# 6 CRITICALITY EVALUATION

## 6.1 Review Objective

The objective of this U.S. Nuclear Regulatory Commission (NRC) criticality evaluation is to verify that the transportation package design meets the nuclear criticality safety requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, "Packaging and Transportation of Radioactive Material."

## 6.2 Areas of Review

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

- description of criticality design
  - packaging design features
  - codes and standards
  - summary table of criticality evaluations
  - criticality safety index (CSI)
- contents
- general considerations for criticality evaluations
  - model configuration
  - material properties
  - analysis methods and nuclear data
  - demonstration of maximum reactivity
  - confirmatory analyses
  - moderator exclusion under hypothetical accident conditions
- single package evaluation
  - configuration
  - results
- evaluations of package arrays
  - package arrays under normal conditions of transport
  - package arrays under hypothetical accident conditions
  - package arrays results and CSI
- benchmark evaluations
  - experiments and applicability
  - bias determination

- burnup credit evaluation for commercial light-water reactor (LWR) spent nuclear fuel (SNF)
    - — limits for the certification basis
    - — model assumptions
    - — code validation—isotopic depletion
    - — code validation—$k_{eff}$ determination
    - — loading curve and burnup verification
- 33appendix

## 6.3 Regulatory Requirements and Acceptance Criteria

This section summarizes those sections of 10 CFR Part 71 that are relevant to the criticality review areas addressed in this standard review plan (SRP) chapter. Table 6-1 identifies the regulatory requirements that are relevant to the areas of review covered in this chapter. The reviewer should refer to the exact language in the listed regulations. The reviewer should also refer to the regulations to ensure that no requirements are overlooked as a result of unique packaging design features or contents.

The packaging must be designed and the contents specified such that the package is subcritical under the design-basis conditions, normal conditions of transport, and hypothetical accident conditions (see 10 CFR 71.55(b), (d), and (e), respectively). The application should include evaluations of arrays of packages under normal conditions of transport and under hypothetical accident conditions to determine the maximum number of packages that may be transported in a single shipment, in accordance with 10 CFR 71.59, “Standards for Arrays of Fissile Material Packages.” The application should describe the packaging and the contents in sufficient detail to provide an adequate basis for their evaluation. The analyses in the application should show that the package (packaging and contents) design meets the following acceptance criteria:

- The sum of the effective neutron multiplication factor ($k_{eff}$), two standard deviations (95-percent confidence), and all biases and bias uncertainties should not exceed 0.95 to demonstrate subcriticality by calculation. A bias that reduces the calculated value of $k_{eff}$ should not be applied.

- The assumption of water inleakage for the analysis pursuant to 10 CFR 71.55(b) should consider the packaging and contents to be in their most reactive condition, consistent with the package design, including tolerances. All criticality analyses should include package tolerances.

- The regulatory criteria for uranium hexafluoride packages in 10 CFR 71.55(g) must be met. Note that this requirement allows exception of these packages from the 10 CFR 71.55(b) requirements, if certain conditions are met.

- Criticality evaluations for packages intended for air transport of fissile material or plutonium should also include 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, “Accident Conditions for Air Transport of Plutonium.” For packages intended to transport plutonium by air, this would include optimum internal moderation of the package.

**Table 6-1 Relationship of Regulations and Areas of Review for Transportation Packages**

| Areas of Review | 10 CFR Part 71 Regulations | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 71.31 | 71.33 | 71.35 | 71.41 | 71.43 | 71.51 | 71.55 | 71.59 | 71.61 | 71.63 | 71.64 | 71.71 | 71.73 | 71.74 | 71.83 | 71.87 |
| Description of criticality design | (a)(1), (c) | (a)(1)(5) | (b),(c) | | (f) | (a)(1) | (a),(b), (d),(e), (f),(g) | (a),(b) | | | (a)(1)(iii), (b)(2) | | | | ● | (f),(g) |
| Contents | (a)(1) | (b)(1)(2)(3)(4)(8) | | | | | (b),(d), (e),(f),(g) | | | ● | | | | | ● | (f) |
| General considerations for criticality evaluations | (a)(2), (b) | | (a) | (a),(d) | (d),(f) | (a)(1) | (b),(d), (e),(f),(g) | | ● | ● | (a)(1)(iii), (b)(2) | ● | ● | ● | ● | (f),(g) |
| Single package evaluation | (a)(2), (b) | | (a) | (a),(d) | (f) | (a)(1) | (b),(d), (e),(f), (g) | | ● | ● | (a)(1)(iii), (b)(2) | ● | ● | ● | | (f),(g) |
| Evaluations of package arrays | (a)(2), (b) | | (a),(b) | (a) | (f) | (a)(1) | (d),(e) | (a)(1)(2),(b) | ● | ● | (a)(1)(iii), (b)(2) | ● | ● | ● | | (f) |
| Benchmark evaluations | (a)(2), (b),(c) | | (a) | | | | (b),(d),(e) | (a) | | | | | | | | |
| Burnup credit evaluation for commercial LWR SNF | (a)(2), (b) | (b)(1)(2)(3)(4) | (a),(c) | | | | (b)(1), (d)(1)(3), (e)(1)(2) | (a) | | | | | | | ● | (f) |

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

- Criticality safety design may credit up to 90 percent of the neutron poison material in fixed boron-based neutron absorbers when subject to adequate acceptance and qualification testing (see Section 7.4.7 in this SRP). Otherwise, the packaging model for the criticality evaluation should consider no more than 75 percent of the specified minimum neutron poison concentrations for boron-based absorbers. The amount of credit for nonboron-based absorbers (e.g., cadmium) will be considered on a case-by-case basis and should be supported in the application with proper justification and acceptance and maintenance tests.

- For commercial SNF packages that include nonfuel hardware (NFH) as part of the contents, the applicant should identify and evaluate the most reactive configuration(s) of the contents. In general, the analyses may credit the presence of the NFH if the applicant can demonstrate that the NFH will remain in place under normal conditions of transport and hypothetical accident conditions. The package design description, including drawings, and the package Operating Procedures section of the application should also include descriptions of the components and operations that are necessary to ensure that the NFH remains in its loaded position, consistent with the criticality analyses.

- If credit is taken for residual neutron-absorbing material in NFH, the application should include evaluations demonstrating that the amount credited is appropriate. Credit for residual absorbing material in NFH should be limited to NFH such as pressurized-water reactor (PWR) control element assemblies and reactor-control assemblies, particularly those that are not used as regulating rods in reactor operations. In addition, neutron-absorber material may be credited in unirradiated poison rods or rodlets that are included in the package with the SNF contents.

- The criticality evaluation should include a comparison of the calculation method(s) with applicable benchmark experiments to determine the appropriate bias and bias uncertainties.

- For commercial LWR SNF packages that rely on burnup credit, the burnup credit analysis should follow the criteria and guidance discussed in Section 6.4.7 and Attachment 6A to this SRP chapter.

## 6.4 Review Procedures

Verify that the applicant has adequately described and evaluated the package's criticality design and demonstrated that the package meets the nuclear criticality safety requirements in 10 CFR Part 71. In addition to the guidance provided in this chapter, consult the information and guidance provided in the appropriate section of Appendix A, "Description, Safety Features, and Areas of Review for Different Types of Radioactive Material Transportation Packages," to this SRP, as applicable. Appendix A includes useful guidance that is specific to several package types.

As part of the evaluation, review and consider the package and contents descriptions presented in the General Information section of the application. Coordinate with the reviewers of the other sections of the application, as applicable and as described in the review procedures in this SRP chapter, to ensure that the applicant has adequately evaluated the packaging and the contents for both normal conditions of transport and hypothetical accident conditions and to ensure that the package will be fabricated, operated, and maintained consistent with the criticality evaluation and in a manner that the package meets the regulations. This includes ensuring that the acceptance tests include appropriate tests for those packaging components relied on for nuclear criticality safety (e.g., neutron absorbers, basket dimensions). It also includes ensuring that the package operations descriptions cover necessary operations elements and controls for loading, unloading, and transporting fissile material consistent with the criticality safety evaluation, in accordance with 10 CFR 71.35(c). Figure 6-1 illustrates the information flow and interdependency between the reviews of other sections of the application and the review of the criticality section.

As part of the review, ensure that the certificate of compliance (CoC) includes appropriate conditions for the package design, allowable package contents, package operations, and package acceptance and maintenance tests to ensure that the criticality safety performance of the package will be as designed and will meet regulatory requirements. To do this, see also the guidance in Chapter 1, “General Information Evaluation,” Chapter 8, “Operating Procedures Evaluation,” and Chapter 9, “Acceptance Tests and Maintenance Program Evaluation” of this SRP and work with the reviewers of those chapters.

### 6.4.1 Description of Criticality Design

#### *6.4.1.1 Packaging design features*

Review the General Information section of the application and any additional description of the criticality design presented in the Criticality Evaluation section of the application. Packaging design features important for criticality safety include, but are not limited to, the following:

- dimensions and tolerances of the containment system for fissile material
- structural components that maintain the fissile material or neutron-absorbing and moderating materials in a fixed position within the package or in a fixed position relative to each other, including the dimensions, material compositions, and tolerances for these structural components
- location, dimensions, concentration, and tolerances (both dimensional and composition) of neutron-absorbing materials and moderating materials, including neutron poisons and shielding material
- dimensions and tolerances of any floodable voids, including flux traps, within the package
- dimensions and tolerances of the overall package that affect the physical separation of the fissile material contents in package arrays

Confirm that the text, tables, figures, and sketches describing the criticality design features are consistent with each other; with the information in the General Evaluation section of the application, including the engineering drawings; and with the models used in the criticality evaluation. The drawings are the authoritative source of dimensions, tolerances, and material compositions of components important to criticality safety. The drawings will also become a part of the CoC by reference. Therefore, ensure that the drawings clearly identify and describe, with sufficient specificity, the components and features that provide or affect the packaging’s nuclear criticality safety function (e.g., minimum areal density of boron-10 in neutron absorbers) under design-basis conditions and under normal conditions of transport and the appropriate

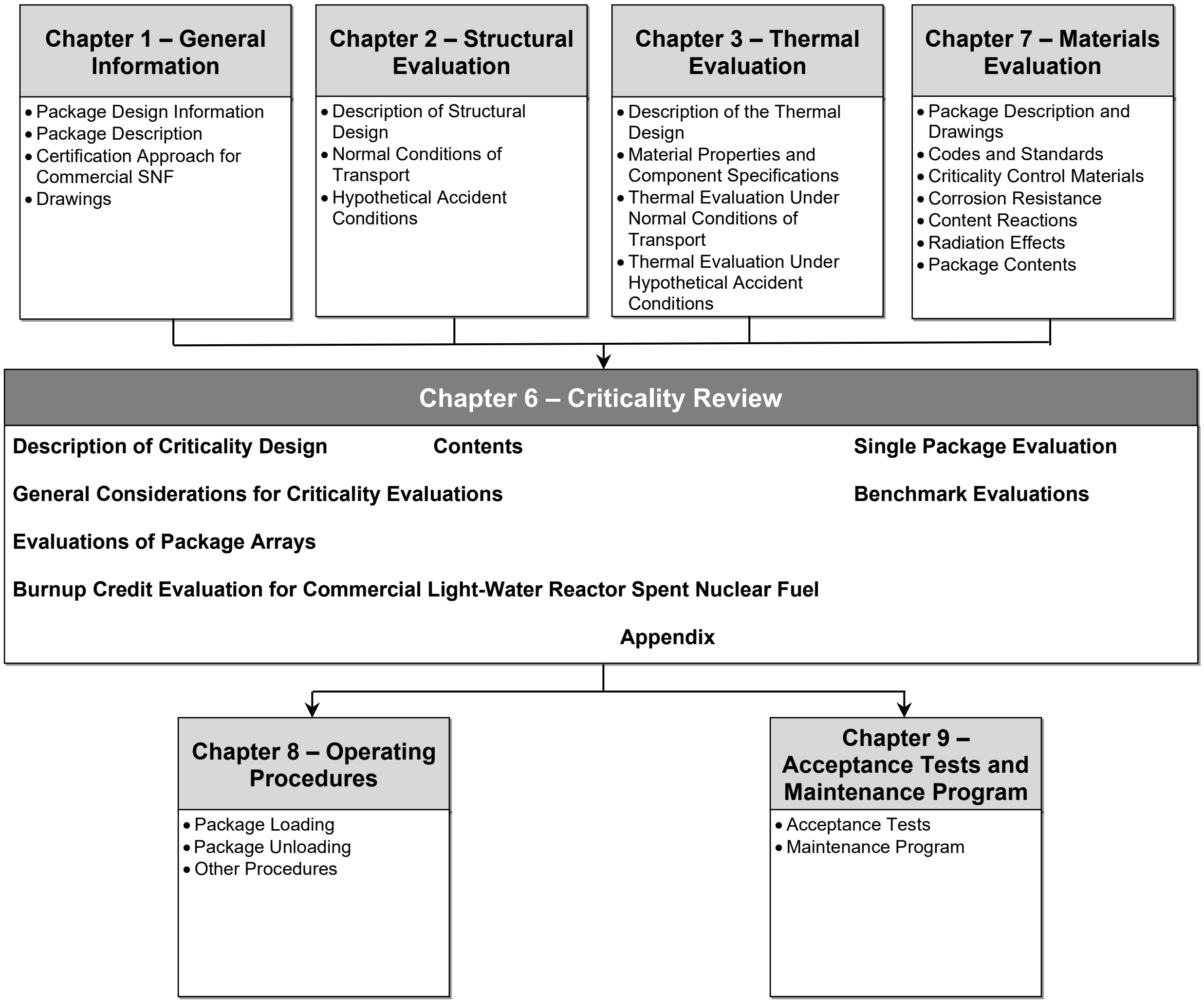


**Figure 6-1 Information Flow for the Criticality Evaluation**

accident conditions (i.e., as applicable, the tests in 10 CFR 71.55(f), 10 CFR 71.73, "Hypothetical Accident Conditions," and 10 CFR 71.74). The degree of specificity should be commensurate with the sensitivity of the package's performance with the particular feature.

Ensure that the specifications in the drawings are consistent with or bounded by the specifications used in the criticality analyses, including reasonable tolerances for dimensions and material specifications. In reviewing the drawings, refer to NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," issued May 1998, and NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," issued February 1996. These documents contain information that may be useful in determining whether the drawings provide sufficient details. Also, coordinate with the structural reviewer to understand the performance of these packaging design features under normal conditions of transport and hypothetical accident conditions.

*6.4.1.2 Codes and standards*

Verify that the applicant identified the established codes and standards used in all aspects of the criticality design and evaluation, if any, and that the applicant used them appropriately.

Coordinate this review, as appropriate, with the other reviewers. For example, review of codes and standards regarding neutron absorber materials should be coordinated with the materials reviewer (see Section 7.4.7 and Attachment 7A to this SRP). Also, consider the staff's position on the use of standards as described in documents such as Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," in determining the acceptability of the applicant's use of any standards in the design or evaluation of the package.

*6.4.1.3 Summary table of criticality evaluations*

Review the summary table of the criticality evaluation, which should address the following cases, as described in Sections 6.4.4 to 6.4.6 in this SRP chapter:

- a single package under the conditions of 10 CFR 71.55(b), (d), and (e)
- an array of 5N undamaged packages under the conditions of 10 CFR 71.59(a)(1)
- an array of 2N damaged packages under the conditions of 10 CFR 71.59(a)(2)

For a fissile material package designed for air transport, the table should also address a single package under the conditions of 10 CFR 71.55(f). For a package for air transport of plutonium, the table should address both a single package and an array of packages under the conditions of 10 CFR 71.64(a)(1)(iii) and (b). This means that the analyses for 10 CFR 71.55(e) and 10 CFR 71.59(a)(2) must use the damaged condition of the package resulting from the 10 CFR 71.74 accident conditions tests instead of the 10 CFR 71.73 hypothetical accident conditions tests, accounting for the additional considerations in 10 CFR 71.64(b). The other conditions of 10 CFR 71.55(e) and 10 CFR 71.59(a)(2), including optimum internal moderation, would still apply to these analyses.

Verify that the table includes results for all relevant cases. Also verify that for each case the table includes the maximum value of $k_{eff}$, the uncertainty, the bias and bias uncertainty, and for the array cases, the number of packages evaluated in the arrays. The table should also show that the sum of $k_{eff}$, two standard deviations (95-percent confidence), and the bias adjustment does not exceed 0.95 for each case. For packages that have multiple fissile material content types or multiple content loading configurations (e.g., canisters containing SNF from a specific reactor versus canisters containing general classifications of SNF assemblies) and for which separate evaluations are performed for each content type, verify that the table includes the results for all relevant cases for each content type.

Confirm that the summary table illustrates that the package meets the above subcriticality criterion for all of the package's types of fissile contents.

*6.4.1.4 Criticality safety index*

The CSI designates the degree of control of accumulation of fissile material packages during transportation (see 10 CFR 71.4, "Definitions"). The CSI is limited to ensure that the number of packages in a shipment does not exceed the number that was evaluated. The CSI is included on the package label for a fissile package shipment. The regulation in 10 CFR 71.59(c) describes the CSI limits for shipments in nonexclusive-use and exclusive-use conveyances. The limits include those for both the individual package CSI and the total CSI for all of the packages shipped in the conveyance, both of which must be met for the type of conveyance to be used.

Based on the number of packages evaluated in the arrays, verify that the applicant has determined the appropriate value of N and calculated the CSI correctly. The appropriate value of N will be the smaller of values determined from the arrays evaluated according to 10 CFR 71.59(a)(1) and (a)(2). For packages with multiple types of fissile contents or multiple content configurations, the applicant may determine a separate CSI for each type of contents or content configuration. In addition, for some packages or some fissile content types in a package, the applicant may determine the CSI, in accordance with 10 CFR 71.22, “General License: Fissile Material,” or 10 CFR 71.23, “General License: Plutonium-Beryllium Special Form Material.” Ensure that the CSI for the package, or for each package content type or content configuration, is consistent with that reported in the General Information section of the application.

### 6.4.2 Contents

Ensure that the application clearly and adequately describes the package contents, providing those specifications that are relevant to the criticality safety of the package. The application should show the entire range of contents specifications, or characteristics, that the applicant considered and should specify the limiting values (maximum or minimum, as appropriate) for the contents specifications. Nominal values may be used if the safety of the package is insensitive to small changes in the specified parameter (e.g., active fuel length). Ensure that the specifications for the contents used in the criticality evaluation are consistent with or bound those in the General Information section of the application. The application should include a description of the contents in an appropriate and easy-to-understand format (e.g., a table of fuel assembly parameters) that is suitable for inclusion in a CoC. There should be a clear nexus between the contents description and the criticality safety analysis. The specificity of the contents description may be different for different package types or may depend on how the applicant performed the analyses. For contents properties that are not known or are not well known, ensure that the applicant has assumed these properties have credible values that maximize reactivity in the criticality analyses, consistent with 10 CFR 71.83, “Assumptions as to Unknown Properties.”

Also, for some package types, the applicant may propose that the material may be exempted from classification as fissile material per 10 CFR 71.15 and therefore exempt from the fissile material package standards in 10 CFR 71.55, “General Requirements for Fissile Material Packages,” and 10 CFR 71.59, “Standards for Arrays of Fissile Material Packages.” In such cases, ensure that the other content descriptions in the application are consistent with the limits in 10 CFR 71.15 for this exemption and that the CoC includes this limitation on the package contents.

An application may include only some contents specifications in the General Information section and place the rest in the different evaluation sections (e.g., the Criticality Evaluation and Shielding Evaluation sections). For this reason, coordinate with the reviewers of those sections too, as needed, to confirm the consistency of contents specifications within the application. Verify that the application clearly identifies and justifies any differences from the specifications in the General Information section and the other relevant application sections. Coordinate with the other reviewers to ensure that a CoC for package approval includes the contents specifications necessary to ensure that the package meets the 10 CFR Part 71 criticality safety requirements. In general, if the applicant takes credit for certain parameters (e.g., confinement features, uranium enrichment, chemical form) or the analyses indicate that certain parameters affect the criticality safety of the package, then the description of the authorized contents should specify those parameters.

For fissile material contents, verify that the application provides significant detail consistent with the criticality analysis of the package. Specifications relevant to the criticality evaluation include fissile material mass, dimensions, uranium enrichment(s), fissile nuclides present and their concentrations, physical and chemical composition and form, density, internal moderation (e.g., moisture, plastic inserts, or wrap for assemblies), and other characteristics, depending on the specific contents. These other characteristics may include the contents' configuration(s) in the package and the inclusion of any materials that act as neutron moderators or neutron poisons and the material, dimension, and configuration specifications of these materials. They may also include spacers or other features used for geometry control, though these features may be considered as part of the packaging design and included in the engineering drawings instead. Because a partially filled container may allow more room for moderators (e.g., water), the most reactive case may be for a mass of fissile material that is less than the maximum allowable contents.

In addition to the characteristics described above, the relevant contents specifications for fuel assembly or fuel element contents include many characteristics that apply to the criticality analysis, such as the following:

- types of assemblies or elements [e.g., PWR, boiling-water reactor (BWR), research reactor (e.g., flat or curved plate fuel, pin fuel)]

- whether the contents are complete assemblies or elements or the contents are loose rods or fuel plates

- dimensions of fuel material (e.g., pellet diameter, including any annular pellets, for rods or thickness and width for fuel plates), cladding material and dimensions, fuel-cladding gap, pitch, and rod or plate length

- inclusion of items to prevent assembly damage during transport (e.g., polymer inserts to prevent wear due to vibration); wrapping of fresh fuel assemblies with plastic is permitted if the top and bottom are open to allow free flow of water sufficient to prevent preferential flooding of the fuel region

- configurations of poison-bearing rods (e.g., fuel rods containing gadolinium oxide) in unirradiated BWR fuel assemblies

- number of rods (and lattice configuration, such as 15x15) or fuel plates per assembly and locations of guide tubes, water rods, and burnable poisons (see Section 6.4.3.2), including numbers and locations of partial-length rods

- inclusion of fuel assembly components, hardware such as BWR fuel channels, or unirradiated neutron-absorber rods

- active fuel length

- mass of heavy metal per assembly or element or per rod or fuel plate

- number of fuel assemblies or elements or the number of individual rods or fuel plates per package

With regard to enrichment, assemblies may have fuel enrichments that vary by rod or by axial lattice location. Ensure that the application clearly describes how the enrichment is defined for the contents and demonstrates that the definition is appropriate for use in ensuring that the fuel assembly contents in the package will be subcritical. The applicant's evaluation should either assume the maximum initial enrichment or demonstrate that another approach (e.g., average enrichment) is bounding.

For irradiated, or spent, fuel, ensure that the application specifies parameters such as enrichment and mass of heavy metal per assembly (or element) as initial (i.e., preirradiation) values. Also, ensure that the application includes the descriptions and specifications of any NFH to be included with the SNF contents. This hardware includes items such as control-rod assemblies, burnable poison rod assemblies, fuel channels, and other items that are operated and irradiated within the fuel assembly envelope in the reactor.

For applications that take credit for residual absorber in commercial reactor control components to be loaded with SNF, ensure that the application includes appropriate specifications, such as maximum burnup (or irradiation exposure) and operational history in the core (e.g., operated in the "bite" position in the core or as a regulating rod) to characterize the amount of absorber material remaining in the nonfuel hardware. Verify that the application includes analyses demonstrating that the amount of residual absorber being credited will be present in the control components and that the analyses are conservative for or consistent with the component's use in the core (e.g., in the "bite" position or as a regulating rod). The analysis should include a depletion analysis of the initial absorber loading for a bounding maximum burnup and should not take any credit for nuclides that may build up in the control component as a result of irradiation. In other words, the criticality analysis should take credit only for residual amounts of the initial absorber material that remains after depletion. Ensure that the depletion analysis uses conservative assumptions (e.g., for neutron-flux factors). Given uncertainties in these analyses that result from things such as lack of data to "benchmark" the depletion of these components and uncertainties in the irradiation history, the applicant should credit only a fraction of the residual absorber material in the criticality evaluation. The applicant should justify that the fraction of credit used in the analysis is appropriate to account for the uncertainties in the depletion analysis for the control component.

In addition, for commercial reactor SNF contents for which the applicant requested burnup credit, ensure that the application specifies appropriate characteristics for assemblies for which burnup is credited. These characteristics include minimum burnups versus maximum enrichment and reactor operating parameters during assembly irradiation (e.g., exposure limits to control-rod insertion, in-core soluble boron concentrations, moderator temperature, and assembly specific power). Section 6.4.7 of this SRP provides guidance regarding burnup credit.

Determine whether the application for an SNF package includes any specifications regarding the condition of the SNF. If the contents include damaged fuel, confirm that the application specifies the maximum extent of damage allowed and that the applicant's criticality analyses show the package containing damaged fuel is subcritical. Fuel rods that have been removed from an assembly should be replaced with dummy rods that displace an equal or greater amount of water unless the criticality analyses consider the additional moderation resulting from their absence. (Because of the additional moderation, the contents with less fissile material might be more reactive). Ensure that the CoC includes specifications regarding the condition of the SNF in the conditions describing the approved contents. Coordinate this review with the materials evaluation reviewer as necessary (see Section 7.4.14 of this SRP).

NUREG/CR-6716, “Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks,” issued March 2001, also includes useful information about the fuel parameters that are important for criticality safety for a commercial SNF transport package. Parameters that are normally controlled in CoC conditions include fuel type, lattice size, enrichment, fuel rod pitch, fuel pellet diameter, cladding thickness, and active fuel length. It is not necessary to limit all parameters if the analysis has shown that they are not important for the package evaluation. For example, if the applicant evaluates the criticality safety of the fuel without taking credit for the clad material being present, the minimum clad thickness may not need to be specified.

If the package is designed for multiple types of contents, including multiple types of SNF or multiple content configurations, verify that the description of the contents is sufficient to permit a detailed criticality evaluation of each type or configuration or to support a conclusion that certain types or configurations are bounded by those that the applicant did evaluate. The application may include a separate criticality evaluation and propose different criticality controls (e.g., fissile mass limits, uranium enrichment limits, CSI) for each content type or configuration. Or the application may include an evaluation that bounds all content types and configurations and propose criticality controls that apply to all content types and configurations. The review procedures in this section and the rest of this chapter apply to each content type, including each type of SNF, and configuration evaluated in the application.

### 6.4.3 General Considerations for Criticality Evaluations

The considerations discussed below apply to the criticality evaluations of a single package, arrays of packages under normal conditions of transport, and arrays under hypothetical accident conditions. NUREG/CR-5661, “Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages,” issued April 1997, provides general guidance for preparing criticality evaluations of transportation packages.

#### *6.4.3.1 Model configuration*

Verify that the applicant’s analysis includes a model for demonstrating compliance with 10 CFR 71.55(b) and that the model is consistent with the as-designed package, including tolerances and materials specifications of package components that maximize reactivity. Coordinate with the structural evaluation, thermal evaluation, and materials evaluation reviewers to determine the effects of the normal conditions of transport and hypothetical accident conditions on the packaging and its contents. Verify that the models used in the criticality calculation are consistent with these effects.

Verify the dimensions of the contents and packaging used in the criticality models. Ensure that they are consistent with the package drawings and contents specifications in the application. Confirm that the applicant has identified and justified any differences between the models and the drawings and contents specifications. For some types of packagings and contents (e.g., powders), the contents can be positioned at varying locations and densities. Verify that the application justifies the relative location and physical properties of the contents within the packaging as those resulting in the maximum multiplication factor. Verify that the application considers dimensional tolerances for parameters such as cavity sizes and poison thickness in a way that maximizes reactivity.

Verify that the application considers deviations from nominal design configurations. For example, fuel assemblies might not always be centered in each basket compartment, and the basket might not be exactly centered in an SNF package. In addition to a fully flooded package, confirm that the application addresses preferential flooding, as appropriate. For fuel assemblies, this includes flooding of the fuel-cladding gap and other regions (e.g., flux traps) for which water density might not be uniform in a flooded package. Also ensure that the application considers partially loaded packages since, in some cases, packages loaded to less than the maximum capacity may be more reactive.

For packages designed to transport fuel assemblies (fresh or spent), determine whether the application includes a heterogeneous model of each fuel rod or homogenizes the entire assembly. With current computational capability, homogenization should generally be avoided. If homogenization is used, the application must demonstrate that it is applied correctly or conservatively. At a minimum, this demonstration should include calculation of the multiplication factor of one assembly and several benchmark experiments (see Section 6.4.6) using both homogeneous and heterogeneous models.

Also, for SNF packages that include damaged fuel contents, determine whether the applicant has adequately evaluated a package containing damaged fuel, including identification of a bounding reconfiguration of the contents. For those evaluations that rely on damaged fuel cans or other features to confine the geometry of the damaged SNF, ensure that the applicant's analyses are consistent with the design specifications of these features. Also ensure that the package drawings, which will become part of the CoC, include these features with the specifications that are important to their function of confining the damaged SNF within a set geometric configuration.

#### *6.4.3.2 Material properties*

Verify the materials that are used in the criticality models for the packaging and contents. Verify that the applicant provided appropriate mass densities and atom densities for materials used in the models of the packaging and contents. Material properties should be at the specifications or tolerances that maximize reactivity and that are consistent with the condition of the package under the tests of 10 CFR 71.71, "Normal Conditions of Transport," and 10 CFR 71.73. For fissile material packages designed or intended for air transport, the material properties should also be consistent with the condition of the package under the tests described in 10 CFR 71.55(f). For plutonium packages designed or intended for air transport, the material properties should be consistent with the condition of the package under the tests described in 10 CFR 71.74 instead of the tests described in 10 CFR 71.73. Verify that the application addresses any differences between normal conditions of transport and the appropriate accident conditions, as identified above. Confirm that the application includes references for the data sources of the material properties.

Ensure that all materials relevant to the criticality design (e.g., poisons, foams, plastics, and other hydrocarbons) are properly specified. Confirm that the values used for neutron poisons match the minimum required values credited in the criticality analysis. Also confirm that, for neutron absorbers that are part of the packaging, the analysis does not credit more than the minimum amount of neutron absorber the acceptance testing and qualification testing verified , subject to the criteria described in Section 6.3 and Section 7.4.7 of this SRP. Ensure that neutron absorbers and moderators (e.g., poisons and neutron shielding) are properly controlled during fabrication to meet their specified properties. The Acceptance Tests and Maintenance Program section of the application should discuss such information in more detail. For packages that include other kinds of absorbers, such as unirradiated poison rods or rodlets loaded with fuel contents or nonboron-based absorbers (e.g., cadmium), confirm that the applicant's analysis credits only an amount of absorber material that is consistent with or bounding for the absorbers, accounting for material and dimensional tolerances, other relevant fabrication variabilities, and neutronics properties. For packages that credit these kinds of absorbers, ensure that the application describes how these absorbers will be maintained in the positions for which they are credited in the analysis. Working with the materials reviewer, ensure that the application includes adequate acceptance tests for these absorbers too, as applicable and appropriate.

In addition, for commercial SNF packages, because of differences in net reactivity resulting from the depletion of fissile material and burnable poisons, in general, no credit should be taken for burnable poisons in the fuel. Also, in general, the application should not credit any negative reactivity from residual neutron-absorbing material remaining in commercial reactor control components also loaded with the commercial SNF as nonfuel hardware. However, this credit may be taken and should be accepted only if (i) the remaining absorbing material content is established through direct measurement or by calculation where a sufficient margin of safety is included, commensurate with the uncertainty in the method of measurement or calculation; (ii) the axial distribution of the poison depletion is adequately determined with appropriate margin for uncertainties; and (iii) the adequate structural integrity and placement of the control components under accident conditions are demonstrated. For evaluations with water in the package, which is always fresh water for package analyses, a bounding analysis would assume that no nonfuel hardware, including control components, are present. The applicant may take credit for water displacement, provided that adequate structural integrity and placement under accident conditions are demonstrated.

Review materials to identify any materials that are relevant to the criticality design that have properties that could degrade during the service life of the packaging. If appropriate, ensure that specific controls are in place to ensure the effectiveness of the packaging during its service life. The Acceptance Tests and Maintenance Program or Operating Procedures sections of the application should discuss such information in more detail.

Coordinate the reviews of the material properties described here with the materials reviewer. For the materials properties of SNF packages that rely on burnup credit, see the burnup credit guidance in Section 6.4.7 of this SRP chapter.

#### *6.4.3.3 Analysis methods and nuclear data*

Verify that the applicant used an appropriate method and appropriate data for the package analyses the regulations required and that are discussed in this SRP chapter. The vast majority of package criticality analysis methods use computer codes and the nuclear data included with those codes. However, depending on the applicant's approach, the applicant may use other methods that may also be appropriate to demonstrate subcriticality. Even for analyses that use computer codes, although the algorithm and calculation process that a computer code uses is a method (e.g., Monte Carlo versus deterministic technique) and should be evaluated that way, the analysis method is more than just the computer code. In other words, the computer code is a part of the analysis method. The analysis method includes the nuclear data, such as the cross-section libraries, used in the analysis and the selection of the data. The method also includes things such as key assumptions and parameters and the approach to modeling the contents and the packaging components. For noncode-based analyses as well, the method includes things such as the nuclear data used in the analysis, key assumptions and parameters, and the approach to analyzing the package contents and packaging components.

Verify that the application uses an appropriate computer code (or other acceptable method) for the criticality evaluation and that the applicant has used the code (or other method) properly. Both Monte Carlo and deterministic computer codes may be used for criticality calculations. Because Monte Carlo codes are generally better suited to analyzing three-dimensional geometry, they are more widely used to evaluate SNF cask designs. The application should clearly reference standard codes, such as SCALE/KENO (ORNL 2011) and MCNP (MCNP5 2003), used in the analysis. KENO is part of the SCALE code system and allows the use of both multigroup and continuous-energy cross sections, while MCNP uses continuous-energy cross sections. If the analysis uses other codes or methods, the application should describe these other codes or methods and provide appropriate supplemental information.

Ensure that the criticality evaluations use an appropriate cross-section library. If multigroup cross sections are used, confirm that the neutron spectrum of the package has been appropriately considered for collapsing the group structure and that the cross sections are properly processed to account for resonance absorption and self-shielding. The use of KENO as part of the SCALE sequence will directly enable such processing. Some cross-section sets include data for fissile and fertile nuclides (based on a potential scattering cross section, $\sigma_p$) that the user can input. If the applicant has used a stand-alone version of KENO, ensure that potential scattering has been properly considered. NRC Information Notice (IN) 91-26, "Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set Provided with the KENO and SCALE Codes," dated April 2, 1991, and NUREG/CR-6328, "Adequacy of the 123-Group Cross-Section Library for Criticality Analyses of Water-Moderated Uranium Systems," issued June 1995, provide additional information addressing cross-section concerns.

In addition to cross-section information, verify that the application identifies other key input data for the criticality calculations. These data include the number of neutrons per generation, number of generations, convergence criteria, and mesh selection, depending on the code used. The application should also include at least one representative input file for a single package, undamaged array, and damaged array evaluation. Verify, as appropriate, that information for the model configuration, material properties, and cross sections is properly input into the code.

Generally, the application should also include at least one representative output file (or key sections). Ensure that the calculation has properly converged and that the calculated multiplication factors from the output files agree with those reported in the evaluation.

#### *6.4.3.4 Demonstration of maximum reactivity*

Verify that the application evaluates each type of allowable contents or clearly demonstrates that some types are bounded by the contents for which the applicant performed evaluations. For packages for fuel assemblies, whether an unirradiated fuel package or an SNF package, this includes verifying that the application evaluates each type of fuel assembly or shows that the evaluated types bound the remaining types.

Verify that, for each contents type, the analyses demonstrate the maximum $k_{eff}$ for each of the cases discussed in Section 6.4.1.3 above (single package, array of undamaged packages, and array of damaged packages for the relevant conditions). Verify that the application clearly identifies and justifies assumptions and approximations.

Ensure that the analysis determines the optimum combination of internal moderation (within the package) and interspersed moderation (between packages), as appropriate. Confirm that preferential flooding of different regions within the package is considered, as appropriate. As noted in Section 6.4.2 of this SRP chapter, the maximum allowable amount of fissile material may not be the most reactive.

NUREG/CR-5661 presents additional guidance on determining the most reactive configurations.

Confirm that the applicant's evaluation demonstrates that the package calculations are adequately converged and addresses the statistical uncertainties of the package calculations. Verify that the applicant applied the uncertainties to at least the 95-percent confidence level. As a general rule, if the acceptability of the criticality evaluation results depends on these rather small differences, question the overall degree of conservatism of the calculations. Considering the current availability of computer resources, enough neutron histories can readily be used so that the treatment of these statistical uncertainties should not significantly affect the results.

*6.4.3.5 Confirmatory analyses*

Perform a confirmatory analysis of the criticality calculations reported in the application, as appropriate. At a minimum, perform an independent calculation of the most reactive case, as well as sensitivity analyses to confirm that the most reactive case has been correctly identified. In deciding the necessary level of effort to perform independent confirmatory calculations, consider the following factors: (i) the calculational method (computer code) the applicant used, (ii) the degree of conservatism in the applicant's assumptions and analyses, (iii) the size of the margin between the calculated result and the acceptance criterion of $k_{eff} \leq 0.95$, and (iv) the degree of similarity to previously approved packages or package contents. A small margin below the acceptance criterion or a small degree of conservatism in the applicant's analyses may likely necessitate a more extensive analysis. This would be particularly true if aspects of the applicant's analysis seem to be questionable and may be significant to the analysis and to the criticality safety of the package (e.g., things the applicant did not include or items that were treated in a possibly nonconservative manner).

To the extent practical, model the package independently and use a different code and cross-section set from those used in the application. If the reported $k_{eff}$ for the worst case is substantially lower than the acceptance criterion of 0.95, a simple model known to produce very conservative results may be all that is necessary for the independent calculations. A review is not expected to validate the applicant's calculations but should confirm that the regulations and acceptance criteria are met.

When the value of $k_{eff}$ is highly sensitive to small variations in design features, contents specifications, or the effects of the relevant test conditions (i.e., 10 CFR 71.71, 10 CFR 71.73, 10 CFR 71.55(f), and 10 CFR 71.74, as applicable), confirm that the applicant appropriately considered such variations.

*6.4.3.6 Moderator exclusion under hypothetical accident conditions*

For commercial LWR SNF, refer to Section 1.4.4 of this SRP, which describes approach options for addressing subcriticality of SNF that is categorized as intact or undamaged fuel[4] under hypothetical accident conditions. Thus, the review guidance in this section applies only to intact or undamaged commercial SNF for hypothetical accident conditions. This section does not

[4] Note that the International Atomic Energy Agency's Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Material," includes similar, but not identical, requirements for fissile material packages.

apply to evaluations for compliance with 10 CFR 71.55(b) and so does not change guidance related to meeting that requirement that is described in the other sections of this SRP chapter.

As described in Section 1.4.4 of this SRP, the applicant may choose to demonstrate package subcriticality under hypothetical accident conditions by showing that (i) reconfigured fuel is subcritical even with water inleakage, or (ii) the package excludes water under hypothetical accident conditions. Verify that the application describes the evaluation approach. Also determine that the applicant has adequately justified use of the selected approach and has adequately demonstrated that the package is subcritical. For this review, consult the guidance in Section 1.4.4 of this SRP and coordinate with the other reviewers (e.g., materials evaluation, structural evaluation) to ensure that the applicant adequately evaluated the package for the selected approach and that the applicant's criticality analysis is consistent with or bounding for the evaluated condition of the package and commercial SNF contents for the applicant's selected approach. Also coordinate with the other reviewers to ensure that the package operating procedures, acceptance tests, and maintenance programs in the application include the appropriate procedures and tests to ensure that the package is operated, fabricated, and maintained consistent with the evaluations in the application.

For the first approach, the fuel reconfiguration geometries should either be based on the material properties of the SNF cladding and the impact loads imposed on the fuel assemblies or be those that are appropriately bounding for criticality. NUREG/CR-6835, "Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks," issued September 2003, and NUREG/CR-7203, "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages," issued September 2015, provide information on the reactivity effects of various postulated fuel reconfiguration scenarios that may be useful for this review. For the second approach, the criticality assessment would use credible or bounding reconfigured fuel configurations and assume moderator exclusion. For analyses to demonstrate compliance with 10 CFR 71.55(b), SNF that is intact or undamaged when loaded into the package can be assumed to be in its as-loaded configuration.

### 6.4.4 Single Package Evaluation

#### *6.4.4.1 Configuration*

Ensure that the criticality evaluation demonstrates that a single package is subcritical in the as-designed condition for compliance with 10 CFR 71.55(b) and under both normal conditions of transport and hypothetical accident conditions for compliance with 10 CFR 71.55(d) and (e), respectively. For packages for air transport of fissile material, ensure that the evaluation also demonstrates that a single package is subcritical under the accident conditions in 10 CFR 71.55(f). For packages for air transport of plutonium, ensure that the evaluation for compliance with 10 CFR 71.55(e) uses the damaged condition of the package resulting from the accident tests in 10 CFR 71.74, consistent with the considerations required in 10 CFR 71.64(a)(1)(iii) and (b). Verify that the evaluation considered the following:

- fissile material in its most reactive credible configuration, consistent with the condition of the package and the chemical and physical form of the contents

- water moderation to the most reactive credible extent, including water leakage into the containment system as specified in 10 CFR 71.55(b)

- full water reflection on all sides of the package, including close reflection of the containment system or reflection by the package materials, whichever is more reactive, as specified in 10 CFR 71.55(b)(3)

#### *6.4.4.2 Results*

Confirm that the results of the criticality calculations are consistent with the information presented in the summary table discussed in Section 6.4.1.3. If the package can be shown to be subcritical by reference to a standard such as American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors” (in lieu of calculations), verify that the standard is applicable to, or bounding for, the package conditions and contents.

Verify also that the package meets the additional specifications of 10 CFR 71.55(d)(2) through 10 CFR 71.55(d)(4) under normal conditions of transport. These requirements address subcriticality, alteration of the geometric form of the contents, inleakage of water, and effectiveness of the packaging.

### 6.4.5 Evaluations of Package Arrays

#### *6.4.5.1 Package arrays under normal conditions of transport*

Ensure that the criticality evaluation demonstrates that an array of 5N packages is subcritical under normal conditions of transport. Verify that the evaluation considered the following:

- the most reactive configuration of the array (e.g., pitch, package orientation), with the most reactive interstitial moderation between the packages

- the most reactive, credible configuration of the packaging and its contents under normal conditions of transport. If the water spray test has demonstrated that water would not leak into the package, water inleakage need not be assumed (as is typically the case for packages such as SNF packages)

- full water reflection on all sides of the array (unless the array is infinite)

Verify that the application clearly identifies the most reactive array conditions and that the results of the analysis are consistent with the information presented in the summary table discussed in Section 6.4.1.3 above.

#### *6.4.5.2 Evaluation of package arrays under hypothetical accident conditions*

Ensure that the criticality evaluation demonstrates that an array of 2N packages is subcritical under hypothetical accident conditions (or the accident conditions resulting from the tests in 10 CFR 71.74 for packages for air transport of plutonium). Verify that the evaluation considered the following:

- the most reactive configuration of the array (e.g., pitch, package orientation, internal moderation)

- optimum interspersed hydrogenous moderation (between packages)

- the most reactive, credible configuration of the packaging and its contents under accident conditions (the appropriate accident conditions from 10 CFR 71.73 or 10 CFR 71.74), including inleakage of water and internal moderation (including optimum moderation and, if applicable, partial flooding)

- full water reflection on all sides of the array (unless the array is infinite)

Verify that the application clearly identifies the most reactive array conditions and that the results of the analysis are consistent with the information presented in the summary table discussed in Section 6.4.1.3 above.

*6.4.5.3 Package arrays results and criticality safety index*

Confirm that the appropriate N value is used to determine the CSI in accordance with 10 CFR 71.59(a) and (b). The appropriate N should be the smallest value that ensures subcriticality for 2N packages under the appropriate accident conditions, whether 10 CFR 71.73 (which will apply to most packages) or 10 CFR 71.74 (for packages for air transport of plutonium), or 5N packages under normal conditions of transport, as discussed in the previous subsections.

Verify that the application includes results, including the CSI determination, for each package content type, if the applicant performed evaluations for or proposes different CSI values for each type of contents. If the applicant proposes a single CSI value, provides results for only a single type of contents, and represents that type of contents as bounding of the others, confirm that the results and proposed value are indeed bounding for all package content types. When developing the CoC, ensure that the certificate conditions specify the appropriate CSI value(s) for the correct content type(s).

### 6.4.6 Benchmark Evaluations

Ensure that the applicant has benchmarked the computer codes for criticality calculations against appropriate critical experiments. Verify that the applicant used the same computer code, hardware, and cross-section library to analyze the benchmark experiments as those used to calculate the multiplication factor for the package evaluations. In the application, the $k_{eff}$ results should include the calculated package $k_{eff}$(s), bias(es) and uncertainty(ies) (i.e., bias uncertainties) from the benchmark calculations, and the $k_{eff}$(s) as adjusted to include the bias(es) and bias uncertainty(ies). Ensure that the applicant's benchmark evaluation is a comparison of the calculated results to the experimental results and not a code-to-code comparison. The staff does not accept code-to-code comparisons as benchmark evaluations. This staff position is consistent with guidance in industry standards regarding benchmarking and validation (e.g., see ANSI/ANS 8.1-1998 (R2007), Section 4.3.1, "Establishment of Bias," including the footnotes).

NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," issued March 1997, and NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," issued January 2001, provide additional information on benchmarking criticality evaluations.

For mixed oxide (MOX) SNF evaluations, the differences between the package and benchmark experiments may be more substantial because there are fewer experiments for MOX than for low-enriched uranium. Thus, it may be more difficult to properly consider these differences and assign a bias value. Refer to Appendix D to this SRP for information regarding available MOX benchmark experiments and their important characteristics and for guidance on selecting appropriate benchmark experiments and determining a conservative bias from the benchmark analysis.

#### *6.4.6.1 Experiments and applicability*

Review the general description of the benchmark experiments, and confirm that they are appropriately referenced.

Verify that the applicant has selected benchmark experiments that apply to the actual packaging design and contents. Verify that the applicant has adequately justified either the selection of any experiments that do not readily appear to be applicable or the neglect of any experiments that would seem to be appropriate for use in benchmarking the package evaluation. The benchmark experiments should have, to the maximum extent possible, the same materials, neutron spectrum, and configuration(s) as the package evaluations for each type of contents. Key package parameters that should be compared with those of the benchmark experiments include type of fissile material, enrichment, H/X ratio (where H is hydrogen (moderator) and X is the fissile material; dependent largely on rod pitch and diameter for commercial SNF cases), poisoning, reflector material, and configuration. Confirm that the application discusses and properly considers differences between the package and benchmarks.

The Nuclear Energy Agency's "International Handbook of Evaluated Criticality Safety Benchmark Experiments," updated annually, provides information on benchmark experiments that may apply to the cask being analyzed.

In addition, verify that the application addresses the overall quality of the benchmark experiments and the uncertainties in experimental data (e.g., mass, density, dimensions, reported $k_{eff}$ results). Ensure that these uncertainties are treated conservatively (i.e., they result in a lower calculated multiplication factor for the benchmark experiment).

In recent years, some analytical tools have been developed that may be useful for identifying applicable benchmark experiments and evaluating the quality of the experiments. These tools include SCALE's TSUNAMI tools, which use sensitivity and uncertainty techniques to provide a quantitative measure of the overall similarity of an experiment to the analyzed package, as well as a variety of indicators to evaluate similarity or utility of experiments with respect to different aspects that may be important to the package evaluation.

#### *6.4.6.2 Bias determination*

Examine the applicant's results for the calculations for the benchmark experiments and the method used to account for biases and bias uncertainties, including the contribution from uncertainties in experimental data.

Confirm that the applicant analyzed a sufficient number of appropriate benchmark experiments and used the results of these benchmark calculations to determine an appropriate bias and bias uncertainty for the package calculations.[5] Confirm that the applicant evaluated the benchmark analysis results for trends in the bias with respect to parameter variations (such as pitch-to-rod-diameter ratio, assembly separation, reflector material, neutron-absorber material, H/X ratio, energy of average lethargy of neutrons causing fission). Evaluate the applicant's trending analysis to verify that the analysis considers appropriate subsets of the entire selection of benchmarks. For example, for a selection of experiments that includes some with neutron-absorber materials and some without absorber materials, the trend in bias for the entire selection of experiments may differ significantly versus the bias trend for the subset of experiments that include neutron-absorber materials. Verify that only negative biases (results that underpredict $k_{eff}$) are considered, with positive bias results (values that decrease $k_{eff}$ when applied) treated as zero bias. Confirm that the applicant has determined the biases and bias uncertainties versus the measured (i.e., experimentally determined) $k_{eff}$ values of the experiments, which may not always be unity or 1.0.

Also verify that the applicant demonstrates that the ranges of applicability of the experiments and bias evaluation adequately cover the package evaluations for the parameters important to criticality safety and that the coverage within the range of applicability is also adequate. Verify that the applicant justified any extrapolation, if done, of the bias and bias uncertainty beyond the ranges of applicability. Verify that the applicant also justified the appropriateness of the bias and bias uncertainty and trending analysis for areas within the range of applicability where data (experiments and calculation results) are limited or significant gaps exist between clusters of data, particularly if the package evaluation results for the higher reactivity configurations are in these gaps. For cases where extrapolation is necessary or data in the range of applicability are limited, confirm that the applicant considered the need to include additional margin in the analyses or uncertainty in the bias. NUREG/CR-5661 and NUREG/CR-6361 provide additional information on determining a bias and its range of applicability.

Confirm that the applicant's evaluation demonstrates that the benchmark calculations are adequately converged and addresses statistical uncertainties in the benchmark calculations. Apply the guidance in Section 6.4.3.4 of this SRP chapter regarding convergence and statistical uncertainties for the applicant's package calculations to the evaluation of the applicant's benchmark calculation.

### 6.4.7 Burnup Credit Evaluation for Commercial Light-Water Reactor Spent Nuclear Fuel

The regulations in 10 CFR Part 71 require that SNF remain subcritical in transportation. While unirradiated reactor fuel ("fresh fuel") has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transportation packages, the nuclide composition changes as the fuel is irradiated in the reactor. Ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to

[5] The benchmark and bias determination methods described in this SRP and related references for criticality safety analyses are based on an analysis of a sufficient number of experiments for which statistical normality has been demonstrated. For experiment sets for which statistical normality has not been demonstrated, including sets that are too few in number to enable this demonstration, the applicant and the staff should use other appropriate statistical methods to evaluate the benchmarks used in the application.

decrease. In the criticality safety analysis, allowance for the decrease in fuel reactivity resulting from irradiation is termed "burnup credit."

This section provides recommendations to the NRC reviewer for accepting, on a design-specific basis, a burnup credit approach in the criticality safety analysis of PWR SNF transportation packages. The guidance represents one method for demonstrating compliance with the criticality safety requirements in 10 CFR Part 71 using burnup credit. Follow this guidance to determine whether the applicant has provided reasonable assurance that the transportation package meets the applicable criticality safety regulations in 10 CFR Part 71. Consider alternative methodologies applicants propose on a case-by-case basis, using this guidance to the extent practicable.

The following recommendations were developed with intact fuel as the basis but may also apply to fuel that is not intact. If an applicant requests burnup credit for fuel that is not intact, apply the recommendations provided below, as appropriate, to account for uncertainties that can be associated with fuel that is not intact, and establish an isotopic inventory and assumed fuel configuration for the as-designed package and for normal conditions of transport and hypothetical accident conditions that bound the uncertainties.

The recommendations in this chapter do not include burnup credit for BWR fuel assemblies, as the technical basis for BWR burnup credit in SNF transportation packages has not been fully developed. The NRC has initiated a research project to obtain that technical basis. BWR fuel assemblies typically have neutron-absorbing material, typically gadolinium oxide ($Gd_2O_3$), mixed in with the uranium oxide of the fuel pellets in some rods. This neutron absorber depletes more rapidly than the fuel during the initial parts of its irradiation, which causes the fuel assembly reactivity to increase and reach a maximum value at an assembly-average burnup typically less than 20 gigawatt-days per metric ton of uranium (GWd/MTU). Then, reactivity decreases for the remainder of fuel assembly irradiation. Criticality analyses of BWR SNF pools typically employ what are known as "peak reactivity" methods to account for this behavior. NUREG/CR-7194, "Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems," issued April 2015, reviews several existing peak reactivity methods and demonstrates that a conservative set of analysis conditions can be identified and implemented to allow criticality safety analysis of BWR SNF assemblies at peak reactivity in SNF transportation packages. Consult NUREG/CR-7194 if the applicant uses peak reactivity BWR burnup credit methods in its criticality analysis.

This SRP does not address credit for BWR burnup beyond peak reactivity. The NRC is currently evaluating this type of burnup credit as part of a research program. The purpose of the program is to investigate methods for conservatively including such credit in a BWR criticality analysis for SNF transportation packages. The NRC does not recommend burnup credit beyond peak reactivity at this time. Consider conservative analyses of BWR burnup credit beyond peak reactivity on a case-by-case basis, consulting the latest research results in this area (i.e., NRC letter reports and NUREG/CRs).

The recommendations in this section also do not include burnup credit analyses for MOX or thorium SNF assemblies. Evaluate MOX burnup credit analyses on a case-by-case basis, noting that there are few MOX data available for isotopic depletion or criticality code validation. Analyses for MOX burnup credit should include substantial conservatism in the representation of MOX material in the criticality model and large $k_{eff}$ penalties for unvalidated fuel materials. Thorium fuel criticality analyses will require a depletion analysis to determine the most reactive fuel composition with irradiation. Similar to the situation for MOX SNF, code validation data are limited for thorium SNF, and criticality analyses should include large conservatisms and $k_{eff}$ penalties for unvalidated materials.

Attachment 6A to this SRP chapter provides more information on the technical bases for the recommendations described below.

*6.4.7.1 Limits for the certification basis*

Available data support allowance for burnup credit where the safety analysis is based on major actinide compositions only (i.e., actinide-only burnup credit) or limited actinide and fission product compositions (see Table 6-2) associated with uranium dioxide ($UO_2$) fuel irradiated in a PWR up to an assembly-average burnup value of 60 GWd/MTU and cooled out of reactor for a time period between 1 and 40 years. The range of available measured assay data for irradiated $UO_2$ fuel supports an extension of the certification basis up to 5.0 weight percent enrichment in uranium-235.

Within this range of parameters, carefully assess whether the analytic methods and assumptions used are appropriate, especially near the limits of the parameter ranges recommended here for the certification basis. Verify that the use of actinide and fission product compositions associated with burnup values or cooling times outside these specifications is accompanied by the measurement data or justified extrapolation techniques, or both, necessary to extend the isotopic validation and quantify or bound the bias and bias uncertainty. If the applicant credits neutron-absorbing isotopes other than those identified in Table 6-2, ensure that the applicant gives assurance that such isotopes are nonvolatile, nongaseous, and relatively stable and provides analyses to determine the additional depletion and criticality code bias and bias uncertainty associated with these isotopes.

A certificate condition indicating the time limit on the validity of the burnup credit analysis may be necessary in light of the possible use of the package to transport SNF that has been in storage for an extended time. Such a condition would depend on the type of burnup credit and the credited post-irradiation decay time.

*6.4.7.2 Model assumptions*

Confirm that the applicant calculated the actinide and fission product compositions used to determine a value of $k_{eff}$ using fuel design and reactor operating parameter values that appropriately encompass the range of design and operating conditions for the proposed contents. Verify that the applicant calculated the $k_{eff}$ value using models and analysis assumptions that allow accurate representation of the physics in the package, as discussed in Section 6A.4 of Attachment 6A to this chapter of the SRP. Pay attention to the need to do the following:

- Account for and effectively model the axial and horizontal variation of the burnup within an SNF assembly (e.g., the selection of the axial burnup profiles, number of axial material zones).

- Consider the potential for increased reactivity because of the presence of burnable absorbers or control rods (fully or partially inserted) during irradiation.

| Table 6-2 Recommended Set of Nuclides for Burnup Credit | |
|---|---|
| **Type of Burnup Credit** | **Recommended Set of Nuclides** |
| Actinide-only burnup credit | $^{234}U$, $^{235}U$, $^{238}U$, $^{238}Pu$, $^{239}Pu$, $^{240}Pu$, $^{241}Pu$, $^{242}Pu$, $^{241}Am$ |
| Additional nuclides for actinide-plus-fission product burnup credit | $^{95}Mo$, $^{99}Tc$, $^{101}Ru$, $^{103}Rh$, $^{109}Ag$, $^{133}Cs$, $^{143}Nd$, $^{145}Nd$, $^{147}Sm$, $^{149}Sm$, $^{150}Sm$, $^{151}Sm$, $^{152}Sm$, $^{151}Eu$, $^{153}Eu$, $^{155}Gd$, $^{236}U$, $^{237}Np$, $^{243}Am$ |

- Account for the irradiation environment factors to which the proposed assembly contents were exposed, including fuel temperature, moderator temperature and density, soluble boron concentration, specific power, and operating history.

YAEC-1937, “Axial Burnup Profile Database for Pressurized Water Reactors,” issued May 1997, provides representative data that can be employed for establishing profiles for use in the safety analysis. However, exercise care when reviewing profiles intended to bound the range of potential $k_{eff}$ values for the proposed contents for each burnup range, particularly near the upper end of the certification-basis parameter ranges stated in this guidance. NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” issued March 2003, provides additional guidance on selecting axial profiles.

A design-basis modeling assumption, where the assemblies are exposed during irradiation to the maximum (neutron absorber) loading of burnable poison rod assemblies (BPR) for the maximum burnup, encompasses all assemblies that may or may not have been exposed to BPRs. Such an assumption in the safety analysis should also encompass the impact of exposure to fully inserted or partially inserted control rods in typical domestic PWR operations. Assemblies exposed to atypical insertions of control rods (e.g., full insertion for one full cycle of reactor operation) should not be loaded unless the safety analysis explicitly considers such operational conditions. If the assumed BPR exposure is less than the maximum for which burnup credit is requested, confirm that the applicant has provided a justification commensurate with the selected value. For example, the lower the exposure, the greater the need to (i) support the assumption with available data, (ii) indicate how administrative controls would prevent a misload of an assembly exposed beyond the assumed value, and (iii) address such misloads in a misload analysis.

For assemblies exposed to integral burnable absorbers, the appropriate analysis assumption for absorber exposure varies depending on burnup and absorber material. The appropriate assumption may be to neglect the absorber while keeping the other assembly parameters (e.g., enrichment) the same for some absorber materials or for exposures up to moderate burnup levels (typically 20–30 GWd/MTU). Thus, a safety analysis including assemblies with integral burnable absorbers should include justification of the absorber exposure assumptions used in the analysis. For assemblies exposed to flux suppressors (e.g., hafnium suppressor inserts) or combinations of integral absorbers and BPRs or control rods, the safety analysis should use assumptions that provide a bounding safety basis, in terms of the effect on package $k_{eff}$, for those assemblies.

Confirm that the applicant’s evaluation includes analyses that use irradiation conditions that produce bounding values for $k_{eff}$, as discussed in Section 6A.4 of Attachment 6A to this SRP chapter. The bounding conditions may differ for actinide-only burnup credit versus actinide-plus-fission product burnup credit and may depend on the characteristics of the SNF intended to be transported in the package (e.g., all PWR assemblies versus a site-specific population). Contents specifications tied to the actual reactor operating conditions may be needed unless the operating condition values used in the evaluation can be justified as those that produce the maximum $k_{eff}$ values for the proposed SNF contents.

*6.4.7.3 Code validation—isotopic depletion*

Confirm that the applicant validated the computer codes used to calculate isotopic depletion. A depletion computer code is used to determine the concentrations of the isotopes important to burnup credit. To ensure accurate criticality calculation results, the selected code should be validated and the bias and bias uncertainty of the code should be determined at a 95-percent probability, 95-percent confidence level. Ensure that the application reflects the following considerations in the selection of the code and code validation approach for the fuel-depletion analysis.

The selected depletion code and cross-section library should be capable of accurately modeling the fuel geometry and the neutronic characteristics of the environment in which the fuel was irradiated. Two-dimensional depletion codes have been effectively used in burnup credit analyses. Although one-dimensional codes have been used in some applications and suffice for making assembly-average isotopic predictions for fuel burnup, they are limited in their ability to model increasingly complex fuel assembly designs and generally produce larger bias and bias-uncertainty values because of the approximations necessary in the models. Section 6A.4 of Attachment 6A to this SRP chapter discusses in detail the modeling considerations for the code validation analyses.

The destructive radiochemical assay (RCA) data selected for code validation should include detailed information about the SNF samples. This information should include the pin location in the assembly, axial location of the sample in the pin, any exposure to strong absorbers (control rods, BPRs), the boron letdown, moderator temperature, specific power, and any other cycle-specific data for the cycles in which the sample was irradiated. Some RCA data are not suitable for depletion code validation, because the depletion histories or environments of these samples are either difficult to accurately define in the code benchmark models or are unknown. NUREG/CR-7108, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions," issued April 2012, provides a recommended set of RCA data suitable for depletion-code validation.

The selected code validation approach should be adequate for determining the bias and bias uncertainty of the code for the specific application. The burnup credit analysis results should be adjusted using the bias and bias uncertainty determined for the fuel-depletion code, accounting for any trends of significance with respect to different control parameters, such as burnup-to-enrichment ratio or ratio of uranium-235 to plutonium-239. NUREG/CR-6811, "Strategies for Application of Isotopic Uncertainties in Burnup Credit," issued June 2003, provides several methods the NRC finds acceptable for isotopic-depletion validation, including the isotopic correction factor, direct-difference, and Monte Carlo uncertainty sampling methods. Section 6A.5 of Attachment 6A to this SRP chapter discusses in detail the advantages and disadvantages of these methods. In general, the isotopic correction factor method is considered to be the most conservative because individual nuclide composition uncertainties are represented as worst case. The direct-difference method provides a realistic "best estimate" of the depletion-code bias and bias uncertainty, in terms of difference in $k_{eff}$ ($\Delta k_{eff}$). The Monte Carlo uncertainty sampling method is more complex and computationally intensive than the other methods, but it provides a way to use the limited measurement data sets for some nuclides. NUREG/CR-7108 gives detailed descriptions of the direct-difference and Monte Carlo uncertainty sampling methods.

Instead of an explicit benchmarking analysis, the applicant may use the bias ($\beta_i$) and bias uncertainty ($\Delta k_i$) values estimated in NUREG/CR-7108 using the Monte Carlo uncertainty sampling method, as shown in Tables 6-3 and 6-4. These values may be used directly, provided that all of the following are true:

- The applicant uses the same depletion code and cross-section library as used in NUREG/CR-7108 (SCALE/TRITON and the ENDF/B-V or ENDF/B-VII cross-section library).

- The applicant can justify that its transportation package design is similar to the hypothetical 32-PWR-assembly-capacity, generic burnup credit cask (GBC-32) system design (NUREG/CR-6747, "Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit," issued October 2001) used as the basis for the NUREG/CR-7108 isotopic-depletion validation.

- Credit is limited to the specific nuclides listed in Table 6-2.

Section 6A.5 of Attachment 6A to this SRP chapter discusses in detail the technical basis for the restrictions on directly applying the bias and bias-uncertainty values. Bias values should be added to the calculated package $k_{eff}$, while bias-uncertainty values may be statistically combined with other independent uncertainties. Table 6-5 summarizes the recommendations related to isotopic-depletion-code validation

#### *6.4.7.4 Code validation—$k_{eff}$ determination*

<u>Actinide-Only Credit</u>

Credit should be limited to the specific nuclides listed in Table 6-2 for actinide-only burnup credit. Criticality validation for these actinides should be based on the critical experiments described in NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," issued September 2008, also known as the HTC data, supplemented by MOX critical experiments as appropriate. NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality ($k_{eff}$) Predictions," issued April 2012, contains a detailed discussion of available sets of criticality validation data for actinide isotopes and the relative acceptability of these sets. Note that NUREG/CR-7109 demonstrates that fresh $UO_2$ experiments are not applicable to burned fuel compositions.

Verify that the applicant determined the bias and bias uncertainty associated with actinide-only burnup credit according to the guidance in NUREG/CR-6361. This guidance includes criteria for the selection of appropriate benchmark data sets, as well as statistics and trending analysis for the determination of criticality code bias and bias uncertainty. Section 6 of NUREG/CR-7109 provides an example of bias and bias uncertainty determination for actinide-only burnup credit.

<u>Fission Product and Minor Actinide Credit</u>

Confirm that the applicant has determined an adequate and conservative bias and bias uncertainty associated with fission product and minor actinide credit. The applicant may credit the minor actinide and fission product nuclides listed in Table 6-2, provided that the bias and bias uncertainty associated with the major actinides is determined as described above. The bias from these minor actinides and fission products is conservatively covered by 1.5 percent of

| **Table 6-3 Isotopic $k_{eff}$ Bias Uncertainty ($\Delta k_i$) for the Representative PWR SNF System Model Using ENDF/B-VII Data ($\beta_i$ = 0) as a Function of Assembly Average Burnup** | | |
|---|---|---|
| **Burnup (BU) Range (GWd/MTU)** | **Actinides Only, $\Delta k_i$** | **Actinides and Fission Products $\Delta k_i$** |
| 0≤BU<5 | 0.0145 | 0.0150 |
| 5≤BU<10 | 0.0143 | 0.0148 |
| 10≤BU<18 | 0.0150 | 0.0157 |
| 18≤BU<25 | 0.0150 | 0.0154 |
| 25≤BU<30 | 0.0154 | 0.0161 |
| 30≤BU<40 | 0.0170 | 0.0163 |
| 40≤BU<45 | 0.0192 | 0.0205 |
| 45≤BU<50 | 0.0192 | 0.0219 |
| 50≤BU≤60 | 0.0260 | 0.0300 |

| **Table 6-4 Isotopic $k_{eff}$ Bias ($\beta_i$) and Bias Uncertainty ($\Delta k_i$) for the Representative PWR SNF System Model Using ENDF/B-V Data as a Function of Assembly Average Burnup** | | |
|---|---|---|
| **Burnup (BU) Range (GWd/MTU)[a]** | **$\beta_i$ for Actinides and Fission Products** | **$\Delta k_i$ for Actinides and Fission Products** |
| 0≤BU<10 | 0.0001 | 0.0135 |
| 10≤BU<25 | 0.0029 | 0.0139 |
| 25≤BU≤40 | 0.0040 | 0.0165 |
| [a]Bias and bias uncertainties associated with ENDF/B-V data were calculated for a maximum of 40 GWd/MTU. For higher burnups, applicants should provide an explicit depletion-code validation analysis using one of the methods described in Attachment 6A to this SRP chapter, along with appropriate RCA data. | | |

| **Table 6-5 Summary of Code Validation Recommendations for Isotopic Depletion** | |
|---|---|
| **Applicant's Approach** | **Recommendation** |
| Applicant uses SCALE/TRITON and the ENDF/B-V or -VII cross-section library, and demonstrates that the design application is similar to GBC-32. | Use code bias and bias uncertainty values from Tables 6-3 and 6-4 of this SRP. |
| - or - | |
| Applicant uses other code or cross-section library, or both, or design application is not similar to GBC-32. | Use either isotopic-correction factor or direct-difference method to determine code bias and bias uncertainty. |

their worth. Because of the conservatism in this value, no additional uncertainty in the bias needs to be applied. This estimate is appropriate if the applicant does the following:

- uses the SCALE code system with the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross-section libraries, or MCNP5 or MCNP6 with the ENDF/B-V, ENDF/B-VI, ENDF/B-VII, or ENDF/B-VII.1 cross-section libraries

- justifies that its transportation package design is similar to the hypothetical GBC-32 system design (NUREG/CR-6747) used as the basis for the NUREG/CR-7109 criticality validation

- demonstrates that the credited minor actinide and fission product worth is no greater than 0.1 in $k_{eff}$

For well-qualified, industry-standard code systems other than SCALE or MCNP, the applicant may use a conservative estimate for the bias associated with minor actinide and fission product nuclides of 3.0 percent of their worth. If the applicant uses a minor actinide and fission product bias less than 3.0 percent, ensure that the application includes additional justification that the lower value is an appropriate estimate of the bias associated with that code system (e.g., a minor actinide and fission product worth comparison to SCALE results or an analysis similar to that described in NUREG/CR-7109 or NUREG/CR-7205, “Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP $K_{eff}$ Calculations for PWR Burnup Credit Casks”). Table 6-6 summarizes the recommendations related to minor actinide and fission product code validation for $k_{eff}$ determination. For actinide criticality validation in all cases, the applicant should perform criticality code validation analyses to determine bias and bias uncertainty associated with actinides using HTC critical experiments, supplemented by applicable MOX critical experiments. Ensure that the applicant performed the validation analyses correctly and adequately.

*6.4.7.5 Loading curve and burnup verification*

Confirm that the applicant provided burnup credit loading curves to determine which fuel assemblies may be loaded into the transportation package. Confirm that the burnup-credit evaluations include loading curves that specify the minimum required assembly-average burnup as a function of initial enrichment for the purpose of loading the SNF transportation package. Confirm that the applicant has established separate loading curves for each content or set of contents. For example, a separate loading curve should be provided for each minimum cooling time to be considered in the package loading. In addition, confirm that the applicant justified the applicability of the loading curve to bound various fuel types or burnable absorber loadings.

Ensure that the Criticality Evaluation and Package Operations sections in the application include performance of burnup verification to ensure that a transportation package evaluated using burnup credit is not loaded with an assembly more reactive than those included in the loading criteria. Verification should include a measurement that confirms the reactor record for each assembly. Confirmation of reactor records using measurement of a sample of fuel assemblies will be considered if the sampling method can be justified in comparison to measuring every assembly.

The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement. NUREG/CR-6998, “Review of Information for Spent Nuclear Fuel Burnup Confirmation,” issued December 2009, contains bounding estimates of reactor record burnup uncertainty.

Measurements should be correlated to reactor record burnup, enrichment, and cooling time values. Measurement techniques should account for any measurement uncertainty (typical within a 95-percent confidence interval) in confirming reactor burnup records. The application should also include a database of measured data (if measuring a sampling of fuel assemblies) to justify the adequacy of the procedure compared to procedures that measure each assembly.

| Table 6-6 Summary of Minor Actinide and Fission Product Code Validation Recommendations for $k_{eff}$ Determination | |
|---|---|
| **Applicant's Approach** | **Recommendation** |
| Applicant uses SCALE code system with ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross-section libraries, or MCNP5 or MCNP6 with the ENDF/B-V, ENDF/B-VI, ENDF/B-VII, or ENDF/B-VII.1 cross-section libraries; design application is similar to GBC-32; and credited minor actinide and fission product worth is < 0.1 in $k_{eff}$. | Use bias equal to 1.5 percent of minor actinide and fission product worth. |
| - or - | |
| Applicant uses other code with ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross-section libraries; design application is similar to GBC-32; and credited minor actinide and fission product worth is < 0.1 in $k_{eff}$. | Use bias equal to 3.0 percent of minor actinide and fission product worth, or provide justification for lower number. |
| - or - | |
| Applicant uses cross-section library other than ENDF/B-V, ENDF/B-VI, or ENDF/B-VII; design application is not similar to GBC-32; or credited minor actinide and fission product worth is > 0.1 in $k_{eff}$. | Perform explicit criticality code validation for minor actinide and fission product nuclides. |

Misload Analyses

Misload analyses may be performed in lieu of a burnup measurement. A misload analysis should address potential events involving the placement of assemblies into the SNF transportation package that do not meet the proposed loading criteria. Confirm that the applicant has demonstrated that the package remains subcritical for misload conditions, including calculation biases, uncertainties, and an appropriate administrative margin that is not less than 0.02 Δk. If any administrative margin less than the normal 0.05 Δk is used, verify that the application provides an adequate justification that includes the level of conservatism in the depletion and criticality calculations, sensitivity of the package to further upset conditions, and the level of rigor in the code validation methods.

If used, ensure that the misload analysis considers (i) misloading of a single, severely underburned assembly and (ii) misloading of multiple, moderately underburned assemblies.

The severely underburned assembly for the single misload analysis should be chosen such that the misloaded assembly's reactivity bounds 95 percent of the discharged PWR fuel population considered unacceptable for loading in the transportation package with 95-percent confidence. The moderately underburned assemblies for the multiple-misload analysis should be assumed to make up at least 50 percent of the package payload and should be chosen such that the misloaded assemblies' reactivity bounds 90 percent of the total discharged PWR fuel population. The NRC finds the results of the most recent U.S. Energy Information Administration's "Nuclear Fuel Data Survey" (RW-859) or later similar fuel data sources (i.e., GC-859), acceptable to estimate the discharged fuel population characteristics.

Also ensure that the misload analysis considers the effects of placing the underburned assemblies in the most reactive positions within the loaded package (e.g., middle of the fuel basket). If removable nonfuel absorbers were credited as part of the criticality safety analysis (e.g., poison rods added to guide tubes), ensure that the misload analysis considers misloading of these absorbers. Additionally, ensure that the misload analysis considers assemblies with greater burnable absorber or control-rod exposure than assumed in the criticality analysis if the assumed exposure is not bounding. NUREG/CR-6955, "Criticality Analysis of Assembly

Misload in a PWR Burnup Credit Cask," issued January 2008, illustrates the magnitude of $k_{eff}$ changes that can be expected as a result of various misloads in a theoretical GBC-32 SNF storage and transportation system.

Administrative Procedures

Confirm that the applicant has included administrative procedures for loading that will protect against misloads. Ensure that the misload analysis is coupled with additional administrative procedures to ensure that the SNF transportation package will be loaded with fuel that is within the specifications of the approved contents. Procedures the applicant may consider to protect against misloads in transportation packages that rely on burnup credit for criticality safety include the following:

- verification of the location of high-reactivity fuel (i.e., fresh or severely underburned fuel) in the SNF pool, both before and after loading
- qualitative verification that the assembly to be loaded is burned (visual or gross measurement)
- under an NRC-approved quality assurance program, verification before shipment of the inventory and loading records of a canister or storage cask that was previously loaded and placed into dry storage and that is to be shipped in or as the package
- quantitative measurement of any fuel assemblies without visible identification numbers
- independent, third-party verification of the loading process, including the fuel selection process and generation of the fuel move instructions

Table 6-7 summarizes the recommendations for burnup verification.

### 6.4.8 Appendix

Confirm that the application includes a list of references, copies of applicable references if not generally available, computer code descriptions, input and output files, test results, and any other appropriate supplemental information. The applicant may include these items in an appendix to the Criticality Evaluation section of the application.

## 6.5 Evaluation Findings

Prepare evaluation findings upon satisfaction of the regulatory requirements in Section 6.3. 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:

F6-1 The staff has reviewed the package and concludes that the application adequately describes the package contents and the package design features that affect nuclear criticality safety in compliance with 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b) and provides an appropriate and bounding evaluation of the package's criticality safety performance in compliance with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a).

| Table 6-7 Summary of Burnup Verification Recommendations | |
|---|---|
| **Applicant's Approach** | **Recommendation** |
| Applicant takes burnup verification measurement. | Measure each assembly to be loaded or a statistically significant sample of assemblies. |
| - or - | |
| Applicant conducts misload analysis and provides additional administrative procedures. | Analyze misload of fuel assembly that bounds reactivity of 95 percent of underburned fuel population with 95-percent confidence. |
| | Analyze misload of 50 percent of package capacity with fuel assemblies with reactivity that bounds 90 percent of total fuel population. |
| | Include additional administrative procedures as part of transportation package loading. |

F6-2 [if applicable] The staff has reviewed the package and concludes that the application identifies the codes and standards used in the package's criticality safety design in compliance with 10 CFR 71.31(c).

F6-3 The staff has reviewed the package and concludes that the application specifies the number of packages that may be transported in the same vehicle through provision of an appropriate CSI in compliance with 10 CFR 71.35(b). [if applicable] The applicant specifies an appropriate CSI for each type of fissile content.

F6-4 The staff has reviewed the package and concludes that the applicant used packaging features and package contents configurations and materials properties in the criticality safety analyses that are consistent with and bounding for the package's design basis, including the effects of the normal conditions of transport and the relevant accident conditions in 10 CFR 71.55(f), 71.73, or 71.74 [select the relevant requirements]. The applicant has adequately identified the package configurations and material properties that result in the maximum reactivity for the single package and package array analyses.

F6-5 The staff has reviewed the package and concludes that the criticality evaluations in the application of a single package demonstrate that it is subcritical under the most reactive credible conditions, in compliance with 10 CFR 71.55(b), 71.55(d), and 71.55(e) [and 10 CFR 71.55(f) for fissile packages transported by air or 10 CFR 71.64(a)(1)(iii) for plutonium packages transported by air]. The evaluations in the application also demonstrate that the effects of the normal conditions of transport tests do not result in a significant reduction in the packaging's effectiveness in terms of criticality safety, in compliance with 10 CFR 71.43(f) and 10 CFR 71.55(d)(4) and, for Type B fissile packages, 10 CFR 71.51(a)(1). The evaluations in the application also demonstrate that the geometric form of the contents is not substantially altered under the normal conditions of transport tests, in compliance with 10 CFR 71.55(d)(2).

F6-6 The staff has reviewed the package and concludes that the criticality evaluation in the application of the most reactive array of 5N undamaged packages demonstrates that the array of 5N packages is subcritical under normal conditions of transport to meet the requirements in 10 CFR 71.59(a)(1).

F6-7 The staff has reviewed the package and concludes that the criticality evaluation in the application of the most reactive array of 2N packages subjected to the tests in 10 CFR 71.73 [or 10 CFR 71.74 for plutonium packages transported by air, per

10 CFR 71.64(a)(1)(iii)] demonstrates that the array of 2N packages is subcritical under hypothetical accident conditions in 10 CFR 71.73 [or under the accident conditions in 10 CFR 71.74] to meet the requirements in 10 CFR 71.59(a)(2).

F6-8 The staff has reviewed the package and concludes that the applicant's evaluations include an adequate benchmark evaluation of the calculations. The applicant identified and evaluated experiments that are relevant and appropriate for the package analyses and performed appropriate trending analyses of the benchmark calculation results. The applicant has determined an appropriate bias and bias uncertainties for the criticality evaluation of the package.

F6-9 The staff has reviewed the package and concludes that the application identifies the necessary special controls and precautions for transport, loading, unloading, and handling and, in case of accidents, compliance with 10 CFR 71.35(c). [For commercial SNF packages evaluated using burnup credit.] These controls include additional contents specifications (e.g., fuel loading curve(s), reactor operating parameters) and administrative procedures to prevent package misloads.

F6-10 The staff has reviewed the package and concludes that the evaluations in the application assume unknown properties of the fissile contents are at credible values that maximize neutron multiplication consistent with 10 CFR 71.83. [For commercial SNF packages evaluated using burnup credit.] This includes following the recommendations in Section 6.4.7 and Attachment 6A to this SRP chapter for crediting the burnup of the SNF contents.

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

> Based on review of the statements and representations in the application, the staff has reasonable assurance that the proposed package design and contents satisfy the nuclear criticality safety requirements in 10 CFR Part 71. In making this finding, the staff considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, and the staff's own independent confirmatory calculations.

## 6.6 References

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

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NUREG/CR-6811, U.S. Nuclear Regulatory Commission, “Strategies for Application of Isotopic Uncertainties in Burnup Credit,” ORNL/TM-2001/257, Oak Ridge National Laboratory, Oak Ridge, TN, June 2003.

NUREG/CR-6835, U.S. Nuclear Regulatory Commission, “Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks,” ORNL/TM-2002/255, Oak Ridge National Laboratory, Oak Ridge, TN, September 2003.

NUREG/CR-6955, U.S. Nuclear Regulatory Commission, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” ORNL/TM-2004/52, Oak Ridge National Laboratory, Oak Ridge, TN, January 2008.

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