8 MATERIALS EVALUATION

8.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) materials review is to ensure adequate materials performance of structures, systems, and components (SSCs) to ensure compliance with Title 10 of the *Code of Federal Regulations* (10 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," for a dry storage system (DSS) or dry storage facility (DSF), which includes independent spent fuel storage installations (ISFSIs) and monitored retrievable storage installations (MRS) involving handing, packaging, transfer, and storage. Materials must meet applicable codes, standards, and specifications and support intended functions of SSCs under all credible loads and environments for normal, off-normal, and accident conditions. The review also includes the evaluation of operations that ensure adequate materials performance, including material qualification, welding, spent nuclear fuel (SNF) drying, inerting of the confinement system, and the management of materials degradation.

8.2 Applicability

This chapter applies to the review of applications for specific licenses for an ISFSI or MRS and certificates of compliance (CoCs) of a DSS for use at a general license facility. Differences between the review of a specific license (SL) and a CoC are noted; in particular, specific licenses may involve SSCs associated with the storage of reactor-related greater-than-Class-C (GTCC) waste and high level radioactive waste (HLW) and facilities associated with SNF and waste handling, packaging, transfer, and storage.

8.3 Areas of Review

This chapter addresses the following areas of review:

- System and Facility Design
  - Drawings
  - Codes and Standards
  - Weld Design, Inspection, and Testing
- Material Properties
  - Mechanical Properties of Metals
  - Thermal Properties
  - Radiation Shielding Materials
  - Criticality Control Materials
  - Concrete and Reinforcing Steel
  - Bolt Applications
  - Seals
- Environmental Degradation; Corrosion and Other Reactions
  - Corrosion Resistance
  - Protective Coatings
  - Content Reactions
  - Management of Aging Degradation
- Fuel Cladding Integrity
  - Fuel Classification

- Uncanned Spent Fuel
- Canned Spent Fuel

8.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. Tables 8-1a and 8-1b match the relevant regulatory requirements to the areas of review covered in this chapter for applications for an ISFSI site license and CoC, respectively. The reviewer should refer to the language in the regulations and verify the association of regulatory requirements with the areas of review presented in these tables to ensure that no requirements are overlooked as a result of unique applicant design features.

Table 8-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations				
	72.24	72.120	72.122	72.124	72.128
Design Criteria	(c)(3)	(a)			(a)
Code Use and Quality Standards	(c)(4)		(a)		
Material Properties	(d)			(b)	
Environmental Degradation; Chemical and Other Reactions		(d)	(b)(1), (c)	(b)	
Fuel Cladding Integrity and Retrievability			(h)(1), (h)(5), (l)		

Table 8-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations			
	72.122[A]	72.124	72.234	72.236
Design Criteria				(b)
Code Use and Quality Standards	(a)		(b)	
Material Properties		(b)		(g)
Environmental Degradation; Chemical and Other Reactions	(b)(1), (c)	(b)		(h)
Fuel Cladding Integrity and Retrievability	(h)(1), (h)(5), (l)			(a), (m)

[A] While not directly applicable to CoCs, DSS design should facilitate general licensee compliance with these requirements.

The materials evaluation seeks to ensure that materials will perform in a manner that supports the functions of the SSCs of storage systems and site facilities by fulfilling the following principal acceptance criteria that reflect the above regulations and areas of review:

- The applicant must provide information on materials of construction, including their fabrication, testing, and general arrangement, with sufficient detail to support a safety finding.

- Materials and special processes must conform to all applicable codes and standards. Non-code materials must have adequate controls for their qualification and fabrication.

- Material properties should have an adequate technical basis and must demonstrate the ability to support the performance of the intended functions of SSCs under credible loads in normal, off-normal, and accident conditions.

- Materials must not undergo adverse environmental degradation, chemical reactions, or other reactions that could challenge the ability of SSCs to safety handle, package, transfer, and store SNF, reactor-related GTCC waste, or HLW.

- The applicant must ensure that the SNF cladding is protected against gross ruptures or otherwise be confined and that the SNF, HLW, and reactor-related GTCC waste are always retrievable.

8.5 Review Procedures

Figure 8-1 shows the interrelationship between the materials evaluation and the other areas of review described in this standard review plan (SRP). The materials reviewer should survey the safety analysis report (SAR) and design drawings to identify the materials issues that are associated with the specific design proposal in the application. Examine the chapters of the SAR on criticality, shielding, confinement, structural, and thermal to identify cross-cutting issues that should be coordinated among the technical disciplines.

8.5.1 Drawings

Licensing drawings usually appear in SAR Chapters 1 or 2. Although developed for the review of transportation packages, the staff considers the guidance in NUREG/CR-5502, “Engineering Drawings for 10 CFR Part 71 Package Approvals,” appropriate for the recommended content of storage drawings. Examine the drawings for material specifications, alternatives, and fabrication instructions including welding and nondestructive examination (NDE) requirements. Ensure that the applicant adequately specified any materials substitutes, either on the drawing or in the SAR. Ensure welding codes are clearly identified.

Standard welding and NDE symbols may be found in AWS A2.4, “Symbols for Welding, Brazing, and Nondestructive Testing,” to aid interpretation of drawings. Section 8.5.3, “Welding,” of this SRP provides additional guidance for the expected level of detail for weld filler metal and welding processes.

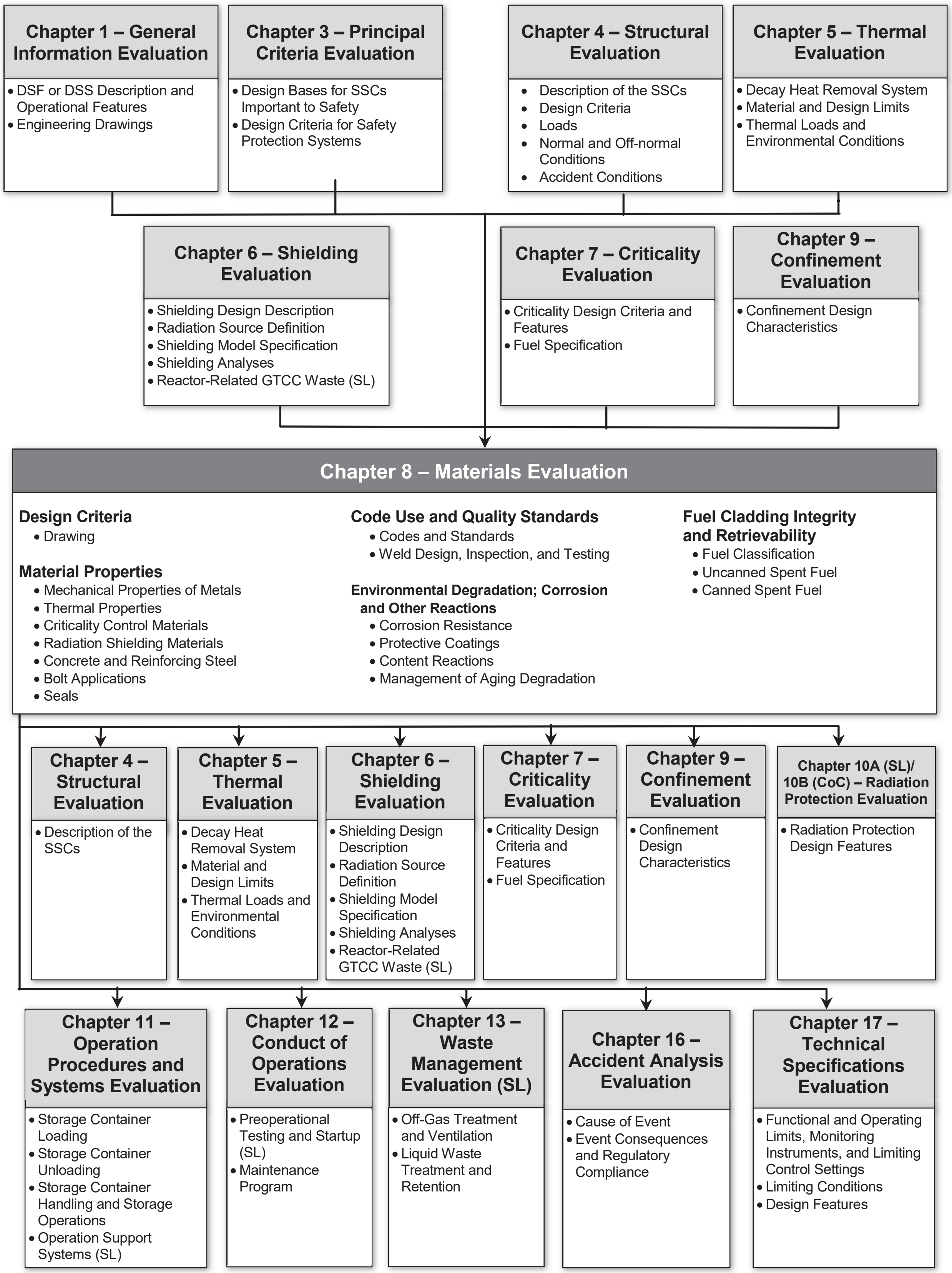

Figure 8-1 Overview of Materials Evaluation

Design drawings often do not identify a year or revision for codes and standards for materials specifications because the latest revision is widely considered to be appropriate for use. For example, metal producers routinely supply plates and forgings only to the latest revision of the ASTM standards, and thus it is expected that the DSS or DSF fabricator will necessarily procure material to the latest revision. Consequently, when a specific revision of a standard is not provided, base the materials review on the latest revision. An exception to this guidance is when this SRP or other NRC guidance recommends a particular version (or elements of an earlier revision) as a basis for the staff review. In that case, either (1) verify that key elements of the recommended earlier revision of the code or standard are still maintained in the latest version, or (2) consider whether the drawings should be revised to specifically cite the recommended earlier revision.

Other technical review disciplines may recommend that drawings include specific revisions of a code or standards associated with their review areas (e.g., SRP Section 4.5.1.1, "Structures, Systems, and Components Important to Safety," for the structural design code). In that case, ensure that materials specifications are appropriate for the specific code version cited in the drawings.

8.5.2 Codes and Standards

The following guidance describes the materials codes and standards that the NRC finds acceptable for the construction of DSSs and DSFs. In several cases, the NRC staff recommends exceptions or additions to the codes and standards to address unique aspects of DSS and DSF designs.

8.5.2.1 *Usage and Endorsement*

For SSCs important to safety, ensure that the applicant specifies U.S. industry consensus codes and standards, such as the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), American Welding Society (AWS) Code, American National Standards Institute (ANSI) standards, American Concrete Institute (ACI) Code, and ASTM International (ASTM) standards. Foreign codes and standards generally are not acceptable for SSCs or materials important to safety and would only be approved on a case-by-case basis. If used, ensure that foreign codes cross reference the appropriate ASME B&PV Code.

Approved storage containers are those that have been designed in accordance with the ASME B&PV Code. The NRC has accepted the design of confinement SSCs fabricated in accordance with ASME B&PV Code, Section III, "Rules for Construction of Nuclear Facility Components," Subsection NB, "Class 1," criteria; of fuel basket structures fabricated in accordance with ASME B&PV Code Section III, Subsection NG, "Core Supports"; and of other safety structures fabricated in accordance with ASME B&PV Code Section III, Subsection NF, "Supports." For SSCs not associated with the confinement boundary or fuel basket, the NRC has accepted alternatives to the ASME B&PV Codes cited above. For example, the NRC has accepted the design of transfer casks to ASME B&PV Code Section III, Subsection NC, "Class 2," criteria, and of other steel structures to the American Institute of Steel Construction (AISC) "Manual of Steel Construction." Finally, as discussed in detail in Section 8.5.8, "Concrete and Reinforcing Steel," of this SRP, NRC-accepted concrete structure designs have used ACI Codes.

The reviewer should ensure that the materials and their fabrication are consistent with the construction code or standard. Although written for the design of shipping containers, NUREG/CR-3854, "Fabrication Criteria for Shipping Containers," may be used to identify where materials and fabrication criteria (e.g., heat treatment, examination, testing) are defined in the ASME B&PV Code sections. SSCs important to safety that are constructed in accordance with ASME B&PV Code Section III are normally fabricated from ASME Section II materials. Important-to-safety attachments to the confinement boundary, as well as structural components of the overpack, may be ASME or ASTM materials, depending on the code of record for the component. For non-ASME SSCs important to safety, ASTM materials may be used.

Codes and standards frequently reference one another, and the reviewer should note these relationships when verifying their proper use by the applicant. For example, all ASME materials are a subset of AWS and ASTM materials. However, not all ASTM materials are endorsed for use by ASME or other codes that may be used in storage system designs.

The applicant should describe proprietary materials important to safety (specifically neutron poisons and polymeric neutron shields) adequately for the staff to make a safety finding. The reviewer should ensure that the technical specifications incorporate by reference the governing quality assurance and quality control documents, key manufacturing procedures, and key testing protocols for proprietary materials. The use of proprietary materials should be reviewed by NRC on a case-by-case basis.

The applicant may specify non-important-to-safety items by generic names such as “stainless steel,” “aluminum,” or “carbon steel,” provided that the reviewer has sufficient information to evaluate potential impacts that components that are not important to safety may have on components of packaging that are important to safety (e.g., galvanic corrosion).

8.5.2.2 *Code Case Use and Acceptability*

The reviewer should assess any referenced ASME B&PV Code cases against Regulatory Guide (RG) 1.193, “ASME Code Cases Not Approved for Use.” Note that the NRC has found Code Case N-595 (any revision) unacceptable. The reviewer should also review any referenced ASME B&PV Code cases against RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” Table 1 of RG 1.84 provides lists of cases acceptable to the NRC, while Table 2 of RG 1.84 provides a list of conditionally approved cases. The reviewer should verify that all of the supplemental requirements are met in order to provide an acceptable level of quality and safety. Also examine Tables 3, 4, and 5 of RG 1.84 to ensure that they do not reference annulled or superseded codes cases.

8.5.3 Welding

The ASME B&PV Code defines required welding criteria, including welding processes, filler metal, qualification procedures, heat treatment, and examination and testing. Review the relevant portions of the ASME B&PV Code to ensure that the SAR and drawings for the storage confinement boundary and fuel baskets are consistent with the code-required welding criteria. This review should include the relevant articles in ASME B&PV Code, Section III, Subsection NB-4000 and, in particular, Article NB-4330, “General Requirements for Welding Procedure Qualification Tests.” Although written for the welding of shipping containers, NUREG/CR-3019, “Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials,” is a relevant resource for identifying the locations in the ASME B&PV Code of the welding criteria for storage containers.

The welding of SSCs not associated with the confinement boundary or fuel baskets are frequently governed by the ASME B&PV Code (transfer casks constructed per Section III Subsection NC) or

AISC standards (canister support structures), which, in turn, may reference AWS Codes. Similar to the ASME B&PV Code, AWS D1.1, “Structural Welding Code-Steel,” and AWS D1.6, “Structural Welding Code-Stainless Steel,” provide detailed welding criteria and weld procedure qualification requirements.

If the DSS or DSF design is consistent with the ASME or AWS codes, and the SAR and design drawings clearly define the applicability of the code, there is no need to review and verify the presence of specific welding criteria, such as the filler metal and weld process, in the drawings. The staff considers the ASME and AWS codes to have been proven to be effective in controlling qualification methodology, materials, heat treating, inspection, and testing. Note that this guidance is only applicable if the materials of construction comply with the ASME or AWS codes.

If materials and welding processes are not fully consistent with the ASME or AWS codes, verify that the application provides a technical basis for the integrity of the non-code welds and that the SAR and drawings sufficiently describe the welding criteria. The technical basis for non-code welds should demonstrate that the alternative material or welding process has been qualified in a manner similar to that described in accepted codes. The specified weld metal strength should equal or exceed the specified base metal strength. In addition, filler metals and the welding parameters should be selected in consideration of the potential for microstructural phase instabilities in the weld and the weld heat affected zone. These microstructural changes may include the formation of secondary or intermetallic phases that reduce ductility or fracture toughness and/or increase the susceptibility of the weld or the weld heat affected zone to environmental degradation such as corrosion or stress corrosion cracking. For example, welding of austenitic stainless steels that are not low carbon grades can result in sensitization of the weld heat affected zones, which can increase susceptibility to intergranular corrosion and stress corrosion cracking in corrosive environments. Secondary phase formation in duplex stainless steels as a result of slow cooling during welding can significantly reduce fracture toughness.

Detailed guidance is provided below for welds associated with the confinement boundary. Confinement boundary welds provide both structural integrity and confinement leak tightness. Ensure that the applicant provided sufficient detail to demonstrate that the welds are capable of fulfilling these functions. The guidance for the design, inspection, and testing of confinement boundary welds follows the ASME B&PV Code as practicable; however, exceptions are allowed to accommodate the unique application of the codes to DSSs.

8.5.3.1 *Confinement Weld Design*

The preferred construction code for the storage confinement boundary is the ASME B&PV Code, Section III, Division 1, Subsection NB for Class 1 nuclear facility components. ASME B&PV Code Section III is supplemented by supporting code sections that detail how special processes such as welding and NDE are to be qualified and executed. ASME B&PV Code Section IX, “Welding, Brazing, and Fusing Qualifications,” details the requirements for specifying and qualifying a welding procedure and for testing and qualifying welders. ASME B&PV Code Section V, “Nondestructive Examination,” describes the required qualifications for NDE examiners and the requirements and methods for performing NDE.

Review the relevant articles in ASME B&PV Code Subsection NB-2000 to verify that the applicant specified the appropriate testing requirements for materials of construction of the confinement boundary. In addition, verify that the confinement boundary welds are full-penetration welds, constructed in accordance with ASME B&PV Code Subsection NB-4240 requirements, with the following exception: Because of the difficulty with the fabrication of full-penetration welds for some joint geometries, canister top closure welds may be partial-penetration welds. These excepted welds include the shell-to-top cover welds and the welds associated with siphon and vent port covers.

8.5.3.2 *Confinement Weld Inspection*

Inspections are performed to verify the structural integrity of the welded joints. ASME B&PV Code Subsection NB-5200 requires welds to be inspected by both volumetric and surface techniques. Volumetric techniques may include either radiographic (RT) or ultrasonic (UT) testing. Surface techniques may include either liquid penetrant (PT) or magnetic particle testing (MT). Note that magnetic particle testing is applicable only to ferromagnetic materials such as carbon and low-alloy steels. The applicant should examine austenitic and duplex stainless steel canisters by the liquid penetrant method.

For certain welds, progressive surface examinations may be performed during the buildup of the weld in lieu of the post-weld volumetric examination. This exception is permitted when the geometry of the joint or the material prevent effective volumetric examinations. For example, there currently are no approved techniques for the volumetric examination of fillet welds associated with the austenitic stainless steel canister shell-to-lid joint. A progressive surface examination is defined as performing an examination of weld deposit layers at pre-calculated intervals in addition to the surface examination of the root and final weld layers.

8.5.3.2.1 Austenitic Stainless Steel Closure Lid Welds

The progressive, or multipass, surface examinations of austenitic stainless steel structural welds may be used in lieu of the volumetric examination provided that the following conditions are met:

- Structural calculations apply a stress-reduction factor of 0.8 to the allowable design stress to account for imperfections or flaws that may be missed by progressive surface examinations.

- The interval between surface examinations during the buildup of the weld are calculated as follows:

  1. Calculate the critical flaw size (depth) assuming a buried flaw. Postulate a full circumferential (360-degree) flaw. Use the requirements in ASME Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Division 1, IWB 3600, for alternative flaw acceptance criteria. Use of J-integral or net section stress is acceptable. Verify that the analysis is consistent with the expected failure mode. The approach used in ASME B&PV Code Section XI Nonmandatory Appendix C, “Evaluation of Flaws and Piping,” may be reviewed for guidance.

  2. Establish the maximum allowable surface examination interval by using the critical flaw depth calculated in Step 1.

  3. PT the root layer, every intermediate layer established in Step 2, and the final weld layer. It is assumed that the root layer is single pass. If the root layer is multipass, calculate the critical flaw depth (Step 1) to establish the maximum allowable intermediate weld deposit depth inspection interval. Assume a surface connected flaw when calculating the critical flaw depth for a multi-pass root layer.

Regarding criterion (3), verify that, if the applicant desires to use a thicker root pass, the applicant should limit the amount of weld deposit to the ratio of the fracture toughness K values (or J values) for the different flaw types (buried K divided by surface K) multiplied by the maximum depth. This will limit the depth of the root pass to the critical flaw size for a surface connected flaw. Thus, if an applicant desires to use a thicker weld deposit for the root pass, then a limiting flaw size analysis establishes a structural basis.

The staff recognizes that, for stainless steel, K, or even J, is not entirely correct for evaluating failure in austenitic stainless steel due to the large capacity for plastic deformation. Generally the result is failure due to net section stress, not fracture. However, the stress intensity ratio suggested above is acceptable for this purpose.

Evaluate the applicant's analysis of the critical flaw size using the above methodology based on service temperature, dynamic fracture toughness, and critical design stress parameters as specified in ASME Section XI, Division 1.

8.5.3.2.2 Duplex Stainless Steel Closure Lid Welds

The progressive, or multipass, surface examinations of duplex stainless steel structural welds may be used in lieu of the volumetric examination provided that the following conditions are met:

- Structural calculations apply a stress-reduction factor of 0.8 to the allowable design stress to account for imperfections or flaws that may be missed by progressive surface examinations.

- The interval between surface examinations during the buildup of the weld are calculated using the critical flaw size as described in Section 8.5.3.2.1 above.

Verify that the applicant included specific qualification testing and acceptance criteria for duplex stainless steel welds that are consistent with the assessment of the critical flaw size. For example ASTM A923-14, "Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels," may be used to define acceptance criteria for impact toughness testing of base metal, welds, and weld heat affected zones.

8.5.3.2.3 Carbon and Low-Alloy Steel Closure Lid Welds

Verify that UT examination of the structural lid weld is in accordance with the ASME Section III, Division 1, Subsection NB-5000 requirements and acceptance criteria.

Note that the NRC can approve progressive surface examinations utilizing a PT or MT on a case-by-case basis only if unusual design and loading conditions exist. For progressive PT or MT without a volumetric NDE of the closure lid welds, a stress-reduction factor of 0.8 is imposed on the weld strength of the closure joint to account for imperfections or flaws that may have been missed by progressive surface examinations. Verify that the applicant has determined an allowable interval between surface examinations during the buildup of the weld using an assessment of the critical flaw size, as discussed in Section 8.5.3.2.1.

In addition, also verify that the applicant has considered all the closure lid weld material and technique improvements that accrued from previous DSS design and fabrication experience. For example, refer to the technical evaluation in NRC Confirmatory Action Letter 97-7-001, where instances of cracking of ASTM SA-516 Grade 70 steel welds led to improvements, such as the use of low-hydrogen electrodes, low-carbon equivalent materials, and maintenance of proper preheat and postheat treatments.

8.5.3.3 *Confinement Weld Testing*

The entire confinement boundary should be pressure tested by either hydrostatic or pneumatic methods to the requirements of ASME B&PV Code Section II, Division 1, Subsections NB-6220 or 6300, respectively.

Following the application of the test pressure for the required time, all joints, connections, and regions of high stress, such as regions around openings and thickness transition sections, should be visually examined for leakage. This visual examination shall be performed in accordance with ASME Code requirements and shall be performed at a pressure equal to or greater than the design pressure or three-fourths of the test pressure. This pressure test and visual examination applies to both the canister body constructed at a fabrication facility and the lid-to-shell welds fabricated and closed in the field.

8.5.3.3.1 Pressure Testing

The entire confinement boundary should be pressure tested by either hydrostatic or pneumatic methods to the requirements of ASME B&PV Code Section III, Division 1, Subsections NB-6220 or 6300, respectively.

Following the application of the test pressure for the required time, all joints, connections, and regions of high stress, such as regions around openings and thickness transition sections, should be visually examined for leakage. This visual examination shall be performed in accordance with ASME Code requirements and shall be performed at a pressure equal to or greater than the design pressure or three-fourths of the test pressure. This pressure test and visual examination applies to both the canister body constructed at a fabrication facility and the lid-to-shell welds fabricated and closed in the field.

8.5.3.3.2 Helium Leakage Testing

The applicant should conduct a helium leakage test of the confinement boundary in accordance with ANSI N14.5, “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials,” with an allowed exception discussed below. The leakage test provides reasonable assurance that the confinement body is free of defects that could lead to a leakage rate greater than the allowable design-basis leakage rate specified in the confinement analyses. This ensures that the following conditions are met:

- The helium inerting gas will remain in the canister in sufficient quantity over the license period to protect the fuel assembles and cask or canister internals from the deleterious oxidizing effects of moisture.

- The helium gas heat transfer medium will remain in sufficient quantity over the licensing period to assure that fuel cladding temperatures are controlled at safe levels.

The applicant should test the confinement boundary at the fabrication shop to the extent practicable. Leakage testing of lid-to-shell welds and welds associated with the siphon and vent ports may be tested in the field by the cask user.

The large lid-to-shell confinement boundary field welds of austenitic stainless steel canisters with redundant confinement closures may be excepted from the leakage testing, provided that the following conditions are met:

- The weld is multipass with at least three distinct weld layers. Each layer should be complete across the width of the weld joint and may be composed of one or more adjacent weld beads.
- If only three weld layers comprise the full thickness of the weld, each layer is PT examined.
- For more than three weld layers, not all weld layers need to be PT examined. The maximum weld deposit depth allowed before a PT examination is necessary is based upon flaw-tolerance calculations described in the volumetric examination exception discussion in Section 8.5.3.2.1. Regardless, at least three different weld layers should be examined (e.g., the root pass, a mid-layer, and the cover pass).
- The weld cannot have been executed under conditions where the root pass might have been subjected to pressurization from the helium fill in the canister itself.

The above exception to the leakage testing requirement is not applicable to the siphon and vent port covers. It is assumed that mechanical closure devices (e.g., a valve or quick-disconnect) permit helium leaks. Consequently, welds potentially subjected to helium pressure by way of leakage through a mechanical closure device should be subsequently helium leak tested.

8.5.3.3.3 Leakage Testing Review Examples

The redundant weld requirement for the confinement system closure creates two closure boundaries. Verify that at least one of the redundant boundaries is helium leakage tested, or that some closure welds are leakage tested and the remaining closure welds of the same boundary designed so that the above leakage test exception criteria are met. Only a boundary that is testable or excluded from testing per this guidance should be considered the confinement boundary of the redundant closures. The application of these criteria to two currently approved designs is provided here.

Leakage Testing of a Single Lid with Cover Plate Design (Figure 8-2)

In Figure 8-2, 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 shell, through the large, multipass weld joining the canister shell to the combined shield and structural lid. The boundary continues through the lid to the small weld joining the lid to the vent-and-drain-port closure plate, and back to the lid. For all cases, the remainder of the boundary (and sketch) is assumed to be symmetrical with or similar to the half-sketch portion that is shown.

This boundary demonstrates confinement integrity by means of the large multipass weld leakage exception criteria for the canister shell-to-lid weld and by helium leakage testing of the small vent-and-drain-port closure plate weld. The large, canister shell-to-lid weld is exempted from the helium leak test because it is a multipass weld meeting the flaw tolerance and other appropriate portions of this guidance. Note that this weld is executed before filling the canister with helium (excluding purging and welding gas, as applicable).

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 vent or drain port closure plate is welded into place. Since the vent or drain port closure weld may have been pressurized from the helium fill gas because of assumed leakage from the closure valve, it should be helium leakage tested in accordance with the methods described in ANSI N14.5.

This completes the first closure boundary. Here again, one weld was exempted from the helium leak test by the design criteria, and the other weld was leak tested. This closure boundary demonstrates compliance with regulatory requirements and is consistent with the staff guidance by ensuring at least one of the two redundant closure boundaries is leak tested or exempted from leak testing by conformance with the multipass weld exception guidance.

The second boundary, delineated by line (2), can be traced from the canister shell on the left side of the sketch up through the fillet weld joining the canister shell to the structural lid cover plate. The boundary continues through the cover plate to the fillet weld joining the cover plate to the canister lid. The welds joining the cover plate to the canister shell and lid cannot be helium leak tested since there is no feasible means to do so. However, since the first closure boundary, delineated by line (1), was tested (or exempted through design), the need to helium leak test at least one of the closure boundaries has been satisfied. Since this second boundary does not meet all the criteria for a confinement boundary, it may not be designated as the confinement boundary. The first closure is thereby the confinement boundary in this design, as it meets all the applicable criteria for a confinement boundary.

Leakage Testing a Dual Lid Design (Figure 8-3)

In Figure 8-3, the dotted line marked (1) defines one of the redundant closure boundaries. It may be traced from the canister shell on the left side of the sketch. The boundary proceeds through the partial penetration weld joining the canister wall to the shield lid and into the shield lid. It continues through the small fillet weld joining the vent or drain port cover plate and back through the same fillet weld to the shield lid.

This closure boundary may satisfy the leak test guidance by several methods, depending on details of the weld design. The canister shell-to-shield-lid weld may be designed in several ways. The weld may be a small seal weld, which would necessitate subsequent helium leak testing. Conversely, it could be a large, multipass weld consistent with the leakage test exception guidance described in this chapter. Either way, note that this weld (canister-to-shield-lid weld) is executed before filling and pressurizing the canister with helium (use of purge or backing gas for welding operations is not considered filling or pressurizing).

Next, the canister is drained, dried, purged, and filled with helium to the design operating pressure. The helium line connection is closed off. The vent or drain port cover plate is welded into place. Since this weld may potentially be pressurized from the helium fill gas because of assumed leakage through the closure valve, it should be helium leakage tested.

This completes the first closure boundary. Note that one weld was either helium leakage tested or excepted from the leak test by the design criteria. The other weld was leak tested. Thus, this closure boundary demonstrates compliance with regulatory requirements and is consistent with staff guidance by ensuring at least one of the two redundant closures is leak tested or excepted by conformance to this guidance. This closure may therefore be designated as the confinement boundary.

The secondary boundary, delineated by line (2), can be traced from the canister shell on the left side of the sketch up through the canister shell-to-structural-lid weld and into the structural lid. The weld joining the canister shell and structural lid cannot be helium leakage tested because helium is not present. Note, however, that this weld may comply with the leakage testing exception criteria described in this chapter. In this case, the second closure also qualifies for designation as the confinement boundary.

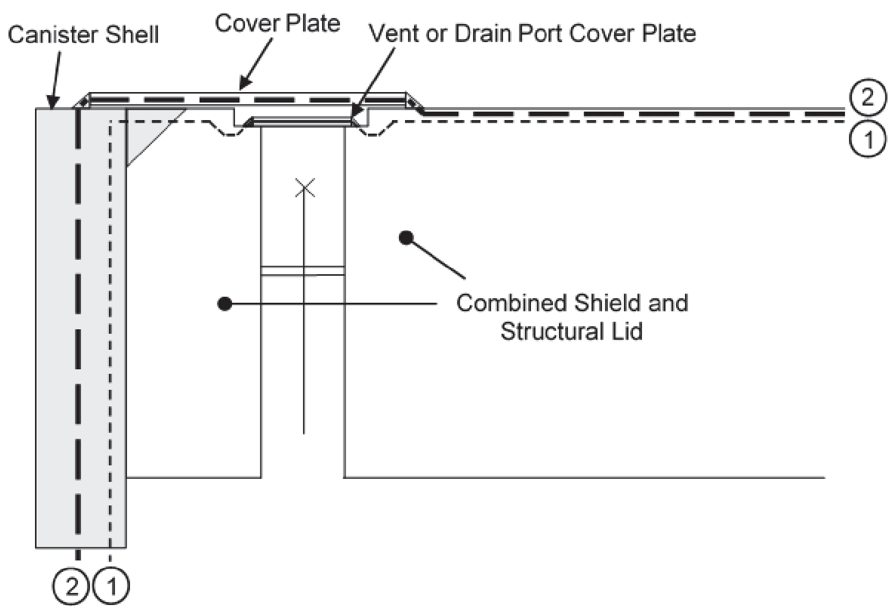

Figure 8-2 Single Lid with Cover Plate Design

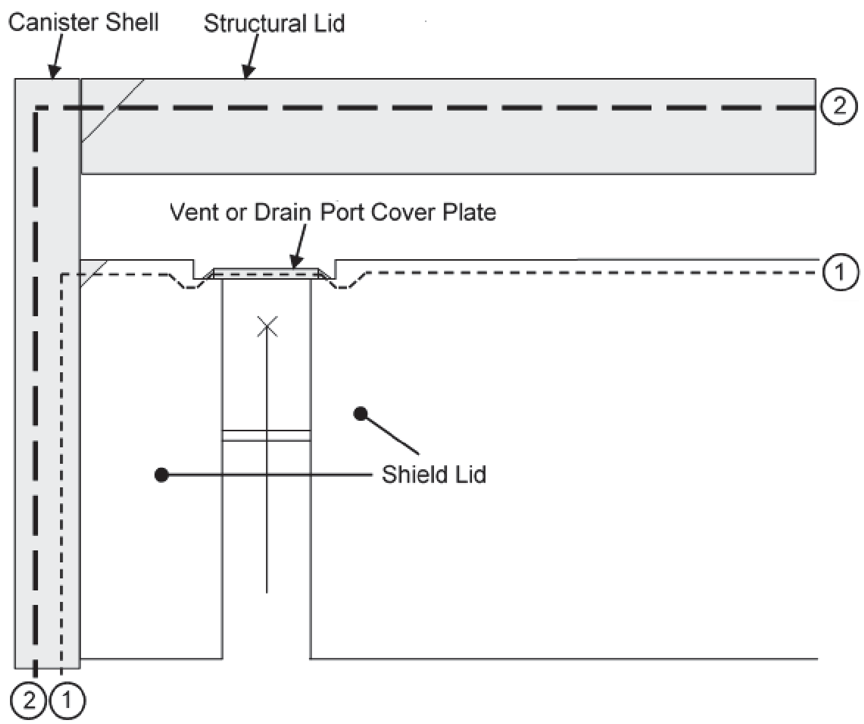

Figure 8-3 Dual Lid Design

For this design in Figure 8-3, the designer therefore has the freedom to designate either of the redundant closures as the confinement boundary. Only one of the two closures is designated as the confinement boundary.

8.5.4 Mechanical Properties of Metals

Assess the acceptability of all material mechanical properties for SSC subcomponents that have a structural role, that is, are relied on for maintaining the analyzed configuration for the stored SNF, HLW or reactor-related GTCC waste (e.g., canister, cask, basket, overpack), excluding the SNF subcomponents. Ensure that the mechanical properties used in the structural evaluation are adequate upon consideration of the environmental and operating conditions during the requested license or storage term (e.g., 40 years), including loading, transfer, storage, and retrieval operations.

8.5.4.1 *Tensile Properties*

Verify that the SAR clearly references acceptable sources of all material properties. Acceptable material properties, allowable stresses, temperature limits, and other requirements include those provided in ASME B&PV Code Section II, Part A, “Ferrous Metals;” Part B, “Nonferrous Metals;” Part C, “Welding Rods, Electrodes, and Filler Metals;” and Part D, “Properties.” The use of certified material test reports for defining mechanical properties is generally not permissible. These property values may be nonconservative because samples may be taken at a portion of the ingot, billet, or forging that have optimum materials properties during certification. Coordinate with the structural reviewer (SRP Chapter 4, “Structural Evaluation”) if the applicant selected inadequate material properties.

Confirm that the SAR and SSC drawings identify the design criteria (codes, standards, specifications) for SSC subcomponents providing structural support. Verify that material mechanical properties used in the structural evaluation are consistent with the design criteria. For example, if an SSC subcomponent is designed to a particular subsection of ASME B&PV Code Section III, the material properties and requirements for the given SSC should be consistent with those allowed by that subsection.

The application may contain a tabulated list of all materials used for SSC subcomponents providing structural support and the proposed service conditions during loading, transfer, storage, retrieval, and waste management operations. The tables may list the subcomponent name, safety classification, intended safety function, fabrication specification (i.e., grade, type and class of material), and material property values (e.g., elastic modulus, yield strength, tensile strength) assumed in the structural evaluation. This information may also be found in the design drawings and multiple tables, as applicable, across the SAR. Evaluate the assumed property values upon consideration of the thermal, radiation, or other applicable environments that may impact structural performance.

8.5.4.2 *Fracture Resistance*

The reviewer should be familiar with ASME Section III NB-2300, "Fracture Toughness Requirements for Material," when evaluating a new DSS or new material for an SSC. Metals having a face-centered, cubic-crystal structure, such as austenitic stainless steels, remain tough and ductile to very low temperatures and are not a concern in this regard. Note that ASME Section III NB-2311(a)(7) includes nonferrous material as material for which impact testing is not required. This notation only applies to nonferrous materials that are included in ASME Section II, Tables 2A and 2B. For some DSS designs, SSCs not part of the confinement boundary use materials that are not included in ASME Section II Tables 2A and 2B. In these cases, determine if fracture toughness testing of these materials is necessary. Review the materials that provide a structural function to determine adequate resistance to fracture.

The reviewer should verify that calculated values of fracture toughness using correlation equations based on impact toughness data such as Charpy V-notch toughness are appropriate for the materials considered. Numerous correlations have been developed for pressure vessel steels and other specific alloys (Roberts and Newton 1981). Ensure that the applicant justified the use of a correlation equation that was not developed for the alloy system used in a DSS SSC that has a structural function.

Because embrittlement of metals may occur under exposure to neutron radiation, the NRC staff calculated the maximum potential accumulated neutron fluence on DSS components, considering components most directly exposed to the radiation source (middle of the fuel basket) and assuming fuel is loaded immediately after it is removed from the reactor vessel and stored for 100 years. To further provide a bounding estimate, the staff assumed a cask design that uses 40 Westinghouse 17 x 17 pressurized-water reactor (PWR) fuel assembles with an average burnup of 70 gigawatt days per metric ton of uranium (GWd/MTU) and 4.0 fuel enrichment. The staff calculated the neutron source term for neutrons with energy at or greater than 1 million (mega) electron volts (MeV) using the Origen/Arp computer code of the SCALE 6.1 computer code system. At this location, the total accumulated neutron fluence after 100 years of storage was calculated to be 2.63×10^{16} neutrons per square centimeter (1.70×10^{17} neutrons per square inch). This value is several orders of magnitude lower than fluence levels known to affect the mechanical properties of steel (Nikolaev et al. 2002; Odette and Lucas 2001), stainless steel

(Gamble 2006; Caskey et al. 1990), and aluminum (Farrell and King 1973; Alexander 1999). As a result, there is no need to consider the effects of irradiation on the fracture resistance of these metals.

8.5.4.2.1 Ferritic Steels

Several types of ferritic steels may become brittle at low service temperatures. ASME B&PV Code, Section III, contains requirements for material fracture toughness; however, these requirements were developed for reactor components and do not address hypothetical accident conditions for storage systems (e.g., impacts at low temperatures). Therefore, in the evaluation of ferritic steels, refer to the guidance for fracture toughness criteria and test methods described in RG 7.11, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches,” and RG 7.12, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater Than 4 Inches, But Not Exceeding 12 Inches.”

RG 7.11 and RG 7.12 specify the types of tests and data needed to qualify a material for designs that specify ferritic steels other than those listed. Those tests and data include dynamic fracture toughness and nil-ductility or fracture appearance transition temperature test data. Toughness testing (e.g., Charpy impact) of welds is governed by the ASME B&PV Code, Section III, as supported by Section IX.

8.5.4.2.2 Duplex Stainless Steels

Duplex stainless steel has both ferritic and austenitic phases and are susceptible to phase instability that may affect fracture toughness. Verify that the applicant included specific qualification testing and acceptance criteria for duplex stainless steel welds that are consistent with the assessment of the critical flaw size. For example, ASTM A923-14, “Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels,” may be used to define acceptance criteria for impact toughness testing of base metal, welds, and weld heat affected zones.

8.5.4.3 *Performance of Aluminum Components*

Storage container basket assemblies use aluminum alloys, aluminum-based metal matrix composites (MMCs), and laminates consisting of aluminum and boron carbides (e.g., Boral™) and are particularly susceptible to property changes at elevated temperatures. Thus, use the detailed guidance below to verify that the DSS or DSF design uses appropriate aluminum properties, considering all service temperatures.

8.5.4.3.1 Tensile Properties of Aluminum

The reviewer should verify that the applicant considered appropriate tensile properties for storage container basket components with a structural function and manufactured from aluminum alloys. There are a variety of aluminum alloys that have been used in basket construction. Some aluminum alloys, such as Al 6061, are precipitation-hardened and are commercially available in several tempers with significantly different yield and tensile strengths and ductility values. Al 6061 has magnesium and silicon as its major alloying elements and is available in pre-tempered grades such as annealed 6061-O and tempered grades such as 6061-T6 and 6061-T651. Good combinations of strength and ductility are obtained in Al 6061 by heat treating it to induce a finely distributed precipitate of magnesium silicide phase. The T6 condition consists of annealing at

532 degrees Celsius (°C) (990 degrees Fahrenheit (°F)) for 1 hour, quenching in water to room temperature, then aging (tempering) at 160 °C (320 °F) for 18 hours to precipitate the magnesium silicide phase.

Elevated temperatures can affect the properties of Al 6061. Temperature affects the allowable stress for all tempers including T4, T451, T6, and T651, but especially for the T6 and the T651 tempers. Aging at higher temperature or holding at higher temperature after aging at 160 °C (320 °F) will coarsen the magnesium silicide precipitates and correspondingly reduce the strength of the alloy (Farrell 1995). Verify that the mechanical properties used for precipitation-hardened aluminum alloys for structural components exposed to elevated temperatures account for the microstructural changes that affect yield and tensile strength. Note that ASME Section II, Part D, Table 1B, requires the use of time-dependent properties for precipitation-hardened Al 6061 at temperatures at or above 177 °C (350 °F).

8.5.4.3.2 Fracture Resistance of Aluminum

The fracture toughness of traditional aluminum alloys varies widely and is dependent on composition and alloy condition for heat-treated or precipitation-hardened aluminum alloys. Compare the applicant's reported value of fracture toughness to tabulated values in materials handbooks and peer-reviewed publications as necessary (e.g., ASM Metals Handbook Desk Edition; Kaufman et al. 1971).

The fracture toughness of aluminum MMCs depend on many factors including (1) particle composition, (2) particle size, (3) particle loading, (4) particle distribution or clustering, (5) alloy composition, and (6) alloy condition for aluminum alloys that can be age hardened. The fracture toughness of aluminum MMC has been found to range from 8 to 30 ksi·in$^{1/2}$ (Flom et al. 1989; Flom and Arsenault 1989; Lewandowski 2000; Miserez 2003; Rabiei et al. 2008). Verify that the applicant has assessed the fracture resistance of aluminum MMCs for structural applications using valid fracture toughness data.

8.5.4.3.3 Creep Behavior of Aluminum

More recent storage system designs have specified ever higher design temperatures for the fuel basket components in order to accommodate higher loading densities and higher burnup fuel. This trend has pushed the various aluminum components well into creep regime operating temperatures.

Review the design maximum temperatures and stresses for aluminum components and verify that the applicant has performed a creep analysis if any structural load-bearing aluminum components operate at a design temperature above approximately 93 °C (200 °F). In the event temperatures exceed the ASME B&PV Code, Section II, nominal 204 °C (400 °F) temperature limit for aluminum, other sources for creep data should be examined. One previously cited reference for this information is Wilson et al. (1969); however, the reviewer should recognize that designs evaluation through the time of this writing have had design stresses (on the order of tens of pounds per square inch) that were substantially below the creep-rupture stresses provided in the referenced report. Nevertheless, ensure that the design calculations include an assessment of creep deformation over the storage period.

Borated aluminum neutron poison materials should be considered on a case-by-case basis if they are subjected to structural load bearing beyond their own dead-weight loads. These materials have inherently low ductility and generally unknown creep properties.

8.5.5 Thermal Properties

Coordinate with the thermal reviewer (SRP Chapter 5, "Thermal Evaluation") to determine the properties of the materials important to the thermal analysis. Verify the material compositions and thermal properties, such as thermal conductivity, thermal expansion, specific heat, and heat capacity, as a function of the temperature over the range in which the components are to operate. Verify that the applicant has evaluated the potential change in these material properties due to material degradation over their service life. Temperature and anisotropic dependencies of thermal properties should be considered.

8.5.6 Radiation Shielding Materials

8.5.6.1 *Neutron Shielding Materials*

Boron-filled polymers are often used for neutron shielding materials. Dose limits are calculated at the site or controlled area boundary, as applicable, and not the canister surface; therefore, these materials are considered important to safety.

Ensure that the SAR describes the composition and geometries of shielding materials. Ensure that the SAR includes references for all materials used, including nonstandard materials (e.g., proprietary neutron shield material), for the source of the material composition and density data along with validation of the data.

In-service performance monitoring of these materials typically is conducted during periodic radiation surveys. Should a decline in the shielding effectiveness be detected, the staff expects that there will be enough time and opportunity for engineering evaluation and corrective action. Therefore, the qualification and acceptance testing of neutron shielding materials are not expected to be included in the technical specifications.

8.5.6.2 *Assessing Previously Unreviewed (New) Neutron Shielding Materials*

Confirm that temperature-sensitive (e.g., polymeric) neutron shielding materials will not be subject to temperatures at or above their design limits during normal conditions. Determine whether the applicant properly examined the potential for shielding material to experience changes in material densities at temperature extremes. For example, elevated temperatures may reduce hydrogen content through loss of water in concrete or other hydrogenous shielding materials.

With respect to polymeric neutron shields, verify that the SAR describes the following:

- the test(s) demonstrating the neutron-absorbing ability of the shield material
- the testing program, data, and evaluations that demonstrate the thermal stability of the resin over its design life while at the upper end of the design temperature range
- the nature of any temperature-induced degradation and its effect(s) on neutron shield performance
- what provisions exist in the neutron shield design to assure that excessive neutron streaming will not occur as a result of shrinkage under conditions of extreme cold; this description is required because polymers generally have a relatively large coefficient of thermal expansion when compared to metals

- any changes or substitutions made to the shield material formulation; for such changes, describes how they were tested and how that data correlated with the original test data regarding neutron absorption, thermal stability, and handling properties during mixing and pouring or casting

- the acceptance tests conducted to verify that any filled channels used on production storage containers did not have significant voids or defects that could lead to greater-than-calculated dose rates (see SRP Section 12.5.2.4, “Shielding Tests”)

- the material’s ability to withstand the effects of heat and irradiation (e.g., the possibility of heat and radiation altering polymer structures to reduce ductility and fracture toughness, and also creating gaseous products such as hydrogen)

Confirm that the SAR describes the potential for shielding materials to experience changes in material properties at temperature extremes and accumulated radiation exposure.

8.5.6.3 *Gamma Shielding Materials*

Concrete, steel, cast iron, uranium, and lead typically serve as gamma radiation shields. Collaborate with the shielding reviewer (SRP Chapter 4) to ensure that the material compositions and densities used in the shielding models are consistent with the design features described in the SAR. The shielding properties should account for manufacturing tolerances and expected degradation from corrosion reactions, elevated temperature, and accumulated radiation exposure.

Confirm that the SAR describes the physical dimensions of shielding materials, including seams, penetrations, or voids. For example, lead shielding may be put into place as stacked bricks or plates, and lead wool is occasionally used to fill gaps. Ensure that manufacturing controls are in place to address any potential paths for gamma streaming.

8.5.7 Criticality Control Materials

Various materials containing boron are used in the nuclear industry as neutron absorbers for criticality control. Neutron absorbers can consist of alloys of boron compounds with aluminum or steel in the form of sheets, plates, rods, liners, and pellets. Likewise, neutron absorbers can consist of a core containing mixed aluminum and boron carbide particles, clad on both sides with aluminum (a composite).

8.5.7.1 *Neutron Absorbing (Poison) Material Specification*

The neutron absorber material must be demonstrated to be adequately durable for the service conditions of the application (10 CFR 72.124(b)). The materials should have excellent physical and chemical stability, including a high resistance to radiation and corrosion. Further, these materials should experience no reduction in effectiveness under normal, off-normal, and accident conditions. These assurances are usually obtained during qualification testing of the material. In addition, acceptance tests (SRP Chapter 12, “Conduct of Operations Evaluation”) are performed on samples from each production run of the material. This procedure will ensure that the properties for the plates or other shapes produced are in compliance with the specifications and requirements of the application. The uniformity of the distribution of boron-10 may be addressed in both the qualification and the acceptance tests.

For all boron-containing absorber materials, verify that the SAR and its supporting documentation describe the absorber material's chemical composition, physical and mechanical properties, fabrication process, and minimum poison content. If the applicant intends to use an absorber material with a specific trade name, verify that the manufacturer's data sheet is submitted to supplement the above information. In the case of absorber plates or sheets, the SAR should specify the minimum poison content as an areal density (e.g., milligrams of boron-10 per cm^2).

Qualification testing of neutron absorber (poison) materials is conducted to ensure the following:

- The material used will have sufficient durability for the application for which it has been designed.

- The physical characteristics of the components of the absorber materials will meet the design requirements, and the uniformity of the distribution of boron-10 is sufficient to meet the requirements of the applications for which the absorber materials will be used. Materials that have passed the qualification tests should be acceptance tested (see SRP Chapter 12) for use in systems to be employed in the storage or transport of nuclear fuel.

ASTM C1671-15, "Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging," with some exceptions, additions, and clarifications, is considered appropriate for staff use in its review activities. Appendix 8A, "Clarifications, Guidance, and Exceptions to ASTM Standard Practice C1671-15," to this SRP provides the exceptions, additions, and clarifications to this standard. The use of ASTM C1671 is not a regulatory requirement; alternative approaches are acceptable if technically supported.

8.5.7.2 *Computation of Percent Credit for Boron-Based Neutron Absorbers*

This section illustrates one method used by the materials reviewer to compute the level of credit to be allowed for 1/v neutron absorber materials[1] in the criticality safety analysis of packages for storing fissile materials, including fresh nuclear fuel and SNF. The allowed level of credit uses the results of neutron attenuation measurements performed on samples of the absorber material placed in a beam of thermal neutrons.

The staff has accepted an upper limit of 90-percent credit to be applied to boron-based solid absorbers, meaning that the material is computationally modeled as containing only 90 percent of the boron-10 shown to be present. The staff has concluded that limiting the poison credit to 90 percent adequately accounts for the uncertainties arising in extrapolating the validation for boron-based absorber materials.

Neutron channeling has been shown to occur in a commercial product that uses coarse particles of B_4C dispersed in an aluminum matrix. The nonuniformities and channeling effects for heterogeneous absorber materials further limit the poison credit to levels below 90 percent. For

[1] Involves that region at the low end of the neutron energy spectrum where neutron absorption is inversely proportional to particle velocity.

heterogeneous absorber materials, the reviewer should verify the applicant's value for poison credit using the following definitions and equations:

A_a = manufacturer's acceptance value of neutron absorber density based on neutron attenuation measurements,

T = lower tolerance limit of neutron absorber density as calculated in ASTM C1671-15.

The value of A_a should be based on a qualified homogeneous absorber standard such as zirconium diboride, or a heterogeneous calibration standard that is traceable to nationally recognized standards, or calibrated with a monoenergetic neutron beam to the known cross section of boron-10. Calibration standards should be evaluated at 111 percent (i.e., 1/0.90) of the poison density assumed in the criticality computational model.

Thus, in addition to the 90-percent limit on poison credit that is used to offset validation uncertainties for all absorbers, the additional penalty for heterogeneous absorbers should be calculated as follows:

If $T \geq A_a$, then 90-percent credit is given.

If $T < A_a$, then compute the fractional credit from 0.75 to 0.90 as follows:

Fractional Credit = $0.30 + 0.6(T / A_a)$.

If the fractional credit is less than 0.75, the absorber is regarded as unsuitable and should be given no credit.

Other remedies beyond the scope of this guidance may be necessary in addressing the potentially more complex neutron-spectral effects and validation uncertainties encountered with materials based on non-1/v-absorbers such as cadmium or gadolinium. The current guidance applies only to 1/v absorbers such as boron or lithium.

8.5.7.3 *Qualifying the Neutron Absorber Material Fabrication Process*

For the qualification of properties not associated with neutron attenuation, in past reviews the staff has accepted the following qualification testing:

1. Mechanical testing to ensure that the neutron poison material is structurally sound, even if the absorber is not used for structural purposes.

   In the past, the staff has accepted ASTM B557, "Standard Test Methods for Tension Testing Wrought and Cast Aluminum- and Magnesium-Alloy Products," for the tensile testing of samples that demonstrated the following:

   – 0.2-percent offset yield strength no less than 1.5 thousand pounds per square inch (ksi)

   – ultimate strength no less than 5.0 ksi

   – elongation no less than 1 percent

Alternatively, the staff has accepted bend tests under ASTM E290, "Standard Test Methods for Bend Testing of Material for Ductility," with a 90-degree bend without failure as the passing criteria.

2. Porosity measurements to ensure that the corrosion resistance (which is directly linked to hydrogen generation in the spent fuel pool) of the neutron poison material is maintained, and that the general structural characteristics of the material are controlled.

   The methodology for porosity is up to the discretion of the applicant. The technical specifications should explicitly state limits on both the total porosity of the material and the "open" or "interconnected" porosity of the material. Excluding Boral™, the total open porosity of the neutron poison material should be limited to 0.5 volume percent or less.

   In general, the conditions of SNF loading, unloading, and storage do not require qualification testing to demonstrate resistance to thermal-, radiation-, or corrosion-induced degradation if the neutron absorber is only made of boron carbide and an aluminum alloy meeting ASTM chemical requirements for the 1000 or 6000 series of aluminum. Other aluminum alloys (particularly those that are not heat-treatable) may also be acceptable to the staff without qualification testing. However, porosity measurements on the neutron poison material should not be waived, regardless of the aluminum alloy used in the neutron absorber.

3. A sufficient number of samples should be used to measure the thermal conductivity of the neutron poison material at room and elevated temperature. Note that clad neutron poison materials are thermally anisotropic.

4. For clad materials, the qualifying tests should include a test demonstrating resistance to blistering during the drying process. In the past, the staff has accepted testing where samples of clad materials are soaked in either pure or borated water for 24 hours and then inserted into a preheated oven at approximately 440 °C (825 °F) for a minimum of 24 hours. The samples are then visually inspected for blistering and delamination before undergoing qualifying mechanical testing.

Additional qualifying tests should be conducted for structural neutron poisons. Mechanical and thermal tests should include tensile testing, impact testing (or K_{IC} measurements), creep testing, and (if applicable) mechanical testing of weldments.

Samples of neutron poison material (i.e., the use of transmission electron microscopy or scanning electron microscopy) should be examined for the following changes:

- redistribution or loss of boron
- dimensional changes (material instability)
- cracking, spalling, or debonding of the matrix from the boron-containing particles
- weight changes caused by leaching, dissolution, corrosion, wear, or off-gassing
- embrittlement
- chemical changes such as oxidation or hydriding
- molecular decomposition of the material as a result of radiation (radiolysis)

Coupons should be taken so as to be representative of the neutron poison material. To the extent practical, test locations on coupons should be stratified to minimize errors because of location or position within the coupon. Locations should include the ends, corners, centers, and irregular locations. These locations represent the most likely areas to contain variances in thickness. Adequate numbers of samples should be taken from components (e.g., plate, rod) produced from a lot to obtain a good representation. A lot is defined as all plates from a single billet. Overall, the coupons should be a representative sample of the material.

For containers that will be loaded or unloaded in a SNF pool or similar environment, verify that the applicant has evaluated or tested absorber material for environmental and galvanic interactions and the generation of hydrogen in the pool environment. If environmental testing is employed, the test conditions (time, temperature) should equal or exceed those expected for loading, unloading, and transfer operations. For environmental tests, the absorber materials should be coupled to dissimilar metals, as may be appropriate to the application. The environment may be borated or deionized water, as appropriate. The evaluation should also consider the effects of any residual pool water remaining in the container after removal from the pool. Generally, for common engineering materials, an evaluation based upon consultation of a corrosion reference (galvanic series) should suffice for pool loading and unloading situations.

The applicant should take appropriate measures to assess the strength or ductility of the material, depending on the structural requirements of the application.

Acceptance testing of the fabricated materials is discussed in Chapter 12 of this SRP.

8.5.8 Concrete and Reinforcing Steel

8.5.8.1 *Embedment Materials*

The reviewer should evaluate the material to be used for embedments, inserts, conduits, pipes, or other items embedded in the concrete. Embedments should satisfy the requirements of the code used in designing the reinforced concrete structure in which they are embedded (e.g., ACI 359, “Code for Concrete Reactor Vessels and Containments,” ACI 349, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” or ACI 318, “Building Code Requirements for Structural Plain Concrete and Commentary”). Zinc, zinc-rich coatings, zinc-clad materials, and aluminum should not be used for any embedded objects in structures designed to ACI 349 or ACI 359 that will be in contact with wet concrete because of the potential for concrete degradation from an adverse chemical reaction. Embedments and attachments are considered to include components cast or grouted into the reinforced concrete structure, inserts, embedded pipes, conduits, or lightning protection and grounding systems.

Unless otherwise specified in this SRP, steel structural attachments should comply with the appropriate requirements in ACI 349.

8.5.8.2 *Concrete Design and Temperature Limits*

The NRC accepts the use of ACI 318 for the design and material specifications for reinforced concrete structures, although such structures typically are not important to safety. If ACI 349 is used for the design of such structures, the NRC accepts the use of ACI 318 for construction. The NRC also accepts the following criteria as an alternative to the temperature requirements of ACI 349, but only for the specified use and temperature ranges:

1. If concrete temperatures in general or local areas are a maximum of 93 °C (200 °F) in normal conditions, off-normal conditions, or occurrences, no tests are needed to prove capability for elevated temperatures or reduced concrete strength.

2. If concrete temperatures in general or local areas exceed 93 °C (200 °F) but are less than 149 °C (300 °F), no tests are required to prove capability for elevated temperatures or reduced concrete strength if Type II cement is used and temperature-appropriate aggregates are used. The following criteria for fine and coarse aggregates are acceptable:

   – Satisfy the requirements in ASTM C33, “Standard Specification for Concrete Aggregates,” and requirements references in ACI 349 for aggregates.

   – Have a demonstrated coefficient of thermal expansion (tangent in temperature range of 20–38 °C (70–100 °F) no greater than 11×10^{-6} millimeter (mm)/mm/°C (6×10^{-6} inches (in.)/in./°F), or be one of the following materials: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.

3. If concrete temperatures in general or local areas under normal or off-normal conditions do not exceed 107 °C (225 °F), the criteria 1 and 2 (above) apply to the coarse aggregate. Fine aggregate that meets 1 (above) and is also composed of quartz sands or sandstone sands may be used in place of 2 (above) and satisfy the criteria.

The strength and modulus of elasticity of concrete increase as it ages for about the first 10 years after fabrication (Washa et al. 1989; NRC 1996b). For example, for a normal weight concrete typical of that used in the construction of storage pads, strength has been shown to increase by 67 percent relative to the recorded 28-day strength. For drop and tipover events, such increases in concrete pad hardness can result in more severe accelerations on storage system components. The reviewer should ensure that changes in concrete properties with time are considered in structural calculations that evaluate the capability of SSCs to withstand design-basis accidents.

8.5.8.3 *Omission of Reinforcement*

Frequently, designers specify the omission of reinforcing steel (“rebar”) in concrete aboveground structures that have the purpose of gamma shielding only. This is acceptable since it is to avoid the inadvertent formation of voids in the concrete because of the presence of the rebar, which can act to block the aggregate in the concrete from filling all intended areas.

Concrete applied around buried steel structures should be reinforced to alleviate shrinkage crack propagation. Concrete alleviates soil corrosion by creating a beneficial chemical buffering effect (high pH) around the steel, except in environments with high chloride concentrations. Cracks allow ground water plus electrolyte intrusion, which reduces the effectiveness of the concrete protective barrier.

8.5.8.4 *Radiation Damage*

Radiation effects on concrete properties depend on the gamma and neutron radiation doses, temperature, and exposure period. Gamma radiation can decompose and evaporate water in concrete (Bouniol and Aspart 1998). Because most of the water is contained in the cement paste, the effect of gamma radiation on cement paste is more significant than on the aggregates. Gamma radiation can also decompose the silicon monoxide bond within calcium silicate hydrate (Kontani et al. 2010). Neutron radiation deteriorates concrete by reducing stiffness, forming cracks by swelling, and changing the microstructure of the aggregates. This, consequently, reduces concrete strength (Kontani et al. 2010). The changes in aggregate microstructure also can lead to higher reactivity of aggregates to certain aggressive chemicals.

NUREG/CR-7171, “A Review of the Effects of Radiation on Microstructure and Properties of Concretes Used in Nuclear Power Plants,” provides a comprehensive review of the effects of gamma and neutron radiation on the microstructure and properties of concrete used in nuclear power plants. Concrete structures have been regarded as being sound as long as the cumulative radiation does not exceed critical levels over the life of the structure. In general, the critical radiation levels to reduce concrete strength and elastic modulus are considered to be approximately 1×10^{19} n/cm^2 (6.5×10^{19} n/in^2) for fast neutrons (neutron energy greater than 1 MeV) and $1^{-2} \times 10^{10}$ rad ($1^{-2} \times 10^{8}$ grays) for gamma rays (Hilsdorf et al. 1978; EPRI 2012; IAEA 1998; ASME B&PV Code).

As discussed in Section 8.5.4.2 above, the maximum accumulated neutron fluence for any storage system SSC was estimated by the staff to be 2.63×10^{16} n/cm^2 (1.70×10^{17} n/in^2) after 100 years of storage, which is three orders of magnitude below the level that would lead to a reduction of concrete strength and elastic modulus. The gamma dose is also expected to be several orders of magnitude less than the limits defined in the above references, per the specific DSS design bases.

Review the radiation damage analyses for concrete structures to determine that the critical radiation levels discussed above will not be exceeded during dry storage operations.

8.5.9 Bolt Applications

If threaded fasteners are employed for SSCs important to safety, verify that the bolt material(s) have adequate resistance to corrosion and brittle fracture and a coefficient of thermal expansion similar to the materials being bolted together. Also, verify that the fasteners have adequate creep resistance under expected service conditions.

For pressure-retaining and confinement boundary bolts, verify that the applicant has followed the requirements of ASME B&PV Code, Section III, NB-3230, “Stress Limits For Bolts,” and has used the mechanical properties, temperature limits and design stress intensity limits listed in ASME B&PV Code, Section II, Part D, Table 4, “Section III, Classes 1, TC, and SC; and Section VIII, Division 2, Design Stress Intensity Values S_m For Bolting Materials.” Generic guidance on closure bolts for transportation canisters is available in NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks”; however, ASME B&PV Code Section III, NB-3230 is preferred for bolt materials included in ASME B&PV Code Section II, Part D, Table 4.

Coordinate with the structural reviewer (SRP Chapter 4), who has the responsibility to verify that closure bolt stresses are within allowable limits.

8.5.10 Seals

Applicants for SNF storage canisters with metallic seals generally rely on data from the seal manufacturer to determine the maximum service temperatures for seals. Seals that may potentially be exposed to high temperature may not have been tested by independent laboratories (such as the National Institute of Standards and Technology and Factory Mutual). Because of the importance of seal integrity, ensure that the SAR includes laboratory test results using qualified procedures or data sheets that reference such test results.

8.5.10.1 *Metallic Seals*

Bolted lid canisters employ redundant metallic seals as part of the confinement boundary. These seals are SSCs important to safety. The primary materials issue is the temperature resistance of the seal spring material. Generally, this is a nickel-base alloy with excellent temperature and creep resistance. Verify that the metallic seal spring is constructed of a material that will not creep to an extent that may degrade its sealing performance. The seal cover material may be soft aluminum or silver. Aluminum-faced seals have failed in service because of corrosion from inadvertent rainwater intrusion (see NRC Information Notice 2013-07, "Premature Degradation of Spent Fuel Storage Cask Structures and Components from Environmental Moisture," dated April 16, 2013). Substitution of silver alloy-faced seals appears to have alleviated the susceptibility of mechanical seals to this corrosion-induced failure mechanism. If the applicant uses aluminum-faced seals, verify that the design includes provisions to prevent corrosion, such as the use of weather covers.

8.5.10.2 *Elastomeric Seals*

Bolted lid canister designs may also employ a weather cover to preclude rainwater from the confinement boundary seals. These weather covers may be sealed against the weather with an elastomeric seal such as Viton. As such, these seals may be susceptible to thermal- and radiation-induced aging (hardening). Consequently, a replacement program may be warranted if the heat or radiation exposure is sufficient. The seal manufacturer can generally provide guidance as to radiation or thermal resistance. Elastomeric seals have never been SSCs important to safety in storage canisters.

Radiation generally causes polymerization of elastomers to an extent that would adversely affect the performance when the dose reaches 10^5 grays (10^7 rads). For higher-dose rate environments, elastomer O-rings should not be specified. The use of fluorocarbons, which are known to be particularly susceptible to radiation damage, should be restricted if the dose is expected to exceed 100 grays (10^4 rads).

The reviewer should verify that O-ring seals do not reach their maximum operating temperature limit during normal and off-normal conditions of storage. Ensure that the SAR includes the O-ring manufacturer's data sheets specifying temperature and radiation tolerances. The applicant's evaluation should demonstrate that the minimum normal operating temperature (usually -40 °C (-40 °F)) will neither fail the O-ring seal by brittle fracture nor stiffen the O-ring (lose elasticity) to an extent that prevents the seal from meeting its service requirements.

Verify that, under the environmental conditions expected in storage service, O-ring seals will not chemically react or decompose in a manner that would significantly affect other components of the DSS.

8.5.11 Corrosion Resistance

The corrosion rates of engineering alloys are dependent on a number of factors including humidity, time of wetness, atmospheric contaminants, and oxidizing species (Fontana 1986). Consider the range of environmental conditions that are encountered for the DSS and DSF SSCs. For example, storage containers may be exposed to a variety of environments associated with fuel loading, canister closure, fuel drying, container transfer, and storage.

The following sections address specific considerations for commonly used engineering alloys for SSCs important to safety that may be exposed to environments where the effects of corrosion should be considered. In addition to material selection, other corrosion-control measures may be employed, provided adequate documentation is supplied to demonstrate efficacy. For example, coatings may be specified to alleviate the coastal atmospheric corrosion issue. However, unless supporting data are available to demonstrate the predicted coating life, the coating should be periodically inspected and maintained. Verify that any coating that is relied upon for corrosion resistance is screened in as important to safety. See Section 8.5.12 below for additional guidance on coatings.

8.5.11.1 *Environments*

Materials within the SNF container interior will be in an environment that contains very little water and is backfilled with helium to provide heat transfer and maintain a nonoxidizing environment for the fuel cladding and canister internals. Contaminants, such as chloride and sulfur species, can significantly accelerate general corrosion rates of engineering alloys. However, these species are strictly controlled in operating reactor coolant (EPRI 2000, 2007) and thus are not expected to be present in any residual moisture remaining inside the storage container after drying. Evacuating the canister or cask under vacuum and backfilling with an inert gas such as helium will significantly reduce the water content and humidity inside the canister and also reduce the oxidizing potential of the environment, both of which will significantly decrease the uniform corrosion rate of carbon steel and the potential for localized corrosion of passive alloys such as aluminum alloys and stainless steels.

While operational experience has shown only a few cases of atmospheric degradation of the external surfaces of DSS or DSF SSCs, it should be recognized that inspections have been limited. Generally, the DSS or DSF SSCs are subjected (long term) to a mild atmospheric environment. The range of environmental conditions may be limited and well defined for a site-specific application. For a CoC application, assume that the DSS SSCs may be exposed to a range of atmospheric conditions, including exposures to chloride-containing environments such as marine atmospheres, roadway deicing salt, and cooling-tower effluents. The presence and accumulation of chloride-containing salts can accelerate atmospheric corrosion rates. In addition, consider the effects of temperature fluctuations at the range of possible ISFSI sites when evaluating DSS designs and material selection. Corrosion rates for engineering alloys, including carbon and low-alloy steels, stainless steels, and aluminum alloys in a range of natural and industrial environments, may be found in corrosion references such as “Corrosion Engineering” (Fontana 1986), “Corrosion Data Survey by the National Association of Corrosion Engineers,” (Graver 1985), “Corrosion and Corrosion Control,” (Revie and Uhlig 2008), “Uhlig’s Corrosion Handbook” (Revie 2000), and the ASM Handbook Volume 13, “Corrosion.” Additional information on alloys and materials in specific environments is available in specialized publications such as the ASTM Special Technical Publications series. The National Aeronautics and Space Administration’s Kennedy Space Center Corrosion Technology Laboratory has also issued numerous reports on corrosion of alloys exposed to marine environments as well as testing of coatings to prevent corrosion.

8.5.11.2 *Carbon and Low-Alloy Steels*

For carbon and low-alloy steels that are not in an inert environment or embedded environment such as concrete, verify that the corrosion allowance specified is adequate for the applied term of the license or certificate. Corrosion rates for carbon steel in air may be found in the corrosion references discussed above in Section 8.5.11.1.

In environments such as locations that are in marine atmospheres, near roadways where deicing salts are used, or exposed to cooling-tower effluents, the presence and accumulation of heavy chloride-containing salts can significantly accelerate the normally slight atmospheric corrosion rates to unacceptable values for some storage canister or cask module designs, such as those that employ carbon steel structural elements inside a storage overpack.

To address the increased atmospheric corrosion rates found at coastal marine (salt water) sites, some applicants have specified the use of "weathering steels," such as Cor-Ten, that form a protective layer of corrosion products that reduce additional loss of material. Weathering steels usually contain a minimum of 0.20 percent copper, but they also typically contain small additions of nickel, chromium, and phosphorous (Murata 2000). The Kennedy Space Flight Center has collected data that have demonstrated the benefit of copper-bearing and weathering steels for significantly reducing corrosion at coastal marine sites. Therefore, for coastal marine sites, the use of copper-bearing steels (containing a minimum of 0.20 percent copper) or weathering steels may be necessary. Such steels are covered by ASTM A242, "Standard Specification for High-Strength Low-Alloy Structural Steel," and ASTM A588, "Standard Specification for High-Strength Low-Alloy Structural Steel, up to 50 ksi [345 MPa] Minimum Yield Point, with Atmospheric Corrosion Resistance," supplemental requirements to ASTM A36, "Standard Specification for Carbon Structural Steel," and other specifications.

8.5.11.3 *Austenitic Stainless Steels*

When stainless steel is used for storage containers, the primary concern generally is not corrosion but rather various types of localized corrosion such as pitting or crevice corrosion and stress-corrosion cracking. These corrosion mechanisms are possible in environments that contain chlorides. In the case of dry storage canisters, these corrosion mechanisms may be initiated in environments where airborne chlorides can be transported to the canister surfaces and deliquescence of the deposited chloride-containing salts results in an aqueous film containing chloride ions. Localized corrosion and chloride-induced stress-corrosion cracking (CISCC) of stainless steel components exposed to marine environments have been observed at operating reactors, as documented in the NRC Information Notice 2012-20, "Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters," dated November 14, 2012. However, no occurrence of localized corrosion or CISCC has been observed in the limited inspections of DSSs conducted to date.

NUREG/CR-7170, "Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts," describes laboratory tests for CISCC of austenitic stainless steels and the evaluation of the temperature and relative humidity conditions needed for deliquescence of chloride-containing salts. These tests show that all austenitic stainless steels used for DSS confinement boundaries are susceptible to CISCC, with lower alloy grades, such as 304, more susceptible than the low-carbon, molybdenum-containing 316L. Tests also showed that sensitized material was more susceptible to CISCC than nonsensitized material. The Electrical Power Research Institute (EPRI) conducted a review of the conditions under which CISCC has been observed, evaluated the effects of CISCC, and developed susceptibility assessment criteria for ISFSI locations and welded austenitic stainless steel canisters (EPRI 2013, 2014a, 2014b, 2015).

Based on testing and reviews of operational experience, degradation of austenitic stainless steels as a result of CISCC is expected to be limited to welded structures with tensile residual stresses in environments with elevated airborne chloride concentrations. In addition, CISCC can only occur when the combination of atmospheric conditions and the temperatures of SSCs allow the formation of a chloride ion containing aqueous phase. In most environments, the development of these conditions would take years or decades to develop on the surfaces of DSS or DSF SSCs. Further, the rates of CISCC propagation are limited by a number of factors, including atmospheric conditions and residual stresses. Consequently, CISCC is not a degradation mode that is expected to affect SSCs important to safety constructed from welded austenitic stainless steels in the initial storage period. According to NUREG-1927, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," aging management programs may be needed to address CISCC in periods of extended operation.

8.5.11.4 *Duplex Stainless Steels*

In aggressive environments, where CISCC is more likely to occur, an applicant may specify more corrosion-resistant materials. For the confinement boundary, verify that the materials specified are approved for ASME B&PV Code, Section III, Class 1 construction. Duplex stainless steel UNS S31803 has been approved in ASME B&PV Code Case N-635-1 (for construction of Class 1 components, and the NRC has accepted this code case in RG 1.84. Stainless steel S31803 is a 22-percent chromium, 5-percent nickel stainless steel that has both ferritic and austenitic phases. Duplex S31803 has greater corrosion resistance to pitting, crevice corrosion, and CISCC and has been used in offshore oil production applications where harsh environmental conditions are expected. Note that ASME B&PV Code Case N-635-1 is specific to S31803. A similar duplex stainless steel, S32205, was introduced subsequent to S31803. Duplex S32205 has tighter compositional specification ranges for chromium, molybdenum, and nitrogen. Dual-certified material (i.e., material that meets the requirements of S32205 and S31803) has been produced.

Note that 22-percent chromium, 5-percent nickel duplex stainless steel such as S31803 and S32205 are susceptible to microstructural alteration during welding that can have a significant effect on corrosion resistance (Leonard 2003). Liou et al. (2002) showed that cooling rate and nitrogen content had a marked effect on the austenite to ferrite content. Chen et al. (2002) showed significant decreases in impact energy for S32205 exposed to temperatures in the range of 800 to 950 °C (1,472 to 1,742 °F) for periods of 10 minutes or less, corresponding to 5 percent σ (sigma) phase. Sieurin and Sandstrom (2007) compared time-temperature-transformation curves and critical cooling temperature curves for S32205 duplex stainless steels and concluded that, in order to avoid sigma precipitation and at the same time obtain a sufficient ferrite–austenite phase balance, the cooling rate should be approximately in the range 0.25–50 Kelvin (K)/second. In addition, Sieurin and Sandstrom (2007) stated that, in order to avoid more than 1 percent σ (sigma) phase, the cooling rate from the solution treatment temperature should exceed 0.23K/second, and the aging time must not exceed 134 seconds at the most critical temperature 865 °C (1,590 °F).

As a result of the operational experience with welded duplex stainless steels, the American Petroleum Institute (API) published Technical Report 938-C, “Use of Duplex Stainless Steels in the Oil Refining Industry,” which provides guidance for the acceptance and welding of duplex stainless steels. The API guidance references ASTM A923, “Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels,” which includes specific screening tests, microstructural evaluation methods for detecting detrimental microstructural phases, and impact toughness requirements for base metals and welded duplex stainless steels. Verify that DSS designs that specify duplex stainless steels for the confinement boundary have (1) adequately addressed the unique microstructural considerations associated with these alloys and (2) included specific testing and acceptance criteria to ensure that fabrication and welding of the duplex stainless steel do not result in detrimental microstructural alterations that negatively impact the corrosion resistance or toughness of the alloy.

8.5.12 Protective Coatings

Coatings in DSSs are used primarily as corrosion barriers or to facilitate decontamination. They may have additional roles, such as improving the heat rejection capability by increasing the emissivity of internal components. Coatings typically are not SSCs important to safety. The SSCs to which the coatings are applied are generally important to safety. No coating should be credited for protecting the substrate material or extending the useful life of the substrate material unless a periodic coating inspection and maintenance program is required.

Coatings generally have low safety significance with the exception of coating issues that may result in adverse chemical or galvanic reactions. Typically, the information the applicant provides on coatings is not generally subject to further confirmation as part of the review. However, the applicant may specify unique or innovative coatings to perform a specific function unique to the storage system. In these instances, use discretion in implementing the review guidance in this section.

8.5.12.1 *Review Guidance*

Determine the appropriateness of the coating(s) for the intended application by reviewing the coating specification for each coating that is applied to an SSC important to safety. A specification that describes the scope of the work, required materials, the coating's purpose, and key coating procedures should ensure that appropriate and compatible coatings have been selected by the DSS designers.

8.5.12.2 *Scope of Coating Application*

Ensure that the SAR describes the function of the coating, a list of the components to be coated, and a description of the expected environmental conditions (e.g., expected conditions during loading, unloading, and dry storage).

8.5.12.3 *Coating Selection*

Verify that the coating specification identifies the manufacturer's name and the type of primers and topcoat(s) comprising the coating system. Because of the unique nature of coating properties and coating application techniques, the manufacturer's literature may be the only source of information on the particular coating.

Verify that the coating selected for the storage container components is capable of withstanding the intended service conditions over the design service life. Verify that the coatings will not react with the container internal components and contents and will remain adherent and inert when exposed to the various service environments. The most prevalent, potentially degrading environments include the immersion in borated SNF pool water during loading and unloading operations, and high-temperature and high-radiation environments encountered during vacuum drying and long-term storage. Failures can be prevented by ensuring that the selection and the application of the coating are controlled by adhering to the coating manufacturer's recommendations.

8.5.12.4 *Coating Qualification Testing*

Ensure that the coatings (including paints or plating) used inside a DSS have been tested to demonstrate the coatings performance under all conditions of loading and storage. The conditions evaluated should include exposure to radiation, high temperature during vacuum drying and storage, and immersion during loading, unloading, and transfer operations. The applicant should demonstrate that the coating will remain intact and inert for the full duration of the DSS design life.

There are a number of standardized ASTM tests for coatings performance. In reviewing ASTM (or other) tests used to qualify coatings for service in storage containers, consider the applicability of a test to the service conditions.

Ensure that a qualified coatings engineer (e.g., certified by the National Association of Corrosion Engineers) performed the planning, execution, and interpretation of coating qualification tests. Ensure that the applicant has employed appropriate, qualified expertise for any coatings qualification program. In addition, unless supporting data are available to demonstrate the predicted coating life, the coating should be periodically inspected and maintained.

8.5.13 Content Reactions

Verify that the contents of SNF, reactor-related GTCC waste, and HLW are stable and that there will be no adverse reactions with the container or internal baskets or supports over the storage period (10 CFR 72.120(d) and 10 CFR 72.236(h)).

Verify that the applicant has provided an adequate description of the contents so that the reviewer can fully evaluate its stability and compatibility with the container. Key parameters of the applicant’s description include the physical and chemical form (e.g., activated metal, process waste), the geometric form (e.g., particulates, bulk solid), the maximum quantity of waste to be stored, and the radionuclide inventory.

8.5.13.1 *Flammable and Explosive Reactions*

Verify that the applicant has demonstrated that the contents will not lead to potentially flammable or explosive conditions.

Metallic contents may be subject to pyrophoricity, or auto-ignition, when the content surface area is sufficiently large (e.g., fine particulates) and oxygen or humidity, or both, are present at elevated temperatures. If metallic contents could potentially support pyrophoricity, the applicant should demonstrate that measures are taken to remove moisture or oxygen from the container, such as through vacuum or inerting. The applicant also should consider the potential for content materials, such as polymers, to decompose when exposed to heat and radiation, which may generate the moisture to support pyrophoricity.

In addition, hydrogen or other flammable gases may be generated during wet loading and unloading operations. For example, aluminum used in basket components can react with moisture to generate hydrogen. Efforts to passivate the aluminum components have proven inadequate to eliminate the generation of hydrogen. The use of zinc, zinc-rich coatings, or zinc-clad materials (e.g., galvanized steel) in particular is known to generate potentially large quantities of hydrogen gas during wet loading in SNF pools. In addition, flammable gas may be generated from waste radiolysis, biodegradation, and chemical reaction. Verify that the operating procedures contain measures for detecting the presence of hydrogen and preventing the ignition of combustible gases during cask loading and unloading operations. The technical specifications (SRP Chapter 17) should incorporate these procedures by reference.

Refer to NRC Bulletin 96-04, "Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks," issued July 5, 1996, for information about operational issues associated with hydrogen generation. This bulletin describes a case where a zinc coating on a canister interior reacted with borated SNF pool water to generate hydrogen, which ignited during the canister closure welding. Confirm that the applicant has demonstrated that no such adverse reactions will occur between the canister content materials, fuel payload, and the operating environments (10 CFR 72.120(d) and 10 CFR 72.236(h)).

8.5.13.2 *Corrosion*

Corrosive reactions between the contents and the internal environment, as well as reactions between the contents and the confinement container, may degrade structural integrity and confinement, and also may adversely impact retrievably of the SNF. Ensure that the SAR demonstrates that corrosion wastage will not lead to a loss of intended functions.

Refer to Section 8.5.15.2.3 of this SRP for guidance on the review of spent fuel cladding oxidation. For noncladding hardware components, the staff has previously reviewed a number of hardware components and materials for compliance with 10 CFR 72.120(d) to ensure that there are no significant chemical, galvanic, or other reactions. These stainless steel and zirconium alloy components are various neutron source assemblies, burnable poison rod assemblies, thimble plug devices, and other types of control elements. The staff has found the following materials to be acceptable for storage when the canister is constructed of stainless steel with stainless steel and aluminum basket components:

- Neutron source materials composed of stainless steel or zirconium alloy cladding containing antimony-beryllium, americium-beryllium, plutonium-beryllium, polonium-beryllium, and californium—The NRC assessed the exposure of these various contents to the wet loading and dry storage environment and determined that corrosion would not lead to a loss of intended functions.

- Control elements composed of zircaloy or stainless steel cladding containing boron carbide, borosilicate glass, silver-indium-cadmium alloy, or thorium oxide—The NRC assessed the exposure of these various contents to the wet loading and dry storage environment and determined that corrosion would not lead to a loss of intended functions.

8.5.14 Management of Aging Degradation

8.5.14.1 *Initial Storage Term*

In some cases, materials degradation may challenge the ability of a component to fulfill its intended function for the duration of the storage term. If an applicant cannot demonstrate adequate materials performance, then the SAR should describe maintenance programs (e.g., monitoring, inspections) to address issues associated with materials aging degradation. Some examples of such maintenance activities from previous reviews include the following:

- transfer cask maintenance programs that inspect for corrosion, wear, and loose or damaged fasteners

- coatings inspections, in cases where coatings are credited for preventing corrosion, enhancing heat transfer, or where coating debris could interfere with ventilation pathways

- concrete inspections to identify deterioration and basemat settling

- radiation surveys to monitor neutron shield effectiveness

Ensure that proposed maintenance activities provide for timely identification of materials degradation such that corrective actions can be implemented before a loss of component intended functions. Monitoring and inspection activities should take the following measures:

- use methods that are demonstrated to be capable of evaluating the degradation mechanism

- be performed at a frequency that is sufficient to identify degradation before a loss of component function

- include clear, actionable acceptance criteria

Consider appropriate codes and standards, such as ACI 349.3R for the evaluation of concrete and the ASTM standards on coatings assessment that are endorsed in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants."

Coordinate with the operating procedures reviewers (SRP Chapter 11 and Chapter 12, "Conduct of Operations Evaluation").

8.5.14.2 *Amendment Applications Submitted During a Renewal Review or after a Renewal is Issued*

The NRC may renew a specific license or a CoC for a term not to exceed 40 years, in accordance with 10 CFR 72.42(a) or 10 CFR 72.240(a), respectively. Renewal applications must address aging mechanisms and aging effects that could affect SSCs relied upon for the safe storage of SNF.

NUREG-1927, Revision 1, provides detailed staff guidance for reviewing amendments that are submitted (1) concurrently with a renewal application or (2) after a renewal has been issued. Verify that the following information is included in either the amendment application or the renewal application:

- a scoping evaluation that identifies any new SSCs (and associated subcomponents) included in the amendment request and discusses whether the SSCs are included or excluded from the scope of renewal, following the guidance in Chapter 2 of NUREG-1927, Revision 1

- an aging management review that identifies any applicable aging mechanisms and effects for the new SSCs (and associated subcomponents) within the scope of renewal

- changes to the final SAR, which should include the following:

    - scoping results and identification of any new in-scope SSCs

- revised table of analysis model report results

- identification of previously approved time-limited aging analyses that address the new in-scope SSCs, or identification and a summary of any revised or new time-limited aging analyses that support the amendment

- identification of previously approved aging management programs that encompass the new in-scope SSCs, or a summary of any revised or new aging management programs that will apply to the new in-scope SSCs

For concurrent amendment and renewal applications, if there are different materials reviewers for the renewal review and the amendment review, coordinate across the reviews to ensure that renewal aspects are covered for the amendment.

8.5.15 Spent Fuel

The materials review ensures that the mechanical properties of the cladding materials are adequate to ensure that the SNF remains in the configuration analyzed in the SAR.

The review guidance in this section addresses dry storage of all SNF of burnups the NRC currently licenses for commercial power plant operations. SARs with burnup levels exceeding those licensed by the NRC Office of Nuclear Reactor Regulation (NRR), or for cladding materials not licensed by NRR, may require additional justifications by the applicant.

8.5.15.1 *Spent Fuel Classification*

Verify that the SAR (and, where appropriate, the license or CoC) identifies the allowable SNF contents and condition of the assembly or rods consistent with the definitions in this SRP for intact, undamaged, and damaged fuel (see the Glossary of this SRP).

The reviewer should analyze damaged fuel in terms of the characteristics needed to perform functions to assure compliance with fuel-specific and system-related regulations. A fuel-specific regulation defines a characteristic or performance requirement of the SNF assembly. Examples of such regulations include 10 CFR 72.122(h)(1) and 10 CFR 72.122(l). A system-related regulation defines a performance requirement placed on the fuel so that the DSS can meet its regulatory requirements. Examples of such regulations include 10 CFR 72.122(h)(5) and 10 CFR 72.124(a).

Verify that the applicant considered whether the material properties, and possibly the configuration, of the spent fuel assemblies (SFAs) can be altered during extended irradiation or dry storage. If this alteration is significant enough to prevent the fuel or assembly from performing its intended functions during dry storage, then the SFA should be classified as damaged.

Ensure that the SAR discusses the following to support that the SNF (rods, assembly) to be loaded are either intact or undamaged:

1. the acceptable physical characteristics of the SNF (i.e., acceptable assembly defects and cladding breaches)

2. the intended functions the applicant has imposed on the SNF for demonstrating compliance with fuel-specific and system-related regulatory requirements

3. the alteration and degradation mechanisms during dry storage that could credibly compromise the ability of the fuel to meet its fuel-specific or system-related functions

4. discussions or analyses demonstrating that the mechanisms in 3 (above) will not reasonably affect the physical characteristics of the SNF (as defined in 1 above) or result in reconfiguration beyond the safety analyses in the SAR

Recognize that SFAs with any of the following characteristics, as identified during the fuel selection process, are expected to be classified as damaged unless an adequate justification is provided for otherwise:

- There is visible deformation of the rods in the SFA. This is not referring to the uniform bowing that occurs in the reactor; instead, this refers to bowing that significantly opens up the lattice spacing.

- Individual fuel rods are missing from the assembly. The assembly may be classified as intact or undamaged if the missing rod(s) do not adversely affect the structural performance of the assembly, radiological, and criticality safety (e.g., no significant changes to rod pitch). Alternatively, the assembly may be classified as intact or undamaged if a dummy rod that displaces a volume equal to, or greater than, the original fuel rod is placed in the empty rod location.

- The SFA has missing, displaced, or damaged structural components such that any one of the following conditions occur:

  – Radiological or criticality safety is adversely affected (e.g., significantly changed rod pitch).

  – The structural performance of the assembly may be compromised during normal, off-normal, and accident conditions of storage.

  – The assembly cannot be handled by normal means (i.e., crane and grapple), if the design bases relies on ready retrieval of individual fuel assemblies.

- Reactor operating records or fuel classification records indicate that the SFA contains fuel rods with gross breaches. (See NRC Information Notice 2018-01, “Noble Fission Gas Releases During Spent Fuel Cask Loading Operations”)

- The SFA is no longer in the form of an intact fuel bundle (e.g., consists of, or contains, debris such as loose fuel pellets or rod segments).

Recognize that defects such as dents in rods, bent or missing structural members, small cracks in structural members, and missing rods do not necessarily render an assembly as damaged, if the applicant can show that the intended functions of the assembly are maintained; that is, the performance of the assembly does not compromise the ability to meet fuel-specific and system-related regulations.

The staff considers a gross cladding breach as any cladding breach that could lead to the release of fuel particulate greater than the average size fuel fragment. A pellet is approximately 1.1 cm (0.4 in.) in diameter in 15 x 15 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, because of pellet-to-pellet abrasion. When the rod breaches, about 0.1 gram (3.5 ounces) of this fine powder may be carried out of the fuel rod at the breach site (see NUREG/CR-1773, "Fission Product Release from BWR Fuel Under LOCA Conditions," issued July 1981). Modeling the fragments as either spherical- or pie-shaped pieces indicates that a cladding-crack width of at least 2–3 mm (0.08–0.12 in.) would be required to release a fragment. Hence, gross breaches should be considered to be any cladding breach greater than 1 mm (0.04 in.).

8.5.15.2 *Uncanned Spent Fuel*

The review procedures in this section apply to undamaged or intact SNF that is not placed inside a separate fuel can in the DSS confinement; that is, the safety analyses rely on the integrity of the fuel cladding for maintaining the analyzed configuration.

8.5.15.2.1 Cladding Alloys

Identify the specific cladding alloys (e.g., Zircaloy-2, Zircaloy-4, ZIRLO™, M5®) and maximum burnup of the SNF to be stored. The staff considers the peak rod average burnup as an appropriate measure of maximum fuel burnup in the materials evaluation. Ensure that the SAR indicates that the fuel and cladding alloy contents are consistent with the technical bases in the structural evaluation.

Determine whether the SNF to be stored includes boron-based integral fuel burnable absorbers. Consider the fact that these rods have the potential to increase the fuel rod internal pressure from decay gas generation (helium) when evaluating aging mechanisms during dry storage, particularly for periods beyond 20 years (see Section 8.5.15.1). Decay gases are not generated in rods with gadolinium-based integral fuel burnable absorbers, which will not result in increased rod pressures beyond those generated by the fuel fission products.

8.5.15.2.2 Cladding Mechanical Properties

Ensure that the structural evaluation is bounding to all cladding alloys in the allowable contents (i.e., Zircaloy-2, Zircaloy-4, ZIRLO™, M5®). Ensure that the SAR provides a justification that the cladding mechanical properties are bounding upon consideration of alloy type and fabrication process (cold work stress relieved annealed, recrystallized annealed), hydrogen content, neutron fluence (burnup), oxide thickness, and cladding temperature.

The reviewer should recognize that the applicant may use mechanical properties of as-irradiated/in-reactor or pre-hydrided/irradiated cladding (i.e., not accounting for the potential reorientation of hydrides during DSS loading and storage operations) in the structural evaluation of the SNF assembly.

Alternatively, the applicant may use mechanical properties of cladding accounting for reoriented hydrides in the structural evaluation of the SNF assembly. However, to date, the database for these properties is very limited. Preferred sources of cladding materials data include manufacturer's test data obtained under an approved quality assurance program, NRC-approved topical reports, staff-accepted technical reports, as well as peer-reviewed articles, research reports, and texts. Ensure that the SAR includes adequate justification of the applicability and acceptability of any source of information.

The NRC deems the mechanical property models from PNL-17700 (Geelhood et al. 2008) acceptable for previous licensing and certification actions. However, the determination of acceptability should consider the limitations of these models based on the data used for model validation (refer to Chapter 5 of PNL-17700 for additional details). Note that the models in PNL-17700 were validated with experimental measurements on Zircaloy-4, Zircaloy-2, and ZIRLO™ cladding. Therefore, confirm that the applicant used other references for defining bounding mechanical properties for M5® cladding. Limited nonproprietary data are available for M5® cladding, that is, publicly available data from the French Competent Authority (Institut de Radioprotection et de Sûreté Nucléaire). The SAR should justify that the limited temperature-dependent M5® cladding property data are reasonably bounding upon consideration of hydrogen content, neutron fluence (burnup), oxide thickness, and cladding temperature. Coordinate with the structural reviewer (SRP Chapter 4) to ensure that there is sufficient safety margin in the respective vibration and drop analyses to ensure that the assumed properties are adequate. The reviewer may also rely on engineering judgment, which should be informed by the staff's findings on previous NRC-approved topical reports.

Ensure that the SAR justifies that the assumed hydrogen content and neutron fluence is adequately bounding to the maximum burnup of the cladding contents (refer to Chapter 5 of PNNL-17700 for additional details). In addition, ensure that the SAR justifies the assumed temperature for the cladding mechanical properties. For example, the applicant may choose to use cladding mechanical properties corresponding to the maximum fuel assembly temperature at the location of the peak stress identified in the dynamic analyses.

The models in PNL-17700 only account for mechanical properties of cladding with circumferential hydrides. The staff recognizes that the public database of mechanical properties of materials with both circumferential and radial hydrides is very limited (e.g., Kim et al. 2015a, 2015b). However, based on static bend testing of cladding with a high density of radial hydrides (see NUREG/CR-7198, "Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications," issued in 2017), the staff considers that these mechanical properties are adequate for the design-bases drop accidents during short-term loading operations (postulated accidents under 10 CFR 71.122(b)) or nonmechanistic DSS cask tipover accidents.

8.5.15.2.3 Effective Cladding Thickness

<u>Cladding Oxidation</u>

The structural evaluation should account for the reduced effective thickness of the cladding due to waterside corrosion (i.e., oxidation) during reactor service. The cladding oxide should not be considered load-bearing in the structural evaluation. The extent of oxidation and cladding wall thinning depends on the composition of the cladding (type of alloy) and burnup of the fuel. Note that the oxide will differ for the various cladding alloys and will not be of a uniform thickness along the axial length of the fuel rods. Ensure that the SAR defines an effective cladding thickness that is reduced by a bounding oxide layer to the specific cladding contents to be stored. Verify that the applicant has used a value of cladding oxide thickness that is justified by experimental oxide thickness measurements, computer codes validated using experimentally measured oxide thickness data, or other means that the staff finds appropriate. The NRC has determined that waterside corrosion models in the computer code FRAPCON 3.5 are acceptable for calculating oxide thickness values for Zircaloy-2, Zircaloy-4, ZIRLO™ and M5® cladding (see NUREG/CR-7022, "FRAPCON-3.5: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," issued October 2014).

Hydride Rim

During irradiation, some of the hydrogen generated due to water-side corrosion of the cladding will diffuse into the cladding. This results in the precipitation of hydrides in the circumferential-axial direction of the cladding when the amount of hydrogen generated exceeds the solubility limit in the cladding. The circumferential orientation of the hydrides is from the texture of manufactured cladding. The number density of these circumferential hydrides varies across the cladding wall because of the temperature drop from the fuel side (hotter) to the coolant side (cooler) of the cladding during reactor operation. Further, migration and precipitation of dissolved hydrogen to the coolant side of the cladding results in a rather dense hydride rim just below the corrosion (oxide) layer. The hydride number density and thickness of the rim depend on reactor operating conditions. For example, fuel rods operated at high linear heat rating to high burnup generally have a very dense hydride rim that is less than 10 percent of the cladding wall thickness. Conversely, fuel rods operated at low linear heat ratings to high burnup have a more diffuse hydride distribution that could extend as far as 50 percent across the cladding wall.

The applicant may have conservatively considered the cladding's outer hydride rim as wastage when determining the effective cladding thickness for the structural evaluation. However, there is no reliable predictive tool available to calculate this rim thickness, which varies along the fuel-rod length, around the circumference at any given axial location, from fuel rod to fuel rod within an assembly, and from assembly to assembly. Further, recent ring compression test results from the Argonne National Laboratory indicate that for the range of gas pressures anticipated during drying and storage, the hydride rim remains intact following slow cooling under conditions of decreasing pressure (Billone et al. 2013, 2014, 2015). These results indicate that the hydride rim is load bearing and can be accounted for in the effective cladding thickness calculation if mechanical test data referenced in the structural evaluation has adequately accounted for its presence. Historically, this has been the case during the review of DSSs, as applicants have provided mechanical property data generated from tests with irradiated cladding samples with an intact hydride rim. This includes test data derived from axial tensile tests or pressurized tube tests of samples that do not have a machined gauge section. For example, the mechanical property models used in PNL-17700 have been validated with experimental data from axial tensile tests on full cladding tubes and ring tests with no machined gauge section taken on irradiated recrystallized annealed Zircaloy-2 and Zircaloy-4 and stress-relief annealed ZIRLO™ cladding. As such, the staff considers any prior consideration to treat the rim as wastage to be unnecessary when calculating the effective cladding of the thickness, as the hydride rim has been properly accounted in the mechanical property models.

Drying Adequacy

Evaluate the descriptions related to draining and drying of the DSS confinement cavity during fuel loading operations, as discussed in the operating procedures chapter of the SAR. More specifically, ensure that the SAR clearly describes the procedures for removing water vapor and oxidizing material to an acceptable level, and that those procedures are appropriate.

The NRC staff has accepted vacuum-drying methods comparable to those recommended in PNL-6365, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel" (Knoll and Gilbert 1987). This report evaluates the effects of oxidizing impurities on the dry storage of light-water reactor (LWR) fuel and recommends limiting the maximum quantity of oxidizing gasses (e.g., oxygen, carbon dioxide, and carbon monoxide) to a total of 1 gram-mole per cask. This corresponds to a concentration of 0.25 volume percent of the total gases for a 7.0-cubic-meter (about a 247-cubic-foot) cask gas volume at a pressure of about 0.15 MPa

(1.5 atm) at 300 °K (80.3 °F). This 1 gram-mole limit reduces the amount of oxidants to below levels where cladding degradation is expected. Moisture removal is inherent in the vacuum drying process, and levels at or below those evaluated in PNL-6365 (about 0.43 gram-mole water) are expected if adequate vacuum drying is performed.

If alternative methods other than vacuum drying are used (such as forced helium recirculation), ensure that the applicant provides additional analyses or tests to sufficiently justify that moisture and impurity levels of the fuel cover gas will prevent unacceptable cladding degradation.

The following examples illustrate the accepted methods for cask draining and drying in accordance with the recommendations of PNL-6365 (Knoll and Gilbert 1987):

- The DSS confinement cavity should be drained of as much water as practicable and evacuated to less than or equal to 4.0×10^{-4} MPa (4 millibar, 3.0 mm Hg or Torr). After evacuation, adequate moisture removal should be verified by maintaining a constant pressure over a period of about 30 minutes without vacuum pump operation (or the vacuum pump is running but it is isolated from the cask with its suction vented to atmosphere). The DSS confinement cavity is then backfilled with an inert gas (e.g., helium) for applicable pressure and leak testing. Care should be taken to preserve the purity of the cover gas and, after backfilling, cover gas purity should be verified by sampling.

- The procedures should reflect the potential for blockage of the evacuation system or masking of defects in the cladding of non-intact rods, as a result of icing during evacuation. Icing can occur from the cooling effects of water vaporization and system depressurization during evacuation. Icing is more likely to occur in the evacuation system lines than in the DSS confinement cavity because of decay heat from the fuel. A staged draw down or other means of preventing ice blockage of the cask evacuation path may be used (e.g., measurement of cask pressure not involving the line through which the cask is evacuated).

- The procedures should specify a suitable inert cover gas (such as helium) with a quality specification that ensures a known maximum percentage of impurities to minimize the source of potentially oxidizing impurity gases and vapors and adequately remove contaminants from the cask.

- The process should provide for repetition of the evacuation and repressurization cycles if the DSS confinement cavity is opened to an oxidizing atmosphere following the evacuation and repressurization cycles (as may occur in conjunction with remedial welding, seal repairs). Refer to Appendix 8B, "Fuel Cladding Creep," to this SRP chapter for additional considerations on cladding oxidation and splitting.

Ensure that the drying specifications are consistent with the proposed operating controls and limits described in the technical specifications chapter of the SAR. In addition, assess the need for any additional technical specifications.

8.5.15.2.4 Maximum (Peak) Cladding Temperature

The acceptance criteria below and review procedures are designed to provide reasonable assurance that the spent fuel is maintained in the configuration analyzed in the SAR. The criteria below are applicable to all commercial spent fuel burnup levels and cladding materials. In order to assure integrity of the cladding material, the following criteria should be met:

1. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad). However, for low burnup fuel, a higher short-term temperature limit may be used, if the applicant can show by calculation that the best estimate cladding hoop stress is equal to or less than 90 MPa (13,053 psi) for the temperature limit proposed

2. For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

Coordinate with the thermal reviewer (SRP Chapter 5) to verify that the calculated maximum cladding temperature is based upon the peak rod temperature, not the average rod temperature. By employing the peak rod temperature, the safety analyses are conservatively bounding to all fuel rods in the content. Also confirm that the thermal models (and associated uncertainties) the applicant used for calculating cladding temperatures are acceptable to the thermal reviewer.

8.5.15.2.5 Thermal Cycling during Drying Operations

The reviewer should review fuel loading procedures to assure that any repeated thermal cycling (repeated heatup or cooldown cycles) during loading operations is limited to less than 10 cycles, where cladding temperature variations during each cycle do not exceed 65 °C (117 °F). The intent of the thermal cycling acceptance criteria is to limit precipitation of radial hydrides during loading operations. Evaluate the technical bases provided in support of any thermal cycling inconsistent with this criterion on a case-by-case basis. Further, reflooding of the previously dried high burnup fuel is not allowable unless the technical basis has adequately addressed the consequences of this operation on the performance of the cladding.

The applicant may use mechanical properties of cladding accounting for reoriented hydrides in the structural evaluation of the SFA. However, the database for these properties is very limited. For such applications, the loading procedures do not need to describe any thermal cycling limits if the applicant has adequately justified that the mechanical properties are reasonably bounding to reorientation expected for the design-bases heatup and cooldown cycles.

8.5.15.2.6 Cover Gas

Verify that the application defines the composition of the cover gas for the fuel during dry storage. Once the fuel rods are placed inside of the DSS confinement cavity and water is removed to a level that exposes any part of the rods to a gaseous atmosphere, the applicant must demonstrate that the SNF cladding will be protected against splitting from fuel pellet oxidation (10 CFR 72.122(h)(1)). If that atmosphere is oxidizing, then the fuel pellet may oxidize and expand, placing stress on the cladding. The expansion may eventually cause a gross rupture in the cladding, resulting in SNF that must be classified as damaged since it is not able to meet the requirement in 10 CFR 72.122(h)(1). The configuration of the fuel should remain bounded by the reviewed safety analyses. Further, the release of fuel fines, or grain-sized powder, from ruptured fuel into the confinement cavity may be a condition outside the design-bases for the DSS design. Three possible options exist to address the potential for and consequences of fuel oxidation:

1. Maintain the fuel rods in an inerted environment such as argon, nitrogen gas, or helium to prevent oxidation.

2. Ensure that there are not any cladding breaches (including hairline cracks and pinhole leaks) in the fuel pin sections that will be exposed to an oxidizing atmosphere

3. Determine the time-at-temperature profile of the rods while they are exposed to an oxidizing atmosphere and calculate the expected oxidation to determine if a gross breach would occur. The analysis should indicate that the time required to incubate the splitting process will not be exceeded. Such an analysis would have to address expected differences in characteristics between the fuel to be loaded and the fuel tested in the referenced data. The design-bases maximum allowable cladding temperature should be limited to the temperature at which calculations show that cladding splitting is not expected to occur. Such evaluations should address uncertainties in the referenced database.

If option 3 is chosen, coordinate with the thermal reviewer (SRP Chapter 5) to determine that the operating procedures (SRP Chapter 11, "Operation Procedures and Systems Evaluation") and the technical specifications (SRP Chapter 17, "Technical Specification Evaluation") of the license or CoC, as submitted by the applicant, provide an adequate analysis of the potential for cladding splitting should fuel rods be exposed to an oxidizing gaseous atmosphere.

Fuel oxidation and cladding splitting follow Arrhenius time-at-temperature behavior. For fuel burnups not exceeding 45 GWd/MTU and Zircaloy cladding, the time-at-temperature curves for uranium-based fuel developed to date (e.g., Einziger and Strain 1986) can be used to determine the allowable exposure duration on an oxidizing atmosphere for a given design-bases fuel cladding temperature. For example, using Figure 3-9 of Einziger and Strain (1986), at 360 °C (680 °F) one would expect to incur splitting at between 2 and 10 hours. On the other hand, if one expected the cladding temperature to stay at temperature for 100 hours, then the fuel temperature should be kept below 290 °C (554 °F). Refer to Appendix 8C, "Fuel Oxidation and Cladding Splitting," to this SRP for additional information on cladding splitting.

8.5.15.2.7 High Burnup Fuel Monitoring and Assessment (dry storage periods beyond 20 years)

Under the regulations in 10 CFR 72.42, "Duration of license; renewal," and 10 CFR 72.238, "Issuance of an NRC Certificate of Compliance," an applicant may request an initial license or CoC storage period, respectively, that does not exceed 40 years. Experimental confirmatory data, as described in NUREG/CR-6745, "Dry Cask Storage Characterization Project—Phase 1; CASTOR V/21 Cask Opening and Examination," and NUREG/CR-6831, "Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage," has shown that the integrity of low burnup fuel (less than or equal to 45 GWd/MTU) in dry storage is not expected to be impacted for periods up to 40 years.

For high burnup fuel (i.e., fuel with burnups generally exceeding 45 GWd/MTU), dry storage has been allowed for periods up to 20 years without the need to provide confirmatory data that the SNF configuration will remain as analyzed. However, for a license or storage term exceeding 20 years, verify that the applicant provided a maintenance plan to obtain such confirmatory data. Refer to NUREG-1927, Revision 1, when evaluating proposed maintenance activities for providing confirmatory data. These maintenance activities should be consistent with aging management activities (e.g., aging management program) during a renewed license or CoC storage period; that is, periods between 20 and 60 years. Refer to the discussions on Chapters 2 and 3, and Appendices D and B to NUREG-1927, Revision 1, for the review of acceptable maintenance activities.

8.5.15.2.8 Release Fractions

The materials reviewer should coordinate with the confinement reviewer to ensure that the SAR has provided adequate release fractions for the proposed fuel contents if the DSS confinement is non-leaktight. The technical basis may include an adequate description of the supporting experimental data, including a description of the burnups of the test specimens, number of tests, and test specimen pressure at the time of fracture. Further, the collection method used for quantification of the release fractions should be sophisticated enough to gather respirable release fractions.

The materials reviewer should recognize that high burnup fuel has different characteristics than low burnup fuel with respect to CRUD thickness, cladding oxide thickness, hydride content, radionuclide inventory and distribution, heat load, fuel pellet grain size, fuel pellet fragmentation, fuel pellet expansion and fission gas release to the rod plenum (see Appendix C.5 to NUREG/CR-7203, "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages," issued September 2015 (NRC 2015) for a description of high burnup fuel). Differences in these characteristics affect the mechanisms by which the fuel can breach and the amount of fuel that can be released from failed fuel rods. Hence, the SAR may provide different release fractions (CRUD, fission gases, volatiles, and fuel fines) for low and high burnup fuel in non-leaktight confinement.

8.5.15.3 *Canned Spent Fuel*

SNF that has been classified as damaged for storage must be confined in a can designed for damaged fuel or in an acceptable alternative (10 CFR 72.122(h)(1)). The purpose of a can designed for damaged fuel is to (1) confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask; (2) demonstrate compliance with the criticality, shielding, thermal, and structural requirements; and (3) permit normal handling and retrieval from the storage container (if ready retrieval of the can is required per the design-bases). The can designed for damaged fuel may need to contain neutron-absorbing materials if results of the criticality safety analysis depend on the neutron absorber to meet the requirements in 10 CFR 72.124(a).

The configuration of the fuel inside the fuel can is generally not restricted; therefore, the applicant should perform bounding safety analyses assuming full reconfiguration of the fuel inside the fuel can. Ensure that the assumed mechanical properties of the fuel can are adequate for the calculated temperatures in the reconfiguration analyses. Ensure that the mechanical properties of the fuel can are also adequate for demonstrating adequate structural performance to ensure that the fuel remains confined to the can during normal, off-normal, and design-bases accident conditions.

8.6 <u>Evaluation Findings</u>

The NRC reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Section 8.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of finding should be similar to the following:

Specific License

F8.1 The applicant has met the requirements in 10 CFR 72.24(c)(3) and 10 CFR 72.120(a). The applicant described the materials used for SSCs important to safety in sufficient detail to support a safety finding.

F8.2 The applicant has met the requirements in 10 CFR 72.24(d) and 10 CFR 72.128(a). The properties of the materials in the storage facility design have been demonstrated to support the safe storage and handling of SNF, HLW, and reactor-related GTCC waste for the storage term under normal, off-normal, and accident conditions.

F8.3 The applicant has met the requirements in 10 CFR 72.124(b). Neutron absorbing materials are demonstrated to effectively control criticality without significant degradation over the storage life.

F8.4 The applicant has met the requirements in 10 CFR 72.120(d), 10 CFR 72.122(b)(1), and 10 CFR 72.124(b). Materials and storage contents are compatible with their operating environment such that there will be no adverse degradation or significant chemical or other reactions.

F8.5 The applicant has met the requirements in 10 CFR 72.122(c). Operating procedures contain measures for detecting the presence of hydrogen and preventing the ignition of combustible gases during cask loading and unloading operations.

F8.6 The applicant has met the requirements in 10 CFR 72.122(h)(1). The SNF cladding has been demonstrated to be adequately protected against gross ruptures, or the fuel has been demonstrated to be otherwise confined.

F8.7 The applicant has met the requirements in 10 CFR 72.122(h)(5) and 10 CFR 72.122(l). The packaging of HLW and reactor-related GTCC waste ensures that handling and retrievably is adequately maintained. The storage system is designed to allow ready retrieval of SNF, HLW, and reactor-related GTCC waste.

F8.8 The applicant has met the requirements in 10 CFR 72.24(c)(4) and 10 CFR 72.122(a). The use of codes and standards, quality assurance programs, and control of special processes are demonstrated to be adequate to ensure that the design, testing, fabrication, and maintenance of materials support SSC intended functions.

Certificate of Compliance

F8.9 The applicant has met the requirements in 10 CFR 72.236(b). The applicant described the materials design criteria for SSCs important to safety in sufficient detail to support a safety finding.

F8.10 The applicant has met the requirements in 10 CFR 72.124(b). Neutron-absorbing materials are demonstrated to effectively control criticality without significant degradation over the storage life.

F8.11 The applicant has met the requirements in 10 CFR 72.236(g). The properties of the materials in the storage system design have been demonstrated to support the safe storage of SNF.

F8.12 The applicant has met the requirements in 10 CFR 72.236(h). The materials of the SNF storage container are compatible with their operating environment such that there are no adverse degradation or significant chemical or other reactions.

F8.13 The applicant has met the requirements in 10 CFR 72.236(a) and 10 CFR 72.236(m). SNF specifications have been provided and adequate consideration has been given to compatibility with retrieval of stored fuel for ultimate disposal.

F8.14 The applicant has met the requirements in 10 CFR 72.234(b). Quality assurance programs and control of special processes are demonstrated to be adequate to ensure that the design, testing, fabrication, and maintenance of materials support SSC intended functions.

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

> The staff concludes that the [DSS or DSF designation] design adequately considers material properties, environmental degradation and other reactions, fuel clad integrity, content retrievability, and material quality controls such that the design is in compliance with 10 CFR Part 72. The evaluation of these materials considerations provides reasonable assurance the [DSS or DSF designation] will allow safe storage of [SNF/HLW/GTCC waste content designation] for a licensed (certified) life of [X] 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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