# 7 MATERIALS EVALUATION

## 7.1 Review Objective

The objective of this U.S. Nuclear Regulatory Commission (NRC) material evaluation is to verify that the applicant has adequately evaluated the materials performance of the transportation package under normal conditions of transport and hypothetical accident conditions necessary to meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, “Packaging and Transportation of Radioactive Material.”

In conducting the reviews, the NRC reviewer should ensure that materials meet applicable codes, standards, and specifications to support the intended functions of the components under normal conditions of transport and hypothetical accident conditions. The review also includes the evaluation of operations that ensure adequate materials performance, including material qualification, welding, acceptance testing, and inerting of the containment system.

## 7.2 Areas of Review

The NRC staff should review the application to verify that it adequately describes the package and includes adequately detailed drawings. In general, the staff should review the following information to determine the adequacy of the package description:

- drawings
- codes and standards
  - usage and endorsement
  - American Society of Mechanical Engineers (ASME) Code Component
  - code case use/acceptability
  - non-ASME code components
- weld design and inspection
  - moderator exclusion for commercial spent nuclear fuel (SNF) packages under hypothetical accident conditions
- mechanical properties
  - tensile properties
  - fracture resistance
  - tensile properties and creep of aluminum allows at elevated temperatures
  - impact limiters
- thermal properties of materials
- radiation shielding
  - neutron-shielding materials
  - gamma-shielding materials

- criticality control
    - — neutron-absorbing (poison) material specification
    - — computation of percent credit for boron-based neutron absorbers
    - — qualifying properties not associated with attenuation
- corrosion resistance
    - — environments
    - — carbon and low allow steels
    - — austenitic stainless steel
- protective coatings
    - — review guidance
    - — scope of coating application
    - — coating selection
    - — coating qualification testing
- content reactions
    - — flammable and explosive reactions
    - — content chemical reactions, outgassing, and corrosion
- radiation effects
- package contents
- fresh (unirradiated) fuel cladding
- SNF
    - — spent fuel classification
    - — uncanned spent fuel
    - — canned spent fuel
- bolting material
- seals
    - — metallic seals
    - — elastomeric seals

## 7.3 Regulatory Requirements and Acceptance Criteria

Table 7-1 summarizes the sections of 10 CFR Part 71 that are relevant to the materials review and addressed this chapter of the standard review plan (SRP). The reviewer should refer to the language in the regulations and verify the association of regulatory requirements with the areas of review and ensure that no requirements are overlooked as a result of unique design features.

**Table 7-1 Relationship of Regulations and Areas of Review for Transportation Packages**

| Areas of Review | 10 CFR Part 71 Regulations | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | **71.31** | **71.33** | **71.35** | **71.43** | **71.51** | **71.55** | **71.64** | **71.71** | **71.73** | **71.74** | **71.85** | **71.87** |
| Material description | (a)(1) | ● | | (c),(d),(f) | (a)(1) | (b),(d), (e),(f) | (a),(b) | | | | | |
| Codes and standards; quality controls | (c) | | | | | | | | | | | |
| Material properties | (a)(1)(2) | ● | (a) | (c),(d),(f) | (a)(1) | (b),(d), (e),(f) | (a),(b) | ● | ● | ● | (a) | (a),(b),(c), (f),(g) |
| Corrosion, chemical reactions, and radiation effects | (a)(2) | | (a) | (d),(f) | (a)(1) | (b)(1), (d)(3), (e)(1)(2), (f) | | ● | ● | ● | (a) | (a),(b),(c), (f),(g) |
| Content integrity | (a)(1)(2) | (b) | (a),(c) | (f) | (a)(1)(2) | (b), (d)(2)(4), (e)(1)(2), (f)(1)(2) | (a) | ● | ● | ● | | (a),(f) |

Note: The bullet (●) indicates the entire regulation as listed in the column heading applies.

Acceptability of the design of the packages used for the transport of radioactive materials, as described in the application, is based on compliance with the requirements of 10 CFR Part 71 and regulatory guidance.

The materials evaluation seeks to ensure that materials will perform in a manner that supports the structural, thermal, containment, shielding, and criticality-control functions of the transportation package, in accordance with the requirements of 10 CFR Part 71, under normal conditions of transport, hypothetical accident conditions, and air-transport conditions, as applicable. The application must contain sufficient information on materials of construction, including their fabrication, evaluation, testing, and special processes. The design and construction of the packaging must identify all applicable codes and standards. Noncode materials must have adequate controls for their qualification and fabrication. Material properties, including mechanical, thermal, shielding, and neutron absorption, should have an adequate technical basis and must demonstrate support for the performance and intended functions of components under normal conditions of transport and hypothetical accident conditions. Materials must not undergo significant chemical, galvanic, or other reactions, or radiation-induced degradation that could challenge the ability of the packaging to safely transport radioactive materials and SNF. The transportation package must be designed and constructed such that the analyzed geometric form of its contents and content characteristics described in SRP section 6.4.2 will not be substantially altered and there will be no loss or dispersal of the contents.

## 7.4 Review Procedures

The NRC reviewer should ensure that the application adequately describes and evaluates the materials used in the transportation package under normal conditions of transport and hypothetical accident conditions to demonstrate that they meet the requirements of 10 CFR Part 71. Figure 7-1 shows the interrelationship between the materials evaluation and other areas of review described in the SRP. In addition, since the material review is interdisciplinary, the materials reviewer should coordinate with other reviewers (e.g., structural, thermal, shielding, criticality), as necessary, for identification of materials-related issues in other application chapters.

### 7.4.1 Drawings

General guidance on the content of drawings is provided in Chapter 1, “General Information Evaluation,” of this SRP. Examine the application and verify that the engineering drawings are consistent with the design and description of the package, in accordance with 10 CFR 71.33, “Package Description.” Survey the application and design drawings to identify the various materials used in the packaging design and potential material issues. Use the guidance in NUREG/CR-5502, “Engineering Drawings for 10 CFR Part 71 Package Approvals,” issued May 1999, and Regulatory Guide 7.9, “Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material,” as appropriate, for the recommended content of engineering drawings. Verify that the drawings clearly detail the design features considered in the package evaluation, including the following:

- containment systems
- closure devices
- internal supporting or positioning structures
- neutron absorbing and moderating features affecting criticality

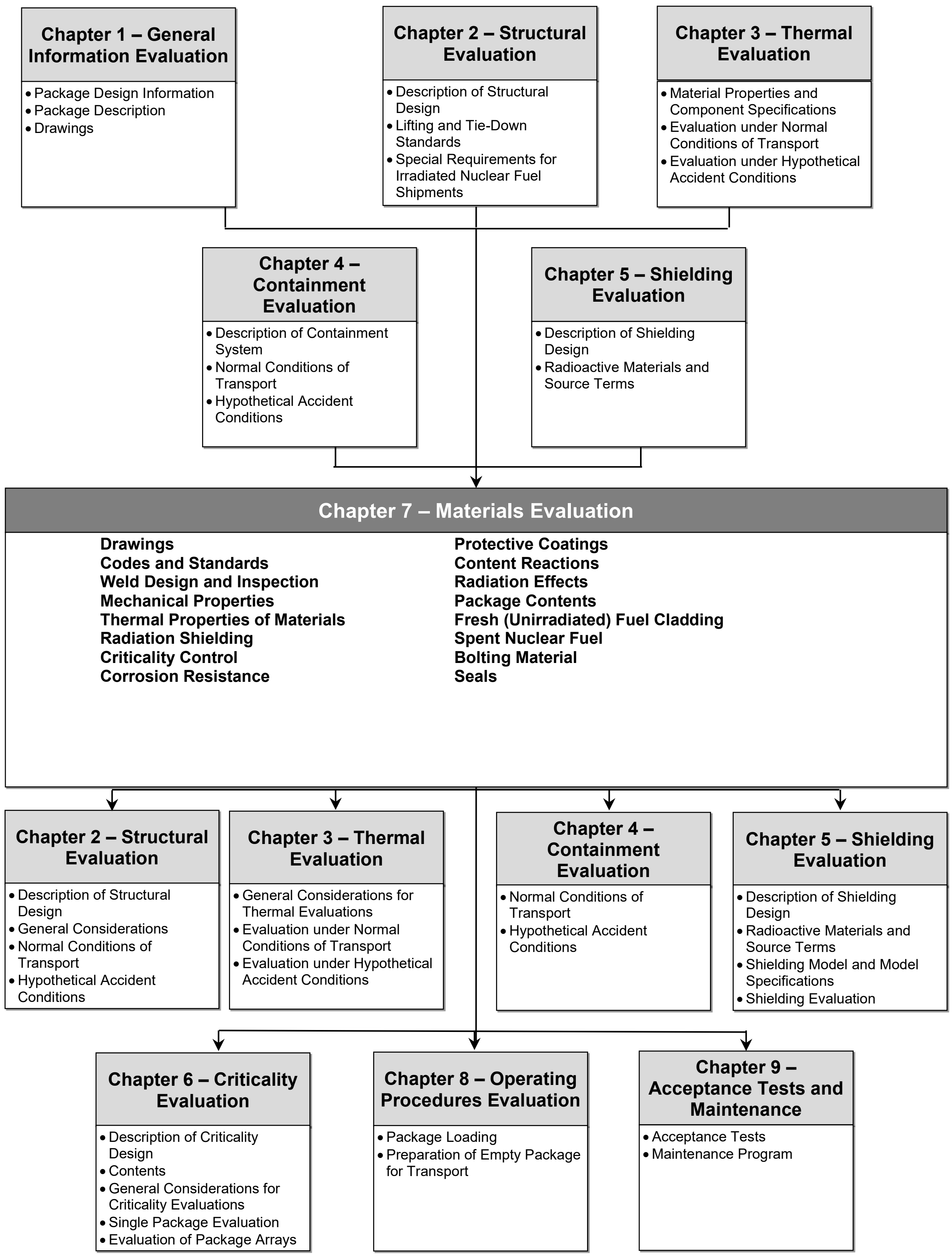


**Figure 7-1 Information Flow for the Materials Evaluation**

- neutron shielding
- gamma shielding
- outer shell or outer packaging
- heat-transfer features
- impact limiters and energy-absorbing features
- lifting and tie-down devices
- personnel barriers

The information should be sufficient for evaluating the material performance of the packaging components and systems important to safety to meet the regulatory requirements. Refer to NUREG/CR-6407 “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” issued February 1996, and NRC Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material," Appendix A, “A Graded Approach to Developing Quality Assurance Programs for Packaging Radioactive Material,” for guidance on safety classification of transportation packaging components. Drawings may include a parts list that identifies the safety classification assigned to each individual component, consistent with the component function and requirements.

Verify that the drawings include the following information:

- materials of construction

- dimensions and tolerances

- codes, standards, or other specifications for materials (e.g., minimum density and minimum hydrogen and boron content for neutron shields and minimum boron-10 areal density for boron-based neutron absorbers), fabrication, examination, and testing

- welding specifications, including location and nondestructive examination (NDE)

- coatings and other special material treatments that perform a safety function

- specifications and requirements for alternative materials

Confirm that the application text and figures that describe the materials are consistent with the engineering drawings.

Verify that standard welding and NDE symbols are included to aid interpretation of the drawings. Standard welding and NDE symbols may be found in American Welding Society (AWS) A2.4, “Symbols for Welding, Brazing, and Nondestructive Testing.”

### 7.4.2 Codes and Standards

The guidance below describes the materials, codes, and standards the NRC staff finds acceptable for the construction of transportation packages. Confirm that the application identifies any established codes and standards proposed for use in package design, fabrication, assembly, testing, maintenance, and use, in accordance with 10 CFR 71.31(c). Because the guidance adopts portions of nuclear reactor facility codes, exceptions or additions to those codes may be recommended to address unique aspects of transportation package designs.

#### *7.4.2.1 Usage and endorsement*

For components of packaging important to safety, ensure that the application specifies the U.S. industry consensus codes and standards, such as the ASME Boiler and Pressure Vessel (B&PV) Code, AWS Codes, American National Standards Institute (ANSI) standards, and American Society for Testing and Materials (ASTM) International standards. Foreign codes and standards generally are not acceptable for components of packaging important to safety and should be approved only on a case-by-case basis. If the application includes foreign codes, verify that they are cross-referenced to appropriate U.S. standards.

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

#### *7.4.2.2 ASME code components*

As discussed in Section 2.4.1.2 of this SRP, the transportation containment system should be designed and constructed in accordance with the ASME Code Section III, Division 1 or Division 3. Historically, Division 1 has been the accepted portion of the ASME Code.

NUREG/CR-3854, “Fabrication Criteria for Shipping Containers,” issued March 1985, describes materials and fabrication criteria that the NRC finds acceptable for the construction of transportation packages. Table 4.1 of NUREG/CR-3854 recommends ASME Code Section III, Division 1, criteria for the fabrication of containment, criticality, and other safety components. For example, for Category I containers (i.e., those that transport SNF), NUREG/CR-3854 recommends that containment components be fabricated in accordance with ASME Code Section III, Division 1, Subsection NB (Class 1) criteria, fuel basket structures be fabricated in accordance with Subsection NG (Core Supports), and other safety structures be fabricated in accordance with Subsection NF (Supports).

The NRC also accepts the use of ASME Section III, Division 3 for the fabrication, welding, examination, testing, inspection, and certification of transportation containment systems. Ensure that the application includes a justification for any deviations from Section III, Division 1 or Division 3 for the containment design or component materials important to safety.

#### *7.4.2.3 Code case use/acceptability*

The NRC reviews of the acceptability of ASME code cases are documented in NRC regulatory Guides (RG), including RG 1.193, “ASME Code Cases Not Approved for Use,” and RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” These regulatory guides are periodically updated (generally about every 2 years). Review any referenced ASME Code Cases against the latest versions of RG 1.193 and RG 1.84 to determine code case acceptability. Table 1 of RG 1.84 provides a list of cases the NRC finds acceptable, while Table 2 of RG 1.84 provides a list of conditionally approved cases. 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 the latest revision of RG 1.84 to ensure that the application does not reference any annulled or superseded codes cases.

#### *7.4.2.4 Non-ASME code components*

Components of packaging important to safety that do not comprise the containment boundary may be constructed of materials the ASME, ASTM, or the American Iron and Steel Institute certified. Components of packaging that are not important to safety can be specified by generic names such as “stainless steel,” “aluminum,” or “carbon steel,” provided that the applicant provided sufficient information to evaluate potential impacts that components not important to safety may have on components of packaging important to safety (e.g., galvanic corrosion).

The NRC approves the use of proprietary materials on a case-by-case basis. Ensure that the application describes proprietary materials important to safety (e.g., impact limiter materials, neutron poisons, polymeric neutron shields) to permit the staff to make a safety finding. The Acceptance Tests and Maintenance Program described in the application should incorporate by reference the governing quality assurance and quality control documents, key manufacturing procedures, and key testing protocols for proprietary materials. In the absence of any codes or standards for a special process, verify that the application includes a description of the process, controls, and quality assurance measures.

### 7.4.3 Weld Design and Inspection

As discussed in Section 7.4.2.2, the transportation containment systems should be designed and constructed in accordance with ASME Code Section III, Division 1 or Division 3. Confirm that the application identifies any established codes and standards proposed for use in package design, fabrication, assembly, testing, maintenance, and use in accordance with 10 CFR 71.31(c). The ASME Code defines required welding criteria, including welding processes, filler metal, qualification procedures, heat treatment, examination, and testing. Refer to the acceptable fabrication criteria for shipping containers in NUREG/CR-3854 along with the relevant portions of the ASME Code to ensure that the application and drawings for the containment boundary and components of packaging important to safety are consistent with the code-required welding criteria.

For containment systems designed in accordance with ASME Code Section III, Division 1, refer to NUREG/CR-3019, “Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials,” issued March 1985. This guidance identifies the locations in the ASME Code where the reviewer can find the welding criteria for containment-related, criticality-related (e.g., fuel baskets), and other safety-related welds. For designs that use Division 3 of the ASME Code rather than Division 1, review that section of the ASME Code to identify the corresponding requirements.

Welds that are not associated with a safety function (e.g., not part of the containment boundary or items relied on for criticality safety or shielding) may be governed by the ASME Code, AWS Codes, or American Institute of Steel Construction (AISC) “Manual of Steel Construction” (AISC 1989). AISC standards may, in turn, reference AWS Codes. Similar to the ASME 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.

There is no need to verify the presence of specific welding criteria, such as filler metal and weld processes, if the transportation package weld design is consistent with the ASME or AWS Codes and the application and design drawings clearly define the code applicability. 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 also comply with the ASME or AWS Codes. Confirm that the application identifies any established codes and standards proposed for use in package design, fabrication, assembly, testing, maintenance, and use, in accordance with 10 CFR 71.31(c).

#### *7.4.3.1 Moderator exclusion for commercial spent nuclear fuel packages under hypothetical accident conditions*

For fissile material packages, 10 CFR 71.55(e) requires that the package be subcritical under hypothetical accident conditions. Verify that the applicant demonstrated that the package remains subcritical by (i) showing that reconfigured fuel is subcritical even with water inleakage or (ii) showing that the package excludes water under hypothetical accident conditions. Thus, the staff has developed options for the evaluations to demonstrate compliance with 10 CFR 71.55(e). Additional guidance for each of these approaches is included in Section 1.4.4 of this SRP.

### 7.4.4 Mechanical Properties

Assess the acceptability of all material mechanical properties for components of packaging important to safety. Ensure that the mechanical properties account for environmental and operating conditions during normal conditions of transport (hot and cold temperatures) and hypothetical accident conditions, considering also the potential for microstructural changes at elevated temperatures, in order to meet the requirements of 10 CFR 71.33, 71.35(a), 71.51(a) and 71.55(b), (d), (e), and (f) and 71.64, "Special Requirements for Plutonium Air Shipments," as applicable. Verify that appropriate exposure temperatures and times at which allowable stress limits are defined are consistent with the thermal conditions evaluated in the thermal analysis.

#### *7.4.4.1 Tensile properties*

Verify that the application clearly references acceptable sources of all material properties. The properties used in the structural evaluation should be consistent with the design criteria (codes, standards, specifications). For example, if a component is designed to a particular subsection of ASME Code Section III, the material properties and requirements for the component should be consistent with those allowed by that subsection.

For components designed to the ASME Code, acceptable material properties, allowable stresses, temperature limits, and other requirements include those provided in ASME Code Section II, Part A, "Ferrous Metals;" Part B, "Nonferrous Metals;" Part C, "Welding Rods, Electrodes, and Filler Metals;" and Part D, "Properties." Verify that the application justifies the Code alternatives in order to enable an assessment of their acceptability. Other references (e.g., Military Handbook and ASTM standards) may be used for components not designed to the ASME Code. Verify that the application provides adequately documented material properties and specifications for the design and fabrication of the packaging.

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.

#### *7.4.4.2 Fracture resistance*

Refer to ASME Section III NB-2300, “Fracture Toughness Requirements for Material,” when evaluating a new package or new material for components of packaging important to safety. 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. Note, however, that this only applies to nonferrous materials that are included in ASME Section II, Tables 2A and 2B. For some package designs, components that are not part of the containment boundary may 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. Materials that provide a structural function should be reviewed to determine adequate resistance to fracture.

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 for components of packaging important to safety.

Ferritic Steels

Several types of ferritic steels may become brittle at low service temperatures. Section III of the ASME Code contains requirements for material fracture toughness; however, these requirements were developed for reactor components and do not address hypothetical accident conditions for transportation packaging. Therefore, 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 in the RGs. Those tests and data include dynamic fracture toughness and nil-ductility or fracture appearance transition temperature test data. ASME Section III, as supported by Section IX, governs toughness testing (e.g., Charpy impact) of welds.

Duplex Stainless Steels

Duplex stainless steels have both ferritic and austenitic phases and are susceptible to phase instability that may affect fracture toughness. Verify that the application includes 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.

The NRC has approved duplex stainless steels for the construction of dual-purpose transportable SNF storage canisters, and NUREG-2215, “Standard Review Plan for Spent Fuel

Dry Storage Systems and Facilities," issued November 2017, provides additional guidance for the review of welding practices for these steels.

Aluminum Alloys and Aluminum Metal Matrix Composites

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 appropriate (e.g., ASM International 1998; Kaufman et al. 1971).

The fracture toughness of aluminum metal matrix composites (MMCs) depends on many factors, including (i) particle composition, (ii) particle size, (iii) particle loading, (iv) particle distribution or clustering, (v) alloy composition, and (vi) thermal treatment for aluminum alloys that can be precipitation hardened. The fracture toughness of aluminum MMC has been found to range from 8 to 30 thousand pounds per square inch (ksi)-in$^{1/2}$ [$5.5 \times 10^7$ to $2.1 \times 10^8$ pascal (Pa)] (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 using valid fracture toughness data. Calculated values of fracture toughness using impact toughness data may be acceptable, provided that the applicant justified the aluminum-specific correlation between the two types of data.

*7.4.4.3 Tensile properties and creep of aluminum alloys at elevated temperatures*

Verify that the application considers appropriate mechanical properties for aluminum components that have a structural function. Many aluminum alloys, including 2000 series and 6000 series alloys, can be thermally treated to increase yield and tensile strength. For example, Al 6061, a common structural aluminum alloy used in basket assemblies, is precipitation-hardened with magnesium sulfide and is commercially available in several tempers with significantly different yield and tensile strengths and ductility values. Al 6061 is available in pre-tempered grades such as annealed 6061-O and tempered grades such as 6061-T6 and 6061-T651. Both 2000 and 6000 series precipitation-hardened aluminum alloys are used in various basket support components of dual-purpose (storage and transportation) canister designs.

The prolonged effects of elevated temperatures during storage of a dual-purpose canister can affect the properties of precipitation-hardened aluminum alloys. For Al 6061, the allowable stress decreases with increasing temperature for all tempers including T4, T451, T6, and T651. Aging at higher temperature or holding at higher temperature after aging at 320 degrees Fahrenheit (°F) [160 degrees Celsius (°C)] will coarsen the magnesium sulfide precipitates and correspondingly reduce the strength of the alloy (Farrell 1995). Verify that the mechanical properties account for such microstructural changes that affect yield and tensile strength. Note that ASME Section II, Part D, Table 1B requires that time-dependent properties be used for precipitation-hardened Al 6061 at temperatures at or above 350 °F [177 °C].

More recent dual-purpose (storage and transportation) canister 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 into creep regime operating temperatures. Refer to the guidance on the assessment of creep of aluminum components in NUREG-2215, Chapter 8, "Materials Evaluation." The NRC considers the storage system review guidance for creep of aluminum components of dual-purpose canisters to be appropriate for evaluating the performance of these materials during transportation.

#### *7.4.4.4 Impact limiters*

Impact limiters often use special materials such as wood, foam, resin, and honeycomb metals to provide specified crushing characteristics. Verify that the applicant has identified appropriate acceptance testing to assure adequate material properties. Also, verify that the force-deflection properties for all directions evaluated for the packaging are based on test conditions (e.g., strain rate, temperature) that are applicable to the transportation package. Note that the use of unreasonably low material strength values may not be conservative, as this can minimize the decelerations considered in the accident analyses. Testing of the impact limiters may be carried out statically if the effect of strain rate on the material crush properties is accounted for and properly included in the force-deflection relationship for impact analysis.

Impact limiter materials may be temperature and time dependent. In addition, wood and polymeric materials may absorb moisture in service, affecting their properties. Verify that the acceptance testing is sufficient to evaluate the mechanical properties of the impact limiter materials under environmental conditions and temperatures that are expected in service.

### 7.4.5 Thermal Properties of Materials

Coordinate with the thermal reviewer to determine the properties of the materials important to the thermal analysis. Confirm that the application identifies materials and package components used for heat transfer in accordance with 10 CFR 71.33(a)(5) and (6). Verify the material compositions and thermal properties, such as thermal conductivity, thermal expansion, specific heat, density, and heat capacity, as a function of temperature over the ranges the components experience under the conditions associated with the tests in 10 CFR 71.71, “Normal Conditions of Transport,” and 10 CFR 71.73, “Hypothetical Accident Conditions,” (and other relevant tests for packages for air transport of fissile material or plutonium in accordance with, respectively, 10 CFR 71.55(f) and 10 CFR 71.7, “ Completeness and Accuracy of Information”). Verify that the applicant has evaluated the change in these material properties from material degradation over their service life. Consider, also, the anisotropic dependencies of thermal properties.

### 7.4.6 Radiation Shielding

Verify that the application describes the compositions and geometries of shielding materials. Steel, lead, depleted uranium, and tungsten typically serve as gamma-shielding materials, while filled polymers are often used for neutron shielding. References for all materials used, including nonstandard materials (e.g., proprietary neutron-shield material), should provide the material composition and density data over the range of temperatures for normal conditions of transport, along with validation of the data. Also, verify that the application describes the geometry of the shielding materials. Coordinate the materials evaluation with the shielding reviewer (Chapter 5, “Shielding Evaluation,” of this SRP) to confirm that the application meets the requirements of 10 CFR 71.43(f), 71.51(a), and 71.64(a), as applicable. Also, in coordination with the shielding reviewer, verify that the applicant has adequately described the acceptance testing conducted for gamma- and neutron-shielding materials, as described in NUREG/CR-3854.

#### *7.4.6.1 Neutron-shielding materials*

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

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

- test(s) demonstrating the neutron-absorbing ability of the shield material
- the testing program, providing 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 effects on neutron-shield performance
- provisions that 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; how such changes 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 confirm the neutron shield's effectiveness and to verify that any filled channels used on production casks do not have significant voids or defects that could lead to greater-than-calculated dose rates
- the material's ability to withstand the combined aging effects of heat and radiation field

Verify that the application (i) describes the potential for shielding material to experience changes in material properties at temperature extremes, (ii) describes or provides a reference for the temperature sensitivities of shielding materials, (iii) addresses degradation from aging, and (iv) accounts for manufacturing tolerances (both material and dimensional).

#### *7.4.6.2 Gamma-shielding materials*

For transportation packaging, steel, depleted uranium, tungsten, cast iron, and lead may be used as gamma radiation shields. Refer to NUREG/CR-3854 for guidance on shield installation and acceptance testing. Collaborate with the shielding reviewer to ensure that the material compositions and densities used in the shielding models are consistent with the design features described in the application. The shielding properties should account for manufacturing tolerances and expected degradation from corrosion reactions, elevated temperature, and accumulated radiation exposure.

Ensure that the application describes the physical dimensions of shielding materials, including seams, penetrations, or voids. For example, lead shielding may be applied by pouring or stacking like bricks or plates and using lead wool to fill gaps. Ensure that the application indicates that manufacturing controls are in place to address any potential paths for gamma streaming. For poured-lead shielding, ensure that the applicant used methods that reduce the possibility of air entrainment in the molten lead during the pouring and removal of the lead froth after pouring.

Some gamma-shielding materials may also undergo degradation at elevated temperatures or under oxidizing conditions. Lead has a relatively low melting point {327 °C [622 °F]}. Verify that the applicant has assessed the potential for lead slumping as a result of loading during normal conditions of transport or from exposure to elevated temperatures.

Coordinate with the shielding and structural reviewers to verify that, for packages that rely on depleted uranium for shielding, the package design ensures that the depleted uranium will not be exposed to the environment (i.e., to air) as a result of the regulatory impact and puncture tests. Depleted uranium exposed to the air for the 10 CFR 71.73 thermal tests can significantly oxidize, resulting in a loss of this material to perform a shielding function. Uranium oxides can have significantly larger volumes than the uranium metal and subsequent volume expansion and may lead to stresses in adjacent packaging components. The formation of uranium hydride can occur when uranium is exposed to moisture under reducing conditions (e.g., in the absence of oxygen). Uranium hydrides in powder form can be pyrophoric. Verify that the package design incorporated features that protect the depleted uranium against oxidation and the formation of uranium hydrides.

### 7.4.7 Criticality Control

Various materials are used 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 ($B_4C$) particles, clad on both sides with aluminum (a composite). They may also consist of other materials such as cadmium, gadolinium, and silver-indium-cadmium that may or may not be alloyed or mixed with other materials.

Coordinate with the criticality control review to assess the packaging design and the contents specified such that the package is subcritical under the design-basis conditions, normal conditions of transport, and hypothetical accident conditions, in accordance with 10 CFR 71.55(b), (d), and (e), and 10 CFR 71.59, "Standards for Arrays of Fissile Material Packages." For packages intended for air transport of fissile material or plutonium, ensure that the application includes analyses that consider the most reactive condition of the package and contents, as determined by the tests in 10 CFR 71.55(f) for fissile material or 10 CFR 71.74 for plutonium. While an applicant may also seek to include credit for residual absorber material in irradiated reactor-control components, the criticality reviewer conducts the review of that credit and is not within the scope of the guidance in this section.

#### *7.4.7.1 Neutron-absorbing (poison) material specification*

For all absorber materials, verify that the application 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 application includes the manufacturer's data sheet to supplement the above information. In the case of absorber plates or sheets, the application should specify the minimum poison content as an areal density (e.g., milligrams of boron-10 per square centimeter).

Qualification testing of neutron-absorber materials is conducted to ensure the following:

- The material used will have sufficient durability (e.g., compatibility with irradiation and elevated temperatures) for the application for which it has been designed.

- The physical characteristics and the uniformity of the distribution of the absorber material or nuclides (e.g., boron-10) are sufficient to meet the design requirements. Materials that have passed the qualification tests should be acceptance tested (see Chapter 9, "Acceptance Tests and Maintenance Program Evaluation," of this SRP) for use in systems to be employed for transportation. Each production run should be acceptance tested.

The NRC considers ASTM C1671-15, with some exceptions, additions, and clarifications, appropriate for staff use in review activities for boron-based absorbers. Attachment 7A to this SRP chapter provides these exceptions, additions, and clarifications. The use of ASTM C1671 is not a regulatory requirement; alternative approaches are acceptable if technically supported.

*7.4.7.2 Computation of percent credit for boron-based neutron absorbers*

This section illustrates one method the materials reviewers use to compute the level of credit allowed for neutron-absorber materials in the criticality safety analysis of packages for transporting 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 NRC has accepted an upper limit of 90-percent credit to be applied to solid absorbers, meaning that the material is computationally modeled as containing only 90 percent of the absorber nuclides shown to be present. The NRC set this limit to account for the uncertainties arising in extrapolating the validation for absorber materials.

Neutron channeling has been shown to occur in an absorber that uses coarse particles of $B_4C$ dispersed in an aluminum matrix. The nonuniformities and channeling effects further limit the poison credit for heterogeneous absorber materials. For heterogeneous absorber materials, 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 the absorber nuclide(s) in the absorber material. Calibration standards should be evaluated at 111 percent (i.e., 1/0.90) of the poison areal 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 75-percent credit is given

If the fractional credit is less than 0.75, the absorber is regarded as unsuitable and should be given no credit. In some cases, where the applicant may seek only a very small fractional credit for the absorber (e.g., 50 percent or less), this amount of credit may be granted with acceptance tests that only ensure proper density and other properties of the absorber in accordance with appropriate standards for fabrication with that absorber material. Such may be the case for unirradiated poison rod assemblies that may need to be inserted with commercial SNF. Coordinate with the criticality reviewer to evaluate such cases.

In order to receive 90-percent credit whether for a homogeneous absorber or a heterogeneous absorber, the presence, uniformity, and effectiveness of the absorber nuclides in the absorber material must be verified by means of a neutron transmission test. Verify that the application demonstrates that the particle sizes of the absorber in the absorber material (e.g., $B_4C$ in a boron-based absorber) are sufficiently fine (diameters on the order of microns) to preclude channeling and nonuniformity effects that occur with absorbers with coarse particles.

*7.4.7.3 Qualifying properties not Associated with attenuation*

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

- Mechanical testing, which ensures 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 B557-06, “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 ksi
  — ultimate strength no less than 5.0 ksi
  — elongation no less than 1 percent

  Alternatively, the staff has accepted bend tests under ASTM E290-14, “Standard Test Methods for Bend Testing of Material for Ductility,” with a 90-degree bend without failure as the passing criteria.

- Porosity measurements, which 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 used for control of porosity is at the discretion of the applicant. The acceptance tests and maintenance program 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.

The qualification of the Boral™ should address the effects of porosity and material passivation on the susceptibility of Boral™ cladding to blistering from hydrogen generation or flash steaming during short-term loading and drying operations.

- 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.

- 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 poison materials such as aluminum MMCs. Verify that the mechanical and thermal tests include tensile testing, impact testing (or $K_{IC}$ measurements), creep testing, and (if applicable) mechanical testing of weldments over a range of temperatures encompassing normal conditions of transport and hypothetical accident conditions. Numerous ASTM testing standards exist for the measurement of mechanical and physical properties of materials. Confirm that the applicant identified and justified the testing standards used for the mechanical and physical properties of the neutron-absorber materials.

Verify that the application indicates that samples of neutron poison material should be examined (i.e., the use of transmission-electron microscopy or scanning-electron microscopy) for the following changes:

- redistribution or loss of the absorber nuclide (e.g., boron in boron-based absorbers)

- dimensional changes (material instability)

- cracking, spalling, or debonding of the matrix from the absorber nuclide-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)

Verify that the application indicates that 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 packages that will be loaded or unloaded in a pool or similar environment, verify that the application indicates that absorber material was evaluated or tested for environmental and galvanic interactions and the generation of hydrogen in the pool environment.  If environmental testing is employed, the test conditions (time, temperature, and number of cycles) 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.  Verify that the evaluation considers the effects of any residual pool water remaining in the container after removal from the pool.  Generally, for common engineering materials, an evaluation based on consultation of a corrosion reference (galvanic series) should suffice for pool loading and unloading situations.

Ensure that the applicant took appropriate measures to assess the strength or ductility of the material, depending on the structural requirements of the application.

Coordinate with the criticality and acceptance tests and maintenance program reviewers to ensure that the acceptance test section of the application includes appropriate qualification and acceptance tests for neutron-absorber materials, as described in this SRP chapter.

### 7.4.8 Corrosion Resistance

The following subsections address specific considerations for commonly used materials for packaging components and systems important to safety that may be exposed to environments where the effects of corrosion should be considered.  Confirm that the applicant has identified materials and package components and assessed the effects of corrosion, chemical reactions, and radiation effects, in accordance with 10 CFR 71.35(a) and 10 CFR 71.43(d).  In addition to material selection, the application may use other corrosion-control measures, provided that adequate documentation is supplied to demonstrate efficacy.  For example, coatings may be specified to alleviate atmospheric corrosion issues.  However, unless supporting data are available to demonstrate the predicted coating life, the coating should be periodically inspected and maintained.  Verify that the application addresses maintenance in the acceptance tests and maintenance program for coatings relied on for preventing corrosion of packaging components, to ensure unimpaired physical condition, in accordance with 10 CFR 71.87(b).

For components that have been previously in service under a 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,” storage license (e.g., dual-purpose cask systems, transportable storage canisters for commercial SNF), evaluate the cumulative effects of corrosion during storage and transportation on the ability of the package to fulfill its important-to-safety functions under normal conditions of transport and hypothetical accident conditions.  During the storage term, these components may have been exposed to a variety of environments associated with content loading, drying, inerting, container transfer, storage during the initial license, and renewed storage during a period of extended operation.  Refer to NUREG-2215 and NUREG-2214, “Managing Aging Processes in Storage (MAPS) Report,”  for additional detail on corrosion processes relevant to commercial SNF storage systems in the initial and renewed storage terms, respectively.  The corrosion of components that have been in service under a renewed storage license likely is addressed by an NRC-approved aging management program.  Evaluate whether storage aging management programs and other maintenance activities should be augmented with pre-transportation inspections and tests to ensure important-to-safety functions are fulfilled during transportation.

*7.4.8.1 Environments*

The corrosion rates of materials 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 components of packaging that are important to safety.

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 (e.g., Fontana and Greene 1978; Graver 1985; Revie and Uhlig 2008; Revie 2000; ASM 2000). 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 (NASA) 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.

Evacuating the transportation package and backfilling with an inert gas such as helium will significantly reduce the water content, humidity, and oxidizing potential of the environment. The inert low humidity inside the backfilled transportation package will significantly decrease the uniform corrosion rate of carbon steel as well as reduce the potential for localized corrosion of passive alloys such as stainless steels.

*7.4.8.2 Carbon and low-alloy steels*

Corrosion rates for carbon and low-alloy steels are dependent on the exposure environment. Corrosion rates for these materials may be found in the corrosion references discussed in Section 7.4.8.1 of this SRP chapter.

For packaging components and systems important to safety that are constructed from carbon or low-alloy steels, control measures may be employed to reduce the loss of material as a result of corrosion. For example, coatings may be specified to prevent atmospheric corrosion. However, as described in greater detail in Section 7.4.9 of this SRP chapter, such coatings should be periodically inspected and maintained. Verify that the application addresses coating inspection and maintenance in the acceptance tests and maintenance program for any coatings that are relied upon for preventing corrosion of packaging, components, and systems important to safety.

*7.4.8.3 Austenitic stainless steel*

When stainless steel is used for transportation packages, the primary concern is not general 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. Localized corrosion and chloride-induced stress corrosion cracking (CISCC) of stainless steel components exposed to marine environments have been observed at operating reactors (NRC 2012). 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.

Sensitization of austenitic stainless steels is caused by thermal exposures that result in the formation of carbides at grain boundaries that deplete the concentration of chromium in the grain-boundary region. The chromium-depleted grain-boundary regions are more susceptible to corrosion, particularly intergranular corrosion and intergranular stress-corrosion cracking. Sensitization of austenitic stainless steels during fabrication can be avoided by specifying low carbon stainless steel grades (including welding consumables).

For transportation packaging that may be susceptible to localized corrosion or CISCC, verify that the system maintenance and operating procedures address the potential for degradation.

### 7.4.9 Protective Coatings

Coatings in transportation packages 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 the transportation package internal components. 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 for the coating. Confirm that the applicant has identified coating materials package components coated and has assessed the effects of corrosion, chemical reactions, and radiation effects, as required by 10 CFR 71.35(a) and 10 CFR 71.43(d).

The NRC established this section of this SRP to alleviate confusion regarding coatings for transportation package components. Use discretion in implementing the detailed review guidance in this section. This section outlines methods and procedures for appropriately assessing coatings. The assessment covers several areas in detail, including the scope of the coating application, type of coating system, surface-preparation methods, applicable coating-repair techniques, and coatings qualification testing.

#### *7.4.9.1 Review guidance*

Verify the appropriate application of the coating(s) by reviewing the coating specifications. 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 for the transportation package design.

#### *7.4.9.2 Scope of coating application*

Verify that the coating specification identifies the purpose of the coating, lists the components to be coated, and describes the expected environmental conditions (e.g., expected conditions during loading, unloading, transportation, and dry storage of commercial SNF packages that have been in dry storage or have components that have been in dry storage).

Verify that the coatings will not react with the package internal components and contents and will remain adherent and inert when the transportation package is exposed to the various environments during transportation and loading and unloading operations.

#### *7.4.9.3 Coating selection*

Verify that the coating specification identifies the manufacturer's name, the type of primers and topcoat used in the coating system, and the minimum and maximum dry coating thickness. 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 transportation package components is capable of withstanding the intended service conditions during transportation, loading, and unloading activities and the regulatory tests conditions. 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 for surface preparation, coating application, and coating repairs.

#### *7.4.9.4 Coating qualification testing*

Any coating (including paints or plating) used for a transportation package must have been tested to demonstrate the coatings performance under all conditions of loading and transportation, including the regulatory test conditions. The conditions evaluated should include exposure to radiation, unloading, and transfer operations.

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

### 7.4.10 Content Reactions

Review the materials and coatings of the transportation package to verify that they will not produce significant chemical or galvanic reactions among packaging contents or between the packaging components and the packaging contents. Confirm that the applicant has identified the contents of the package in accordance with 10 CFR 71.33(b); demonstrated that the package meets the requirements of 10 CFR 71.35(a); and assessed the effects of corrosion, chemical reactions, and radiation in accordance with 10 CFR 71.43(d).

Verify that the applicant has provided an adequate description of the contents such that the stability and compatibility with the packaging components can be fully evaluated. Key parameters include the environment inside the packaging to which the contents are exposed, including requirements for dryness or use of inert gases, physical and chemical form (e.g., activated metal, process waste), the geometric form (e.g., particulates, bulk solid), the maximum quantity of radioactive materials to be transported, and the radionuclide inventory.

#### *7.4.10.1 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, confirm that the application demonstrates that measures are taken to remove moisture or oxygen from the container, such as through vacuum or inerting. Liquid contents that contain water may be subject to water radiolysis, producing a flammable mixture of hydrogen and oxygen. Ensure that the applicant considered the potential for content materials, such as polymers, to decompose when exposed to heat and radiation, which may generate the moisture to support pyrophoricity as well as produce flammable hydrogen and oxygen mixtures. Coordinate with the containment and thermal reviewers to assess the potential for flammable gas generation.

In addition, hydrogen or other flammable gases may be generated during wet loading and unloading operations. Verify that the operating procedures for wet loading and unloading operations contain measures for detecting the presence of hydrogen and preventing the ignition of combustible gases during package loading and unloading operations. The Package Operations section of the application should include these procedures.

NRC Bulletin 96-04, "Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks," documents known operational issues associated with hydrogen generation. This bulletin describes a case where a zinc coating on a canister interior reacted with borated spent fuel 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 among the canister content materials, fuel payload, and the operating environments.

*7.4.10.2 Content chemical reactions, outgassing, and corrosion*

For metallic components of the package that may come into physical contact with one another, confirm that the application considers the possibility of eutectic reactions since such reactions can lead to melting at the interface between the metals at a lower temperature than the melting points of the metals in contact. Such interactions may occur with depleted uranium, lead, or aluminum in contact with steel. If applicable, verify that the applicant has evaluated the potential formation of, and has employed methods to prevent, eutectic reactions.

Ensure that the applicant considered the potential for outgassing of the contents and components in the evaluation of the maximum operating pressure. Outgassing may originate from moisture retained in wood used for dunnage or contaminated sources. Polymers and greases may also outgas under vacuum or at elevated temperatures. NASA has published a data compilation of outgassing data on a wide range of materials (Campbell and Scialdone 1993). NASA-developed testing led to the development of ASTM E595, "Total Mass Loss (TML) and Collected Volatile Condensable Materials (CVCM) from Outgassing in a Vacuum Environment." Verify that the applicant used standard test methods such as ASTM E595 for outgassing data provided by a material vendor.

Corrosive reactions between the contents and the internal environment, as well as reactions between the contents and the package components, may degrade structural integrity and containment. Verify that the applicant demonstrated that corrosion wastage will not lead to a loss of intended functions.

For nonfuel hardware contents in commercial SNF packages, the NRC has previously reviewed a number of hardware components and materials to ensure that there are no significant chemical, galvanic, or other reactions as a result of exposure of these various contents to the wet loading and the package's internal environment. These include components encased in stainless steel and aluminum alloys such as neutron-source assemblies, burnable poison rod assemblies, thimble-plug devices, and other types of control elements. The NRC has found the following components to be acceptable for transportation when the canister is constructed of stainless steel with stainless steel and aluminum basket components:

- neutron-source materials encased in stainless steel or zirconium alloy cladding containing antimony-beryllium, americium-beryllium, plutonium-beryllium, polonium-beryllium, and californium

- control elements encased in zircaloy or stainless steel cladding containing $B_4C$, borosilicate glass, silver-indium-cadmium alloy, or thorium oxide

Ensure that the applicant evaluated any nonfuel hardware components with damaged cladding that exposes the contents such as a burnable poison material or neutron source on a case-specific basis.

### 7.4.11 Radiation Effects

Exposure of materials to radiation can cause microstructural changes that alter mechanical properties and reduce resistance to environmentally induced degradation such as stress corrosion cracking. The effect of radiation exposure is dependent on several factors, primarily the material composition, the type of radiation, and the duration of radiation exposure. Polymeric materials are affected by gamma radiation. Metals and alloys are generally resistant to gamma radiation but are affected by neutron radiation. Confirm that the applicant demonstrated that the package meets the requirements of 10 CFR 71.35(a) and assessed the effects of radiation in accordance with 10 CFR 71.43(d). The following paragraphs provide a brief summary of radiation effects on commonly used materials in transportation packaging systems. Review the references in the following paragraphs for more detailed information.

For alloy steels, measurable changes to mechanical properties are not observed with a neutron fluence below $10^{17}$ n/square centimeter ($cm^2$) [$6.5\times10^{17}$ n/square inch ($in^2$)] (10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix H, "Reactor Vessel Material Surveillance Program Requirements"). Nikolaev et al. (2002) and Odette and Lucas (2001) reported that neutron fluence levels greater than $10^{19}$ n/$cm^2$ [$6.5\times10^{19}$ n/$in^2$] have been found to be required to produce measurable degradation of mechanical properties including increased tensile and yield strength and decreased toughness.

For stainless steels, neutron irradiation can cause changes in stainless steel mechanical properties such as loss of ductility, fracture toughness, and resistance to cracking (Was et al. 2006). Gamble (2006) found that neutron fluence levels greater than $1\times10^{20}$ n/$cm^2$ [$6.5\times10^{20}$ n/$in^2$] are required to produce measurable degradation of the mechanical properties. Caskey et al. (1990) also indicate that neutron fluence levels of up to $2\times10^{21}$ n/$cm^2$ [$1\times10^{22}$ n/$in^2$] were not found to enhance stress-corrosion cracking susceptibility.

Farrell and King (1973) reported the effects of neutron irradiation on aluminum alloys and showed that fluences greater than $10^{20}$ n/$cm^2$ [$6.5\times10^{20}$ n/$in^2$] were necessary to have marked increases in yield or tensile strengths or a decrease in measured ductility.

Radiation exposure is known to cause changes in physical properties of polymers and elastomers (NASA 1970; Bruce and Davis 1981; Lee 1985; Battelle 1961). Bruce and Davis (1981) summarized the lowest reported threshold exposures for material properties of a number of organic materials used in nuclear power plants. The threshold for degradation of natural rubber occurs when the dose reaches $2\times10^4$ grays (Gy) [$2\times10^6$ rads]. Butadiene, nitrile, and urethane rubber have a threshold of $10^4$ Gy [$10^6$ rads]. Fluoroelastomers have a reported threshold dose of $10^3$ to $10^4$ Gy [$10^5$ to $10^6$ rads]. Some fluoropolymers such as tetrafluoroethylene have been shown to be susceptible to radiation damage at a dose of 200 Gy [$2\times10^4$ rads] (NASA 1970).

Coordinate with the shielding reviewer to determine the neutron-fluence rate or the gamma-dose rate, as applicable, for the different package components. Verify that the applicant appropriately considered any damaging effects of radiation on the transportation package materials. These effects may include degradation of seals, sealing materials, coatings, adhesives, and structural materials. Verify that the package operations and package maintenance program descriptions assure the maintenance or replacement of components susceptible to radiation damage before attaining a neutron fluence or gamma dose that degrades the components' performance.

### 7.4.12 Package Contents

Ensure that the application provides an adequate description of the chemical and physical form of the package contents (e.g., canistered vitrified high-level waste, radiation sources). Confirm that the applicant has identified the contents of the package in accordance with 10 CFR 71.33(b); demonstrated that the package meets the requirements of 10 CFR 71.35(a); and assessed the effects of corrosion, chemical reactions, and radiation effects in accordance with 10 CFR 71.43(d). Assess if there are materials and other properties of the contents (e.g., that lead to corrosion, radiolysis, and hydrogen generation) that may affect the intended functions of the package during normal conditions of transport and hypothetical accident conditions, as discussed in Sections 7.4.10 and 7.4.11 of this SRP chapter. Coordinate with other reviewers as needed to understand the contents properties in addition to the physical properties that may affect package intended functions. See the section in Attachment 7A to this SRP relevant to the package and contents type under review for guidance regarding concerns unique to that package and contents type. For SNF packages, refer to Section 7.4.14 of this SRP chapter for guidance unique to SNF contents.

### 7.4.13 Fresh (Unirradiated) Fuel Cladding

Confirm that the mechanical properties of the cladding materials are adequate to ensure that the fresh (unirradiated) fuel remains in the configuration analyzed in the application, in accordance with the requirements of 10 CFR 71.35(a). In addition, confirm that the applicant has identified the contents of the package, in accordance with 10 CFR 71.33(b).

Ensure that the structural evaluation is bounding to all cladding alloys in the allowable contents (i.e., Zircaloy-2, Zircaloy-4, ZIRLO™, M5®). Verify that the application 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) and cladding temperature.

Preferred sources of cladding materials data include standards and codes (e.g., ASTM B351-13/B351M); manufacturer's test data obtained under an approved quality assurance program; NRC-approved topical reports; staff-accepted technical reports; and peer-reviewed articles, research reports, and texts. Ensure that the application adequately justifies the applicability and acceptability of any source of information.

Multiple aluminum alloys have been used for aluminum clad fuel including: 1100, 5052, 5456, 6061, and 8001. The mechanical properties of these alloys are dependent on the heat treatment used in material production. Ensure that the mechanical properties of these cladding alloys are based on manufacturer-provided data. Mechanical properties of many aluminum alloys as a function of temperature are included in ASME B&PV Code Section II Part D.

Types 304, 304L, and 348 stainless steels were originally used as nuclear fuel cladding and were replaced by zirconium alloys starting in the 1960s. Specific information on the fuel designs; physical properties of the stainless steel cladding materials; and mechanical properties, including those of the irradiated stainless steel cladding, are described in Electric Power Research Institute (EPRI) Report NP-2642.

### 7.4.14 Spent Nuclear Fuel

Confirm that the mechanical properties of the cladding materials are adequate to ensure that the SNF remains in the configuration analyzed in the application over the ranges of conditions associated with the tests in 10 CFR 71.71 and 10 CFR 71.73. In addition, confirm that the applicant has identified the contents of the package in accordance with 10 CFR 71.33(b). The review guidance in this section for commercial power plant operations addresses the transport of all SNF of burnups the NRC currently licenses. Applications with burnup levels exceeding those the Office of Nuclear Reactor Regulation (NRR) licensed, or for cladding materials NRR did not license, may require additional justifications.

#### *7.4.14.1 Spent fuel classification*

Verify that the application and the certificate of compliance (CoC) identify the allowable SNF contents and condition of the assembly and rods (i.e., intact, undamaged or damaged fuel—refer to the SRP Glossary).

Verify that the applicant considered whether the material properties of the SNF assemblies can be altered during prior dry storage. If this alteration is significant enough to prevent the fuel or assembly from performing its intended functions during transport, then ensure that the fuel assembly is classified as damaged.

Ensure that the application discusses all of the following conditions to support whether the SNF (rods and assembly) to be loaded is intact or undamaged:

- the acceptable physical characteristics of the SNF (i.e., acceptable assembly defects and cladding breaches)
- the intended functions the applicant has imposed on the SNF for demonstrating compliance with fuel-specific and package-related regulatory requirements
- the alteration and degradation mechanisms of the SNF during transport (or during prior dry storage) that could credibly compromise the ability to meet fuel-specific or package-related functions
- discussions or analyses demonstrating that the mechanisms in the immediately preceding bullet will not reasonably affect the physical characteristics of the SNF (as defined in the first bullet) or result in reconfiguration beyond the safety analyses in the application

Recognize that SNF assemblies with any of the following characteristics, as identified during the fuel-selection process (see Attachment 7B to this SRP chapter), are expected to be classified as damaged, unless the applicant provides an adequate justification:

- There is visible deformation of the rods in the SNF assembly. 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 or rods do not adversely affect the structural performance of the assembly, radiological safety, 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 SNF assembly has missing, displaced, or damaged structural components resulting in the following:

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

  — The structural performance of the assembly may be compromised during normal conditions of transport or under hypothetical accident conditions.

- Reactor operating records or fuel-classification records indicate that the SNF assembly contains fuel rods with gross breaches.

- The SNF assembly 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, as long as 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 package-related regulations.

The NRC 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 centimeters [0.43 inches] in diameter in 15x15 pressurized-water reactor (PWR) assemblies. Pellets from a boiling-water reactor (BWR) are somewhat larger, and those from 17x17 PWR assemblies are somewhat smaller. In general, a pellet's length is slightly longer than its diameter. During the first cycle of irradiation in-reactor, the pellet fragments into 25 to 35 smaller interlocked pieces, plus a small amount of finer powder, from pellet-to-pellet abrasion. When the rod breaches, about 0.1 gram [0.003 ounce] of this fine powder may be carried out of the fuel rod at the breach site (NRC 1981). Modeling the fragments as either spherical- or pie-shaped pieces indicate that a cladding-crack width of at least 2 to 3 millimeters [0.08 to 0.11 inch] would be required to release a fragment. Hence, gross breaches should be considered to be any cladding breach greater than 1 millimeter.

*7.4.14.2 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 transportation package containment (or canister for canister-based packages); that is, the safety analyses rely on the integrity of the fuel cladding for maintaining the analyzed configuration.

Cladding Alloys

Identify the specific cladding alloys (e.g., Zircaloy-2, Zircaloy-4, ZIRLO™, M5®, Aluminum 1100, Type 304 Stainless Steel) and maximum burnup of the SNF to be stored. The NRC considers the peak rod average burnup as an appropriate measure of maximum fuel burnup in the materials evaluation. Ensure that the fuel and cladding alloy contents are consistent with the technical bases in the structural evaluation.

Determine if the SNF to be stored includes boron-based integral fuel burnable absorbers. Note that these rods have the potential to increase the fuel rod internal pressure from decay-gas generation (helium), which should be considered when evaluating the consequences of aging mechanisms during dry storage before transport, particularly for dry storage periods beyond 20 years. Note also that decay gases are not generated in rods with gadolinium-based integral fuel burnable absorbers, which will not result in increased rod pressures beyond those the fuel fission products generate.

Zirconium Alloy 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®, Aluminum 1100, Type 304 Stainless Steel). Verify that the application provides a justification that the cladding mechanical properties are bounding upon consideration of alloy type, fabrication process (cold work stress relieved annealed, recrystallized annealed), hydrogen content, neutron fluence (burnup), oxide thickness, and cladding temperature.

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 at elevated temperatures that may be reached during loading and drying 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; and peer-reviewed articles, research reports, and texts. Ensure that the application adequately justifies applicability and acceptability of any source of information.

While the NRC deems acceptable the mechanical property models from PNL-17700, "PNNL Stress/Strain Correlation for Zircaloy," issued July 2008 (Geelhood et al. 2008), for previous licensing and certification actions, note that 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, ensure that the applicant referred to other references for defining bounding mechanical properties for M5® cladding. Limited, nonproprietary data are available for M5® cladding, such as the publicly available data from the French Competent Authority (Institut de Radioprotection et de Sûreté Nucléaire). Ensure that the application justifies 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 to ensure that there is adequate safety margin in the respective vibration and drop analyses to ensure that the assumed properties are adequate. Consider using engineering judgment from the staff's findings on previous NRC-approved topical reports.

Confirm that the application 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 application 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 drop analysis.

Recognize also that the models PNL-17700 references only account for mechanical properties of cladding with circumferential hydrides. The NRC 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. 2015). However, based on static bend testing of cladding with a high density of radial hydrides discussed elsewhere, the staff considers these mechanical properties adequate for the design-basis drop scenarios during normal conditions of transport and hypothetical accident conditions. Additional considerations for the certification of transportation packages containing high-burnup fuel are provided in a separate technical report.

<u>Effective Zirconium Alloy Cladding Thickness</u>

*Cladding Oxidation*

The structural evaluation should account for the reduced effective thickness of the cladding from 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. 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 application defines an effective cladding thickness that is reduced by a bounding oxide layer to the specific cladding contents to be transported. 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 NRC staff finds appropriate. In 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, the staff determined that the 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.

*Hydride Rim*

During reactor irradiation, some of the hydrogen generated from waterside 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 related to the texture of the 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 of the cladding wall.

Recognize that 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, ring compression test results from Argonne National Laboratory (ANL) indicate that for the range of gas pressures anticipated during drying, storage, and transportation, 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, as long as mechanical test data referenced in the structural evaluation has adequately accounted for its presence. Historically, this has been the case during the review of the transportation package, 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 without 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 previous consideration to treat the rim as wastage to be unnecessary when calculating the effective cladding thickness, as the hydride rim has been properly accounted for in the mechanical property models.

Drying Adequacy

Evaluate the descriptions related to draining and drying of the containment cavity or, for canister-based packages, the canister cavity of the transportation package during SNF loading operations, as discussed in the Operating Procedures section of the application. More specifically, assess whether the procedures used for removing water vapor and oxidizing material to an acceptable level are appropriate.

The NRC staff have 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," issued November 1987 (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 gases (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 247 cubic foot] cask gas volume at a pressure of about 0.15 megapascal (MPa) [1.5 atmosphere (atm)] at 300 °Kelvin (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 of water) are expected if adequate vacuum drying is performed.

If methods other than vacuum drying are used (such as forced helium recirculation), ensure that the application provides additional analyses or tests to sufficiently justify that moisture and impurity levels of the fuel cover gas will prevent unacceptable cladding degradation. The procedures should reflect the potential for blockage of the evacuation system or masking of defects in the cladding of nonintact 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 containment (or canister) cavity of the transportation package because of decay heat from the fuel. A staged drawdown or other means of preventing ice blockage of the package evacuation path may be used (e.g., measurement of package (or canister) pressure not involving the line through which the package (or the package's canister) 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 package (or package canister). The process should provide for repetition of the evacuation and repressurization cycles if the containment cavity of the transportation package is opened to an oxidizing atmosphere following the evacuation and repressurization cycles (as may occur in conjunction with seal repairs). Refer to NUREG-2215, Appendix 8C, "Fuel Oxidation and Cladding Splitting," for additional considerations on cladding oxidation and splitting.

<u>Maximum (Peak) Zirconium Alloy Cladding Temperature</u>

Ensure that the calculated maximum (peak) cladding temperature for the SNF during normal conditions of transport and short-term loading operations (i.e., loading, drying, backfilling with inert gas) does not exceed 570 °C [1,058 °F] for low-burnup fuel, or 400 °C [752 °F] for high-burnup fuel. These temperature limits were defined based on accelerated separate-effects testing to provide reasonable assurance that thermal creep and hydride reorientation will not compromise the integrity of the cladding. Furthermore, previous review guidance called on applicants to justify that the cladding hoop stresses of low-burnup fuel remained below 90 MPa for peak cladding temperatures between 400 and 570 °C [752 and 1,058 °F]. The cladding hoop stress limit of 90 MPa was meant to provide reasonable assurance that hydride reorientation would be limited in low-burnup fuel for the higher-peak cladding temperatures. However, research on hydride reorientation over the past 15 years has provided evidence that hydride reorientation is expected to be minimal in low-burnup fuel because of insufficient hydrogen content and cladding hoop stresses. Therefore, the application is not expected to contain a justification of a cladding hoop stress limit for low-burnup fuel up to peak cladding temperatures of 570 °C [1,058 °F].

If the application proposes the transport of high-burnup fuel that may have experienced a peak cladding temperature exceeding 400 °C [752 °F], ensure that the application provides additional justification that evaluates the consequences of the increased temperature on all credible mechanisms that may affect fuel performance, including aging mechanisms during prior dry storage (e.g., creep, hydride reorientation, delayed hydride cracking). For hypothetical accident conditions, the maximum cladding temperature for all burnups should not exceed 570 °C [1,058 °F].

Coordinate with the thermal reviewer to verify that the calculated maximum cladding temperature is based on 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 contents. Also confirm that the thermal models (and associated uncertainties) used for calculating cladding temperatures are acceptable to the thermal reviewer.

Thermal Cycling of Zirconium Alloy Clad Fuel during Drying Operations

Review the fuel-loading procedures to ensure that any repeated thermal cycling (repeated heatup and cooldown cycles) during loading operations of fuel is limited to fewer 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. The reviewer should 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.

Note that the applicant may use mechanical properties of cladding accounting for reoriented hydrides in the structural evaluation of the SNF assembly. 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-basis heatup and cooldown cycles.

Cover Gas

Verify that the application defines the composition of the cover gas for the fuel during transport. Once the fuel rods are placed inside of the containment cavity (or canister cavity) of the transportation package 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. 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 requirements in 10 CFR 71.55(d)(2), 10 CFR 71.43(f), and 10 CFR 71.51(a). The configuration of the fuel must remain bounded by the reviewed safety analyses. Further, the release of fuel fines or grain-sized powder from ruptured fuel into the containment (or canister) cavity may be a condition outside the design basis for the package 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. This can be done by a review of records (for example, shipping records) or 100 percent eddy current inspection of assemblies. Note that inspection of rods by either eddy current or visual inspection, to the extent needed to ensure there are no pinholes or hairline cracks, is difficult, time consuming, and subject to error.

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-basis 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 the applicant chose option 3, coordinate with the thermal reviewer to determine whether the operating procedures (see Chapter 8, “Operating Procedures Evaluation,” of the SRP) include an adequate analysis of the potential for cladding splitting should fuel rods be exposed to an oxidizing gaseous atmosphere.

Fuel oxidation and cladding splitting conservatively follow Arrhenius time-at-temperature behavior.  For fuel burnups not exceeding 45 gigawatt-days per metric tons of uranium and Zircaloy cladding, use the current time-at-temperature curves for uranium-based fuel (e.g., Einziger and Strain 1986) to determine the allowable exposure duration on an oxidizing atmosphere for a given design-basis 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 8D to NUREG-2215 for additional information on cladding oxidation and splitting.

<u>Release Fractions (Nonleaktight Packages)</u>

Coordinate with the containment reviewer to ensure that the applicant has provided adequate release fractions for the proposed fuel contents if the package containment is nonleaktight.  Additionally, coordinate with the structural or containment reviewer on potential consequence assessment during hypothetical accident conditions using release fractions.  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.  Verify that the collection method the applicant used for quantification of the release fractions is sophisticated enough to gather respirable release fractions.

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, “High-Burnup Fuel,” to NUREG/CR-7203, “A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages,” issued September 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 application may provide different release fractions (CRUD, fission gases, volatiles, and fuel fines) for low- and high-burnup fuel in nonleaktight containment.

<u>Aluminum Alloy Clad Spent Fuel</u>

Research reactor fuel assemblies typically use aluminum alloy cladding materials.  Pitting corrosion of aluminum cladding during wet storage has been noted at the Savannah River Site (SRS).  Several factors are believed to have played the most important role in the corrosion of aluminum-clad SNF in the reactor basins at SRS, including water conductivity and chemistry, cladding scratches and imperfections, and galvanic coupling of the cladding and stainless steel components (Howell 1999).  Peacock et al. (1995) evaluated corrosion aluminum clad fuels in dry storage by using aluminum atmospheric corrosion data extrapolation to 50 years.  The corresponding thickness of metal consumed after 50 years for 1100, 5052, and 6061 aluminum alloys was determined to be 11, 19, and 12 microns [$4.3\times10^{-4}$, $7.4\times10^{-4}$, and $4.7\times10^{-4}$ inch] at 150 °C [302 °F] and 33, 76, and 30 microns [$1.2\times10^{-3}$, $3.0\times10^{-3}$, and $1.2\times10^{-3}$ inch], at 200 °C [392 °F], respectively. For a cladding with a thickness of 762 microns [0.030 inch], this represents a decrease in thickness from corrosion of less than 2.5 percent at 150 °C [302 °F] and less than 10 percent at 200 °C [392 °F]. Based on this evaluation, degradation of aluminum cladding in dry storage is expected to be minimal.

Vinson et al. (2010) developed a methodology to evaluate containment of aluminum-clad SNF, even with severe cladding breaches, for transport. The containment analysis methodology for aluminum-clad SNF, including severely breached fuel, was developed in accordance with the methodology provided in ANSI N14.5 and adopted in NUREG/CR-6487, "Containment Analysis for Type B Packages Used to Transport Various Contents," issued November 1996, to meet the requirements of 10 CFR Part 71. The analysis by Vinson et al. (2010) used a radionuclide inventory developed for the case of fuel from the RA-3 research reactor using conservative estimates of the fuel area exposed by cladding breaches based upon records from the visual examination of the fuel and the containment criterion for Type B packages. The containment analysis of the RA-3 fuel indicates that the SNF can be transported in a Type B package with a leak rate of $1.0\times10^{-6}$ atm·cubic meters per second and maintained within the allowable release rates under normal conditions of transport and hypothetical accident conditions. Coordinate with the containment reviewer that an application's content and conditions are similar to those described in Vinson et al. (2010).

Stainless Steel Clad Spent Fuel

Types 304, 304L, and 348 stainless steels were originally used as nuclear fuel cladding and were replaced by zirconium alloys starting in the 1960s. The change from stainless steel to zirconium alloy cladding was driven by economic considerations and the performance of stainless steel materials in BWRs. EPRI reports NP-2119 and NP-2642 (EPRI 1981; 1982) describe the analyses of stainless steel cladding failures in reactor operations. Information on the physical properties and mechanical properties of irradiated stainless steel cladding materials and the operational history of reactors using stainless steel cladding are included in EPRI Report NP-2642 (EPRI 1982). Verify that the application includes an assessment of the material properties for any stainless steel clad SNF.

#### *7.4.14.3 Canned spent fuel*

SNF that has been classified as damaged for transportation should be placed in a can designed for damaged fuel or in an acceptable alternative. The purpose of a can designed for damaged fuel in transportation is to (i) confine gross fuel particles, debris, or damaged assemblies to a known volume within the transportation package; (ii) demonstrate that compliance with the criticality, shielding, thermal, and structural requirements are met; and (3) permit normal handling and retrieval from the transportation package. 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 of 10 CFR 71.31(a)(2) and 10 CFR 71.35, "Package Evaluation."

The configuration of the fuel inside the fuel can is generally not restricted; therefore, ensure that the applicant performed 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. The mechanical properties of the fuel can should also be adequate for demonstrating adequate structural performance to ensure that the geometric form of the package contents will not be substantially altered during normal conditions of transport and hypothetical accident conditions. Consult with the containment reviewer when evaluating the damaged fuel can design.

### 7.4.15 Bolting Material

If threaded fasteners are employed as components of packaging important to safety, verify that the bolt material(s) have adequate resistance to corrosion and a coefficient of thermal expansion similar to the materials being bolted together. Confirm that the applicant has identified the materials used in bolted connections in accordance with 10 CFR 71.33(a)(5); demonstrated that the package meets the requirements of 10 CFR 71.35(a); and assessed the effects of corrosion, chemical reactions, and radiation effects on the bolting materials, in accordance with 10 CFR 71.43(d). Threaded inserts are commonly used to prevent galling of threaded fasteners. Bolts should have resistance to brittle fracture over the range of possible exposure conditions. Examine the use of bolts manufactured from precipitation-hardened stainless steels such as ASTM A564 Grade 630 (17-4 PH stainless steel) and verify that the thermal treatment specified provides adequate resistance to brittle fracture at low temperatures (Slunder et al. 1967). At temperatures above 316 °C [600 °F] some precipitation-hardened stainless steels can become embrittled (Clarke 1969). Verify that the application considers microstructural changes as a result of elevated temperature exposures in the evaluation of bolt performance. Verify that the applicant has evaluated and determined that the fasteners have adequate creep resistance under normal conditions of transport and hypothetical accident conditions temperature conditions in accordance with the testing requirements of 10 CFR 71.71 and 10 CFR 71.73.

Guidance on closure bolts for transportation packages is available in NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks,” issued April 1991. Coordinate with the structural reviewer to verify that all bolts have the required tensile strength, resistance to creep and brittle fracture, and a coefficient of thermal expansion that is similar to the materials being bolted together. Also verify that the bolting material and any internally threaded components have adequate resistance to general and localized corrosion and galvanic corrosion considering the range of operating conditions. Verify that the bolting materials are not sensitive to stress corrosion cracking under anticipated operating conditions, including loading and unloading.

### 7.4.16 Seals

Applicants for transportation package designs generally rely on data from seal manufacturers to define seal properties. Verify that the specified material properties are adequate for the application and consider the range of operating temperatures and environments for normal conditions of transport and hypothetical accident conditions. Confirm that the applicant has identified the materials used in seals in accordance with 10 CFR 71.33(a)(5); demonstrated that the package meets the requirements of 10 CFR 71.35(a); and assessed the effects of corrosion, chemical reactions, and radiation effects on the seal materials, in accordance with 10 CFR 71.43(d) and (f). Verify that inspection and maintenance for the package gasket or seal required by 10 CFR 71.87(c) considers the potential for radiation-induced degradation of the gasket or seal material and identifies appropriate replacement intervals.

*7.4.16.1 Metallic seals*

Metallic seals constructed of an inner spring and outer cover are frequently specified for high-temperature applications. Nickel-based alloys are often used for the spring material because of their 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. If the application indicates that aluminum-faced seals are used, verify that the design includes provisions to prevent corrosion, as aluminum-faced seals have been observed to fail from corrosion in SNF storage systems (NRC 2013).

*7.4.16.2 Elastomeric seals*

Seals for industrial applications may be manufactured from a wide variety of elastomeric materials. Seals on transportation packages for radioactive materials have specific performance requirements and will likely be exposed to unique environments compared to other industrial applications. Consult with the containment reviewer to assess elastomeric seal properties for transportation packages.

For elastomeric O-rings and seals, verify that the application identifies required specifications (e.g., ASTM) for material and mechanical properties. For example, physical characteristics of butyl rubber containment O-ring seals and sealing washers may specify ASTM D2000, which includes specific ASTM tests to determine mechanical properties such as durometer tensile strength and elongation, heat resistance, compression set, cold temperature resistance, and cold temperature resiliency. Verify that O-ring seals will not reach their maximum operating temperature limit. Also verify that the application demonstrates 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. Commonly used elastomeric seal and O-ring materials include ethylene propylene, butyl rubber (isobutylene, isoprene rubber), and Viton™ (synthetic rubber and fluoropolymer elastomer).

Elastomeric seals may be susceptible to thermal- and radiation-induced aging (hardening). The effect of radiation on elastomeric and polymeric materials is discussed in Section 7.4.11 of this SRP chapter. Compare the radiation exposure from the operating environment to published information on the effect of radiation on elastomeric and polymeric materials (e.g., NASA 1970; Bruce and Davis 1981; Lee 1985; Battelle 1961). The seal manufacturer can generally provide guidance on radiation or thermal resistance. Verify that the applicant has included inspection of seals for damage and specified minimum seal replacement intervals as part of the operating procedures.

Verify that the applicant’s selection of elastomeric seal materials considered the effects of permeability on leakage rate. Some seal materials, such as silicone and fluorosilicone elastomers, can have a much higher permeability compared to natural or synthetic rubbers or other elastomers. Review gas permeability data for common elastomeric seal materials that have been tabulated (Parker Hannifin Corporation 2007; Pickett and Lemcoe 1962) or that can be obtained from the seal manufacturer.

## 7.5 Evaluation Findings

Prepare evaluation findings upon satisfaction of the regulatory requirements in Section 7.3 of this SRP chapter. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F7.1 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.33. The applicant described the materials used in the transportation package in sufficient detail to support the staff's evaluation.

F7.2 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.31(c). The applicant identified the applicable codes and standards for the design, fabrication, testing, and maintenance of the package and, in the absence of codes and standards, has adequately described controls for material qualification and fabrication.

F7.3 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a). The applicant demonstrated effective materials performance of packaging components under normal conditions of transport and hypothetical accident conditions.

F7.4 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.85(a). The applicant has determined that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce the effectiveness of the packaging.

F7.5 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(d), 10 CFR 71.85(a), and 10 CFR 71.87(b) and (g). The applicant has demonstrated that there will be no significant corrosion, chemical reactions, or radiation effects that could impair the effectiveness of the packaging. In addition, the package will be inspected before each shipment to verify its condition.

F7.6 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a) for Type B packages and 10 CFR 71.55(d)(2) for fissile packages. The applicant has demonstrated that the package will be designed and constructed such that the analyzed geometric form of its contents will not be substantially altered and there will be no loss or dispersal of the contents under the tests for normal conditions of transport.

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

> Based on review of the statements and representations in the application, the NRC staff concludes that the materials used in the transportation package design have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

## 7.6 <u>References</u>


10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

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."

American Institute of Steel Construction, *Manual of Steel Construction, 9th Edition*, 1989.

American Society for Metals (ASM) International, "ASM Metals Handbook Desk Edition," p 54, 2nd Edition, J. R. Davis Editor, Materials Park, OH: ASM International, 1998.

ASM International, "ASM Handbook - Volume 13 Corrosion," Materials Park, OH: ASM International, 2000.

American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, 2017.
- Section I, "Power Boilers."
- Section II, "Materials."
- Section III, "Rules for Construction of Nuclear Facility Components."
  - Division 1, "Metallic Components"; Subsection NB through NH and Appendices
  - Division 3, "Containments for Transportation & Storage of Spent Nuclear Fuel and High Level Radioactive Material & Waste" (no NRC position on this has been established).
- Section V, "Nondestructive Examination."
- Section VIII, "Rules for Construction of Pressure Vessels."
- Section IX, "Welding, Brazing, and Fusing Qualifications."
- Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components,"

American Society for Tests and Materials (ASTM) C1671-15, "Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorber Materials for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging," ASTM International, 2015.

ASTM E290-14, "Standard Test Methods for Bend Testing of Material for Ductility," 2014.

ASTM B557-06, Standard Test Methods for Tension Testing Wrought and Cast Aluminum- and Magnesium-Alloy Products," West Conshohocken, PA: ASTM International, 2006.

ASTM B351-13, "Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application," West Conshohocken, PA: ASTM International, 2013.

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