in (steps 5 and 6 above).	<i>DID > Risk</i>	<i>DID is more important than risk since it has the potential to uncover applicant or staff review errors and can provide additional insights for probing the validity of the applicant's analysis.</i>

STEP	SCORE	COMMENTS
8. Overall priority (Combine risk and defense-in-depth values).	<i>H</i>	<i>DID controls final priority.</i>

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**APPENDIX C LIST OF ISGs 1 TO 22 WITH THOSE INCORPORATED INTO
NUREG-1536 IDENTIFIED**

ISG # & Rev.	Title	1536 Status
1 Rev. 2	Damaged Fuel	Added
2	Fuel Retrievalability	NA
3	Post Accident Recovery and Compliance with 10 CFR 72.122(l)	Added
4 Rev. 1	Cask Closure Weld Inspections	Superseded by ISGs 15 and 18
5 Rev. 1	Confinement Evaluation	Added
6	Establishing Minimum Initial Enrichment for the Bounding Design Basis Fuel Assembly(s)	Added
7	Potential Generic Issue Concerning Cask Heat Transfer in a Transportation Accident	Added
8 Rev. 2	Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks	Added
9 Rev. 1	Storage of Components Associated with Fuel Assemblies	Added
10 Rev. 1	Alternatives to the ASME Code	Added
11 Rev. 3	Cladding Considerations for the Transportation and Storage of Spent Fuel	Added
12 Rev. 1	Buckling of Irradiated Fuel Under Bottom End Drop Conditions	Added, new revision pending
13	Real Individual	Added
14	Supplemental Shielding	Added
15	Materials Evaluation	Added
16	Emergency Planning	NA
17	Interim Storage of Greater Than Class C Waste	NA
18 Rev. 1	The Design & Testing of Lid Welds on Austentic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage	Added
19	Moderator Exclusion Under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e)	NA
20	Transportation Package Design Changes Authorized Under 10 CFR Part 71 Without Prior NRC Approval	NA
21	Use of Computational Modeling Software	Added
22	Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel	Added

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