5 SHIELDING EVALUATION

5.1 Review Objective

The objective of this evaluation is to verify that the design of Type B transportation packages meets the external radiation requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, "Packaging and Transportation of Radioactive Materials."

5.2 Areas of Review

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

- description of shielding design
    - shielding design features
    - codes and standards
    - summary tables of maximum external radiation levels
- radioactive materials and source terms
    - source-term calculation methods
    - gamma sources
    - neutron sources
- shielding model and model specifications
    - configuration of source and shielding
    - material properties
- shielding evaluation
    - methods
    - code input and output data
    - fluence-rate-to-radiation-level conversion factors
    - external radiation levels
    - confirmatory analyses

5.3 Regulatory Requirements and Acceptance Criteria

This section summarizes those aspects of 10 CFR Part 71 that are relevant to the review areas, as identified in this standard review plan (SRP) chapter. The NRC staff reviewer should refer to the exact language in the listed regulations. Table 5-1 identifies the regulatory requirements that are relevant to the areas of review covered in this chapter. Table 5-2 identifies the current external radiation level limits in 10 CFR 71.47, "External Radiation Standards for All Packages," that apply to exclusive-use and nonexclusive-use shipments. The table also states the limit in 10 CFR 71.51(a)(2), which applies to both exclusive-use and nonexclusive-use shipments.

Table 5-1 Relationship of Regulations and Areas of Review for Transportation Packages

Areas of Review	10 CFR Part 71 Regulations						
	71.31	71.33	71.35(a)	71.41(a)	71.43(f)	71.47	71.51(a)
Description of shielding design	(a)(1),(b),(c)	(a)			●	●	●
Radioactive materials and source terms	(a)(1),(b)	(b)				●	●
Shielding model and model specifications	(c)			●	●	●	●
Shielding evaluation	(a)(2),(b),(c)		●	●	●	●	●
Areas of Review	**10 CFR Part 71 Regulations**						
	71.61	**71.63**	**71.64(a)(1)(ii),(b)(2)**	**71.71**	**71.73**	**71.74**	**Part 71, App. A**
Description of shielding design			●				
Radioactive materials and source terms		●	●				●
Shielding model and model specifications	●		●	●	●	●	
Shielding evaluation	●		●	●	●	●	●

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

Table 5-2 Package and Vehicle External Radiation Level Limits[a] (continued)

Transport vehicle use	Nonexclusive	Exclusive Use		
Conditions of transport	Open or Closed	Open (flatbed)	Open with Enclosure[b]	Closed
Package (or freight container) limits:				
External surface	2 mSv/h [200 mrem/h]	2 mSv/h [200 mrem/h]	10 mSv/h [1,000 mrem/h]	10 mSv/h [1,000 mrem/h]
1 meter [40 inches] from external surface[c]	0.1 mSv/h [10 mrem/h]	No Limit		
Roadway or railway vehicle (or freight container) limits:				
Any point on outer surface	N/A	N/A	N/A	2 mSv/h [200 mrem/h]
Vertical planes projected from outer edges	N/A	2 mSv/h [200 mrem/h]	2 mSv/h [200 mrem/h]	N/A
Top of ...	N/A	Load: 2 mSv/h [200 mrem/h]	Enclosure: 2 mSv/h [200 mrem/h]	Vehicle: 2 mSv/h [200 mrem/h]
2 meters [80 inches] from ...	N/A	Vertical Planes: mSv/h [10 mrem/h]	Vertical Planes: 0.1 mSv/h [10 mrem/h]	Outer Lateral Surfaces: mSv/h [10 mrem/h]
Underside of …	N/A	Vehicle below load: 2 mSv/h [200 mrem/h]		

Table 5-2 Package and Vehicle External Radiation Level Limits[a] (continued)		
Transport vehicle use	**Nonexclusive**	**Exclusive Use**
Occupied spaces	N/A	Cab or sleeper: 0.02 mSv/h [2 mrem/h][d]
Hypothetical accident, 1 meter [40 inches] from package external surface	10 mSv/h [1,000 mrem/h]	

Note: This table is not a substitute for NRC or U.S. Department of Transportation (DOT) regulations on the transport of radioactive materials. See NRC and DOT regulations for current requirements (10 CFR 71.47 and 49 CFR 173.441(a) and (b), respectively).
N/A = not applicable; mrem/h = millirem per hour; mSV/h = millisieverts per hour.
[a]The limits in this table do not apply to excepted packages and empty packages under DOT shipping regulations (49 CFR Part 173, Subpart I, "Class 7 Radioactive Materials"; specifically, 49 CFR 173.421, "Excepted Packages for Limited Quantities of Class 7 (Radioactive) Materials," 173.422, "Additional Requirements for Excepted Packages Containing Class 7 (Radioactive) Materials," 173.423, "Requirements for Multiple Hazard Limited Quantity Class 7 (Radioactive) Materials," 173.424, "Excepted Packages for Radioactive Instruments and Articles," 173.425, "Table of Activity Limits—Excepted Quantities and Articles," 173.426, "Excepted Packages for Articles Containing Natural Uranium or Thorium," and 173.428, "Empty Class 7 (Radioactive) Materials Packaging").
[b]Securely attached (to vehicle), access-limiting enclosure; package personnel barriers are considered as enclosures. See discussion in Section 5.4.1.2 of this SRP chapter for further information.
[c]Transport index may not exceed 10.
[d]Does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10 CFR 20.1502, with exposures and doses controlled and monitored under a radiation protection program satisfying the requirements of 10 CFR Part 20.

The Shielding Evaluation section of the application should describe and analyze the packaging design features and package configurations that are relevant to shielding, including items that increase radiation levels (e.g., streaming paths) as well as those that reduce radiation levels. This section of the application should also discuss how these features and the results of shielding analyses demonstrate compliance with NRC regulations.

The package design and contents descriptions in the application should be sufficient to provide an adequate basis for the shielding evaluation and to allow for independent review, including confirmatory calculations. Depending on package contents, this includes descriptions that allow for analyzing of secondary radiations such as neutrons from subcritical multiplication in spent nuclear fuel (SNF) contents and contributions of radioactive daughters in source and waste packages. The contents descriptions should be consistent with the assumptions made about the contents in the shielding evaluation.

For some packages, it may be desirable to add supplemental gamma shielding as an auxiliary component of the packaging. In these cases, the application must specifically address the inclusion of such shielding to the package in the package description to meet 10 CFR 71.33(a). The certificate of compliance (CoC) would need to specifically authorize the use of this shielding. Additionally, the application must demonstrate that this shielding remains effective during the applicable conditions (10 CFR 71.71, "Normal Conditions of Transport," 10 CFR 71.73, "Hypothetical Accident Conditions," 10 CFR 71.74, "Accident Conditions for Air Transport of Plutonium") to meet 10 CFR 71.35(a). NRC Information Notice 83-10, "Clarification of Several Aspects Relating to Use of NRC-Certified Transport," dated March 11, 1983, presents additional information regarding the use of supplemental shielding.

The application should describe the model(s) used in the shielding analysis to enable independent review, including confirmatory calculations. The model(s) should be consistent with the package design, the contents descriptions, and how the package is intended to be fabricated and operated, as described in the acceptance tests and package operations sections of the application. The model descriptions should address streaming paths and other locations of shielding changes (e.g., radial surface locations beyond the axial extent of neutron shields, locations of reduced gamma shielding component thickness) and possible positions of package contents in relation to the package's features. The descriptions should include the specifications of the package's shielding components. For nonstandard materials like proprietary neutron shielding and neutron absorbers credited in the analyses, this includes material composition specifications in addition to dimensional specifications. The application should describe differences in package features, dimensions, and material properties for normal conditions of transportation calculations and the hypothetical accident conditions calculations that could affect shielding performance. For example, polymer-based neutron shields usually are assumed to be gone for hypothetical accident conditions. Also, personnel barriers may be credited for normal conditions of transport calculations but not for hypothetical accident conditions.

The application should demonstrate that a package at the minimum shielding effectiveness allowed by the package design, including tolerances, will comply with the NRC regulations for the bounding radiation source(s) of the proposed package contents. The analysis should account for any increases in source terms with time, such as may occur with some package contents that produce radioactive daughters that may have greater source strengths or more penetrating radiation spectra. The analysis should be sufficiently detailed to show compliance for radiation levels at any point of the package surface and at the relevant distances from the package. For packages designed to be used for nonexclusive-use shipments, the analysis should show that the package will not exceed the nonexclusive-use radiation limits in 10 CFR 71.47(a). The NRC expects that packages evaluated to meet the nonexclusive-use limits will be designed, fabricated, and operated to meet these limits during package use. Otherwise, the analysis should show that the package will not exceed the exclusive-use radiation limits in 10 CFR 71.47(b) applicable to how the package is intended to be shipped (see Table 5-2).

The requirements in 10 CFR 71.47 state that each package offered for transportation must be designed and prepared for shipment so that under conditions normally incident to transportation, the package does not exceed the radiation level limits in 10 CFR 71.47(a), except as provided in 10 CFR 71.47(b). The NRC's practice is to ensure the analyses for compliance with the 10 CFR 71.47 limits include the impacts of the evaluations for normal conditions of transport described in 10 CFR 71.71. Inclusion of the impacts of these evaluations reasonably bounds the impacts of "conditions normally incident to transportation," though they are not necessarily the same thing. As described in 10 CFR 71.43(f) and 10 CFR 71.51(a)(1), the package must be designed, fabricated, and prepared for shipment so that under the 10 CFR 71.71 evaluations, the package's surface radiation levels do not significantly increase and the effectiveness of the packaging is not substantially reduced. As identified in the international regulations for radioactive materials transportation [see Specific Safety Requirements No. SSR-6, "Regulations for the Safe Transport of Radioactive Material", 2012 Edition, paragraph No. 648(b)], the international community interprets "significant increase" to mean an increase in excess of 20 percent of the package radiation levels in the preevaluation condition. If the application demonstrates that 10 CFR 71.47 limits are not exceeded for a package evaluated in accordance with 10 CFR 71.71, the NRC accepts the package as sufficient to meet the requirements in 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for the shielding evaluation.

The application should identify and describe, as applicable, the use of any industry codes and standards or NRC guidance as part of the package's shielding design and in the shielding evaluation. While applicants are not required to comply with NRC guidance, the use of NRC guidance is expected to facilitate the staff's review process in evaluating package designs and confirming compliance with NRC regulations.

The following documents also provide useful guidance regarding information the application should include in regard to the package's shielding design and the shielding evaluation:

- Regulatory Guide (RG) 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material," issued March 2005, Section 5, "Shielding Evaluation."

- NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," issued May 1998.

- NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," issued February 1996.

At a minimum, the application should present information consistent with this SRP and guidance in RG 7.9 and other necessary supplemental information used in confirming compliance with NRC regulations. In instances where an applicant has taken a different approach to specific provisions of NRC guidance, the application should provide the basis and justification for taking that approach. The application should include a list of references with applicable pages from referenced documents (providing copies if the references are not generally available); justification of assumptions and analytical procedures used in code models, code tests, and benchmarking results; descriptions of computer programs; sample input and output files supporting all major conclusions (e.g., an input or output file for each type of calculation, for different source or package configurations, and for the normal conditions of transport and for the hypothetical accident conditions); tabulations of source terms, radionuclide distributions, enrichment, fuel burnup rates, isotopic depletion, concentrations, and inventories; tabulations of flux rates; and fluence-to-radiation level conversion factors. The applicant may also consider including photographs of shielding components and assembly.

5.4 Review Procedures

The NRC conducts shielding reviews of Type B packages. This includes all radioactive materials for which the applicant seeks to obtain approval in a CoC as approved contents of the Type B package. If the applicant seeks to add materials that are of Type A quantities to the approved contents of the Type B packages, whether to be shipped alone in the package or together with Type B contents, review the application to ensure that the applicant has adequately evaluated the Type B package for these contents, applying the guidance in this SRP chapter as appropriate. The NRC does not conduct reviews of Type A packages as, with the exception of Type AF (fissile) packages, the regulations allow self-certification of these packages. For Type AF packages, shielding reviews are not necessary because, by the nature of the contents, radiation source terms and radiation levels for these packages are negligible.

Ensure that the applicant has described and evaluated the package design, including the shielding and the contents with their associated source terms, to meet all applicable external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions. For packages for the shipment of plutonium by air, ensure that the applicant has also evaluated the package design to meet the external radiation requirements in 10 CFR 71.64(a)(1)(ii) and (b)(2). For such a package, use methods and processes similar to those described in this chapter for evaluating compliance with the external radiation requirements for normal conditions of transport and hypothetical accident conditions to evaluate compliance with the requirements for air shipments of plutonium.

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 described in the review procedures in this SRP chapter, to ensure that the applicant 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 shielding evaluation and in a manner to meet the regulations. This includes ensuring that the acceptance tests include appropriate shield effectiveness tests for those packaging components relied on for shielding. Figure 5-1 illustrates the information flow and interdependency between the reviews for other sections of the application and the shielding evaluation review.

Also, as part of the review, ensure that the CoC includes appropriate conditions with respect to the package design, allowable package contents, package operations, and package acceptance and maintenance tests to ensure that the shielding performance of the package will be as designed and 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.

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," and the other appendices to this SRP, as applicable. Appendix A includes useful guidance that is specific to several of the package types, with the exception of Tritium-Producing Burnable Absorber Rod (TPBAR) packages, which the NRC certifies. Appendix E, Description and Review Procedures for Irradiated Tritium-Producing Burnable Absorber Rods Packages," includes guidance and other potentially useful information for reviews of TPBAR packages. Appendix B, "Differences between Thermal and Radiation Properties of MOX and LEU Radioactive Materials," and Appendix C, "Differences between Thermal and Radiation Properties of MOX and LEU Spent Nuclear Fuel," also provide useful information to inform reviews of mixed oxide (MOX) fresh fuel and MOX SNF packages, respectively. Except for the information in Appendix C for MOX SNF packages, guidance regarding SNF, including research reactor and commercial [both low-enriched uranium (LEU) and MOX] SNF, is contained within this chapter.

5.4.1 Description of Shielding Design

Ensure that the application includes information about the packaging design. This design information is typically captured in engineering drawings submitted in the General Information section in the application. RG 7.9, NUREG/CR-5502, and NUREG/CR-6407 provide information and describe the recommended format and technical contents for drawings submitted in package applications. Verify that the engineering drawings focus on and provide the necessary details for the features of the package and configuration(s) of components that are important in assessing the shielding performance of the package and demonstrating compliance with 10 CFR Part 71 regulations. These details include dimensions with tolerances as well as materials specifications with tolerances for shielding, such as proprietary neutron shields and other nonstandard materials, lead gamma shielding, and neutron absorbers credited in the analyses. The degree of specificity of the package component descriptions in the drawings should be commensurate with the stated safety functions and with the sensitivity of package

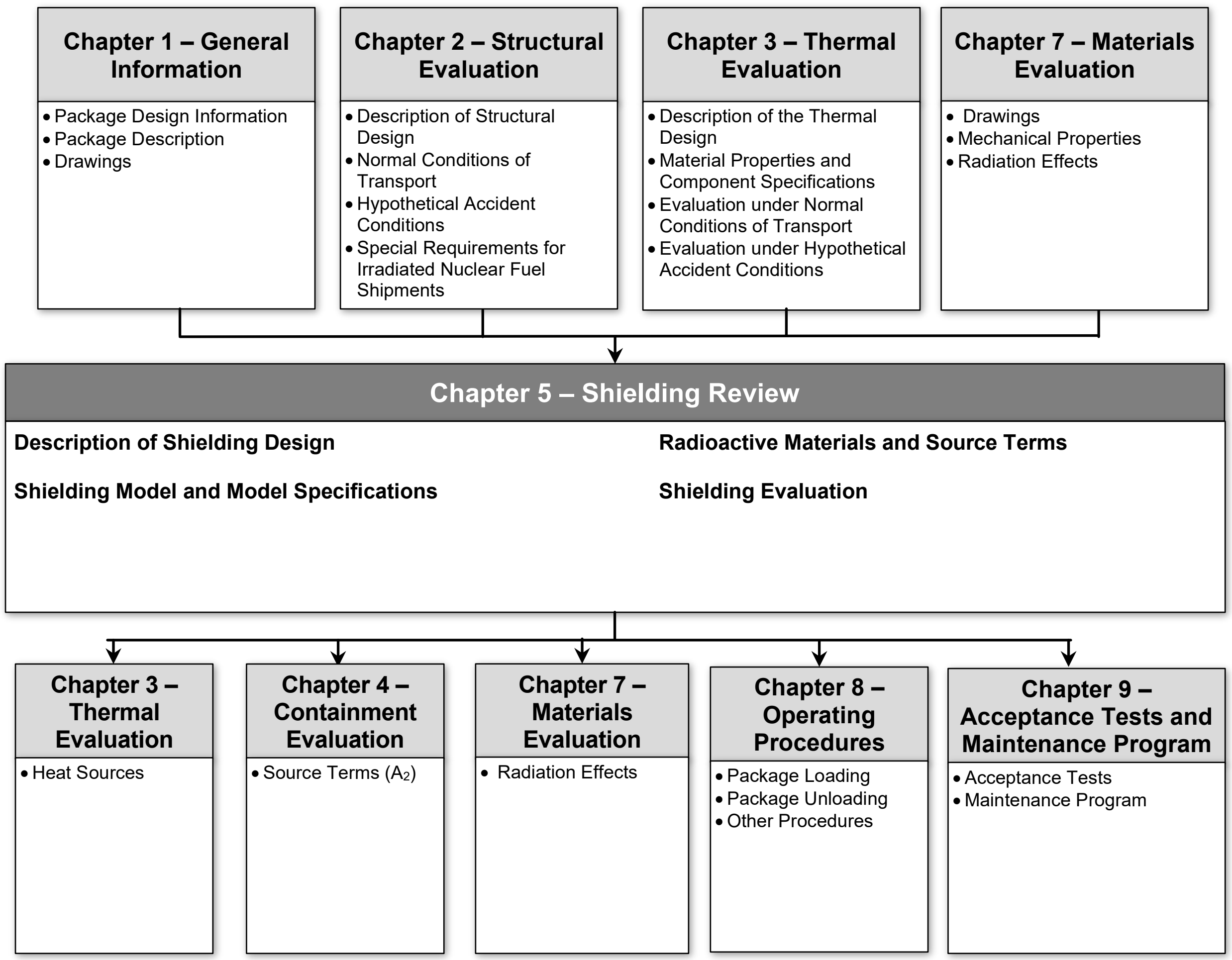

Figure 5-1 Information Flow for the Shielding Evaluation

shielding performance to the properties of the package components (material and dimensional properties, including tolerances). With regard to tolerances, ensure that the drawings specify reasonable tolerances for dimensions and material properties because packaging features may be subject to some variability in fabrication. Whatever tolerances are specified, ensure that the applicant's shielding analyses appropriately use these tolerances to determine maximum package radiation levels (see Sections 5.4.3.1 and 5.4.3.2 of this SRP chapter).

Review the description and evaluation of shielding design features in the Shielding Evaluation section of the application. Ensure that the description, including any sketches and figures, is consistent with that given in the General Information section of the application, including the engineering drawings. Verify that the application identifies any industry codes and standards or NRC guidance the applicant used in the shielding design and evaluation, and verify that the applicant used them properly.

5.4.1.1 Shielding design features

Ensure that the application's description of the shielding design features addresses those items that are important to evaluation of the package's shielding performance, including, but not limited to, the following topics:

- dimensions, tolerances, configurations, and densities of materials for neutron and gamma shielding and those packaging components that can affect package shielding performance; these components include both those that reduce package shielding performance (e.g., streaming paths) as well as those that enhance it, both components the applicant's shielding evaluation considered and those that it did not

- material composition specifications and tolerances on those specifications (e.g., minimum boron and hydrogen content) for nonstandard materials such as proprietary neutron-shield materials

- stability and potential deformation or materials properties changes of shielding materials if exposed to elevated temperatures

- materials and dimensional specifications, with their respective tolerances, of neutron absorbers that are credited in the shielding analyses; the materials specifications should include mass density, atomic density, or areal density of the absorbing material (e.g., boron-10)

- structural components that maintain the package contents in a fixed position within the package, whether for just normal conditions of transport or also for hypothetical accident conditions

- integrity of closure features and seals (and other relevant features) relied on to maintain package contents within certain packaging components; examples include seals or closures of internal containers loaded in the package for which the shielding evaluation assumes the package contents remain in the sealed containers and cannot spread to other areas in the package cavity. The application should include package operating procedures, acceptance tests, and maintenance program checks to ensure the closure features and seals do not allow migration of contents to unintended areas of the package cavity

- dimensions of the transport vehicle, and potentially the impact limiters, that are considered in the shielding evaluation when the applicant's evaluations are for demonstrating compliance with the exclusive-use limits

- appropriate dimensions and properties, including tolerances, of supplemental shielding of which an applicant may wish to allow use with the package (as an auxiliary component of the packaging)

For applications that include allowance of the use of supplemental shielding, coordinate with the General Information review to ensure that the engineering drawings include appropriate details for this shielding. Also, coordinate with the structural, materials, and thermal reviewers to ensure that the application demonstrates that shielding remains effective for the conditions for which the shielding evaluation credits this shielding. Additionally, coordinate with the reviewers of the Package Operations and Acceptance Tests and Maintenance Programs sections of the application to ensure that these sections adequately address the use of this shielding, as appropriate. Ensure, that, if found acceptable, the CoC specifically addresses the use of this supplemental shielding. These above requirements would not apply to any supplemental shielding not attached to the package, the sole purpose of which is to reduce external radiation levels to below regulatory requirements (e.g., additional shielding attached to the sides of the trailer or truck cab) (see NRC Information Notice 83-10).

Confirm with the thermal and materials reviewers that shielding materials will not exceed their allowable maximum temperature limits under normal conditions of transport and, as applicable, hypothetical accident conditions. Also confirm with these reviewers that shielding properties will not degrade during the service life of the packaging (e.g., degradation of hydrogenous materials). For evaluations that credit the neutron absorbers, coordinate with the criticality, materials, and acceptance tests and maintenance program reviewers to confirm the proper specifications of the absorber properties and allowable variations of those properties (see Sections 6.4.1.2, 6.4.3.2, 7.4.7, 9.4.1.6, and 9.4.2.4 of this SRP) and to confirm that the application includes appropriate qualification and acceptance testing of these absorbers.

Coordinate with the materials and acceptance tests and maintenance reviewers to ensure that the application contains appropriate and adequate acceptance tests and maintenance programs to ensure that the package shielding will be fabricated and maintained consistent with the package design and in a manner to meet the regulations (see Sections 7.4.6, 9.4.1.7, and 9.4.2.5 of this SRP). In general, appropriate acceptance tests include gamma scans and measurements of gamma and neutron radiation levels over the package surfaces where gamma and neutron-shielding materials are located in the design. Also, appropriate maintenance programs generally include periodic measurements of radiation levels. Ensure the acceptance criteria for acceptance tests and maintenance programs are consistent with and based on the packaging and contents descriptions in the application. For radiation-level scan or measurement acceptance tests and maintenance program tests, appropriate acceptance criteria would be measured radiation levels that do not exceed those that are calculated for the same radiation source(s) used in the test for package shielding at the minimum properties specified in the engineering drawings over the measured package surfaces. For acceptance tests, ensure that the entire package surface where the shielding is located is tested, whereas a reasonable number of appropriate locations on the package surface may be tested for maintenance program tests. RG 7.7, “Administrative Guide for Verifying Compliance with Packaging Requirements for Shipping and Receiving of Radioactive Material,” Section 2.1.1, “Elimination of Voids,” and NUREG/CR-3854, “Fabrication Criteria for Shipping Containers,” Section 3.2, “Acceptance Testing,” issued March 1985, provide additional guidance and information concerning acceptable shielding effectiveness test methods. Confirm that the acceptance tests also include appropriate chemical and physical tests of proprietary or nonstandard shield materials (e.g., polymer-based neutron shields). Note that for a package, or portions of the package, that rely only on carbon steel or stainless steel packaging components, which are generally fabricated to industry standard specifications, for shielding, visual inspections and dimensional inspections are generally sufficient acceptance tests for ensuring shielding performance. In other words, no additional acceptance tests would be needed for such a package or portions of the package.

Many materials have been used as gamma shielding in the different package types that the NRC has certified. These materials include steel, lead, tungsten, and depleted uranium. Depleted uranium rapidly and significantly oxidizes when exposed to heat and air, although the result may not be evident until some time after the conclusion of the 10 CFR 71.73 thermal test. Therefore, confirm with the structural and materials reviewers to ensure that the 10 CFR 71.71 and 10 CFR 71.73 impact tests, including puncture tests, and the other impact tests that may be appropriate for the package (e.g., 10 CFR 71.74 impact tests) do not damage the packaging cavity containing the depleted uranium in a manner that exposes the depleted uranium to the environment.

5.4.1.2 Summary tables of maximum external radiation levels

Confirm that the application describes the type of use or shipment for which the package is designed or evaluated (i.e., exclusive-use or nonexclusive use). Review the application's summary table listing of expected maximum radiation levels. As described below and in Section 5.4.4.4 of this SRP, ensure that the applicant calculated the maximum radiation levels for all relevant and appropriate surfaces. The summary table should include the maximum radiation levels for these package surfaces and the appropriate distances from these surfaces for the type of transport for which the package is designed and intended. The table should include total radiation levels as well as the separate gamma and neutron components of the radiation levels. For packages with multiple contents, the table should also identify the source or sources that produces the maximum radiation levels. For SNF packages, this includes specifying the burnup, enrichment (or uranium and plutonium composition for MOX SNF), and the cooling time combinations. As part of this review, examine variations in radiation levels at different package locations for general consistency (e.g., decreasing radiation levels with increasing distance or increasing shielding effectiveness), given shielding modeling assumptions and regulatory requirements and NRC guidance. Verify that the radiation levels are within the regulatory limits listed in 10 CFR 71.47 (see Table 5-2) and 10 CFR 71.51(a) for the appropriate conditions and types of shipment. Note that the accident conditions limit for shipments of plutonium by air is given in 10 CFR 71.64(a)(1)((ii), which is essentially the same as the limit for all other packages given in 10 CFR 71.51(a)(2).

Consult Figure 5-2 below in reviewing the application to identify, based on the package design and calculated radiation levels in the application, the surfaces and locations for which the application should provide radiation level results and the appropriate limits for those surfaces and locations. Figure 5-2 illustrates how the radiation level limits apply to different shipment configurations for both exclusive-use and nonexclusive-use shipments. Note that the current version of the DOT's Pipeline and Hazardous Materials Safety Administration's "Radioactive Material Regulations Review" document is another source of useful information regarding transportation requirements, including package radiation limits.

The application may include results for the package's transport index (TI). The value of the TI is the maximum radiation level at 1 meter [40 inches] from the package's surface in mrem/hr. For packages designed and evaluated for nonexclusive-use transportation, the application will include this value, and this value must not exceed the limit of 10 specified in 10 CFR 71.47(a). For exclusive-use shipments, 10 CFR 71.47 does not include a limit for the TI. While a TI is calculated in the application, the actual TI for a package is determined by measurement at the time of shipment. Ensure that the measured TI is placed on the package label. The TI is used in shipments to ensure proper controls are exercised for the shipments, including limiting the number of packages that may be shipped on a conveyance [see 49 CFR 173.441(c)–(e)].

Ensure that the package operating procedures assure the package will be used consistent with the shielding evaluation. This includes ensuring that measured radiation levels that exceed expected levels result in checks that the package has been properly loaded and prepared for transport. For example, for a package that is evaluated to meet the nonexclusive-use limits in

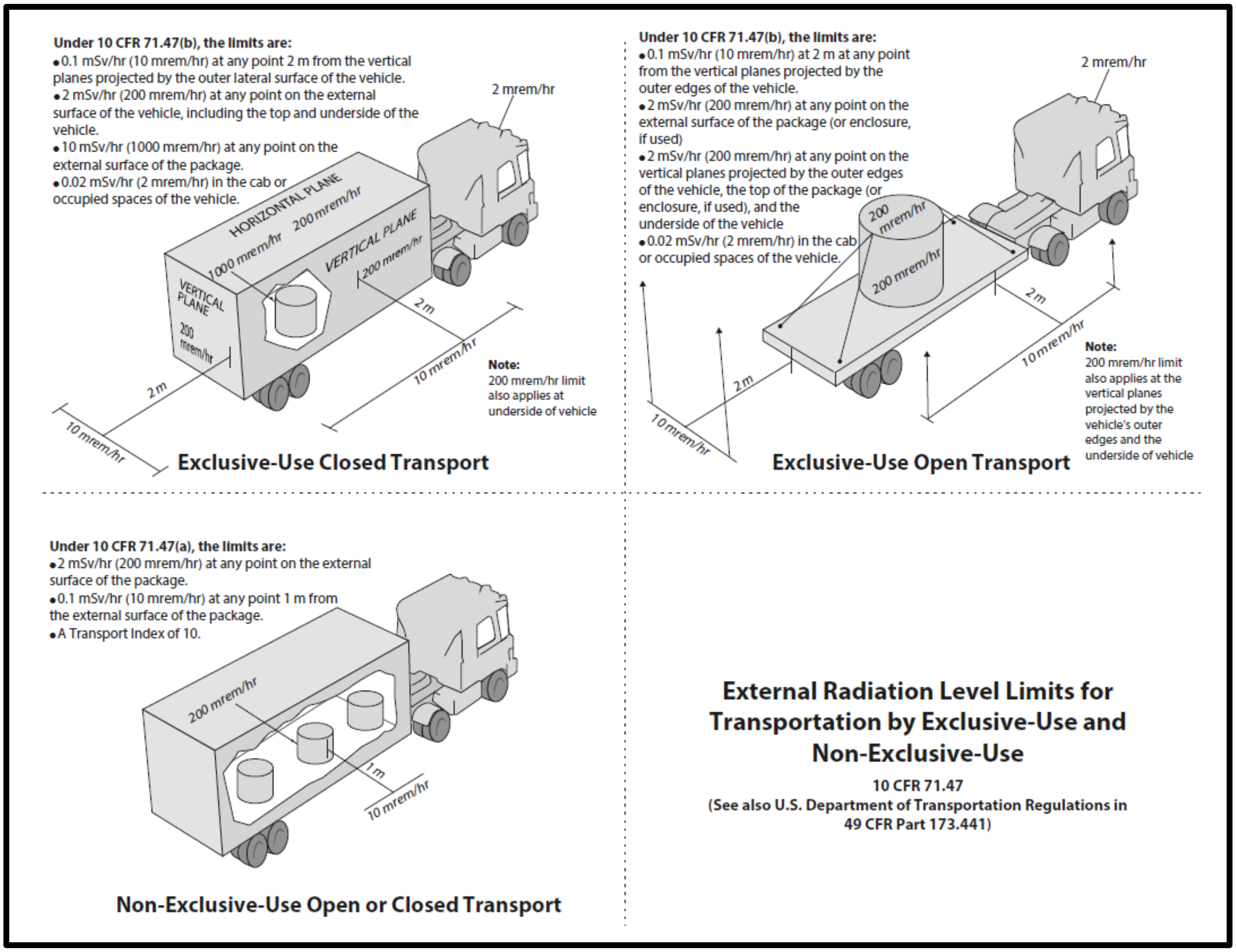

Figure 5-2 Illustration of surfaces to which regulatory radiation limits apply for exclusive-use and nonexclusive-use shipments

10 CFR 71.47, the measured radiation levels for a package prepared for shipment should, in general, not exceed the limits for nonexclusive use.

Confirm that the application states the contents and contents specifications that result in the maximum package radiation levels. For packages with a variety of contents or contents specifications with different source terms or spectra or different source configurations within the package, the same contents or contents specifications may not result in the maximum package radiation levels at all locations of the package surface or at the specified distances from the package surfaces. This may be true for the same package configuration and conditions (e.g., normal conditions of transport). This may also be true for different package configurations and conditions (e.g., normal conditions of transport versus hypothetical accident conditions). Therefore, ensure that the application states the contents and contents specifications that result in maximum radiation levels for each package surface (and at distance) for each package configuration and each set of conditions. For SNF packages, this includes such parameters as fuel type, maximum burnup, minimum enrichment, minimum cooling time, conditions of the SNF (e.g., damaged or undamaged), and the type of nonfuel hardware loaded with the fuel. For SNF, gamma and neutron radiation levels can significantly vary or be the greatest at different fuel specifications. Also, gamma radiation may be more dominant for some package surface locations or package configurations or set of conditions, and neutron radiation may be more dominant in other instances.

5.4.2 Radioactive Materials and Source Terms

Confirm that the contents used in the shielding analyses are consistent with those specified in the General Information section of the application. The contents description should be consistent with the package evaluation. Ensure that the specifications in the General Information section of the application are adequate to define the allowable contents in terms of the shielding evaluation (i.e., to ensure the shielding evaluation adequately bounds the allowable package contents). For applications with less-detailed or broader-scoped descriptions of the contents, the shielding analyses will need to address the variations in contents characteristics that the contents descriptions will allow in terms of properties relevant to shielding. The more detailed or limited in scope the contents description is, the more refined and focused the shielding evaluation can be. The level of detail may be dependent upon the package type as well. For example, a Type B waste package may have a broader description of the contents than a source package designed for multiple sources. If the package is designed for multiple types of contents or contents with a variety of specifications (e.g., SNF), ensure that the applicant clearly identified and evaluated the contents and contents specifications producing the highest external radiation levels at each location. Confirm that the identified contents and contents specification do indeed result in the highest, or bounding, radiation levels at each location.

Ensure that the contents descriptions in both the General Information and Shielding Evaluation sections of the application are sufficient to define the source terms of the allowable contents and the allowable configurations of the source terms, including possible configuration changes under normal conditions of transport and hypothetical accident conditions. Important specifications include the radionuclides present in the contents and their maximum quantities (e.g., maximum activity or maximum specific activity for radioactive material packages [i.e., source packages)], the contents' physical and chemical properties and form, and possible reconfiguration or distribution changes of nuclides and contents. For example, in describing how the radionuclides are distributed within the contents (including how such limits in the CoC conditions are to be interpreted), the applicant may characterize the distribution using terms such as "distributed throughout" and "essentially uniformly distributed," as those terms are defined in NUREG-1608, "Categorizing and Transporting Low Specific Activity Materials and Surface Contaminated Objects," Section 4.2.2. Verify that the applicant correctly identified and characterized all potential radiation sources, even if analysis shows they contribute negligibly to package radiation levels.

Note that a contents specification of simply a set number of A values (i.e., Type A quantities[2]) of radioactive materials or radionuclides is not sufficient for the reasons described in Regulatory Issue Summary (RIS) 2013-04, "Content Specification and Shielding Evaluations for Type B Transportation Packages," dated April 23, 2013. While there are different ways to specify the contents, whatever method is chosen to specify or define the allowable contents, the shielding evaluation should support this definition. The Package Operations section of the application may also need to include specific operations descriptions to ensure that the package user correctly loads the package in accordance with the contents specifications. RIS 2013-04 contains some examples of contents specifications and the associated shielding evaluations the staff has accepted along with the conditions for that acceptance.

For commercial SNF, ensure that the specifications include such things as the fuel types, fuel conditions (e.g., damaged, undamaged; see Section 7.4.14.1 of this SRP for guidance regarding

2 See the definition of a Type A quantity in 10 CFR 71.4, "Definitions."

fuel condition), assembly hardware specifications (material masses and cobalt impurity levels per axial zone), nonfuel hardware (NFH) specifications, maximum burnups, minimum enrichments (fissile uranium and plutonium specifications for MOX SNF), minimum cooling and decay times, and arrangements in the package. NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages," Section 3.3.1, "Active Spent Fuel Region Isotopics," and Appendix B, "Nuclide Importance and Parameter Sensitivity Study for PWR/BWR Source Term Generation," issued May 2003, include information about various commercial SNF parameters and their effects on the source term. Also, while written for SNF storage casks, NUREG/CR-6716, "Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks," issued March 2001, contains information that can be useful for reviewing the commercial SNF contents specifications for a transportation package.

If the contents include high-burnup commercial SNF [i.e., SNF with burnups in excess of 45,000 megawatt-days per metric ton uranium (MWd/MTU)], ensure the contents specifications include how the high-burnup fuel is to be treated, whether as damaged fuel or undamaged fuel, or in some other manner. Coordinate with the materials evaluation reviewer to ensure that the application supports the basis for the applicant's treatment of high-burnup fuel.

For a commercial SNF package, also ensure that the specifications for any NFH contents include the hardware types, component materials and masses per axial zone, quantities, arrangements in the package, maximum burnups, minimum cooling times, neutron flux factors, cobalt impurity levels and other activated materials (e.g., hafnium, silver-indium-cadmium), neutron source types, and strengths. Ensure that the application addresses specifications for those NFH types that may have multiple configurations (e.g., thimble plug devices that may also have water displacement or absorber rods).

For commercial SNF enrichments and burnups, it is acceptable for the values to be assembly-average minimum and assembly-average maximum, respectively, though calculation of the assembly average may require additional consideration for fuel with axial blankets. Natural uranium blankets effectively increase the burnup in the middle of the assembly's active fuel zone, with greater effect as the length of the blankets increases. Variations in fuel assembly type play a secondary role for pressurized-water reactor (PWR) fuel. For boiling-water reactor (BWR) fuel, part-length rods, void fractions, and channel sizes may also affect the strengths of neutron and gamma sources. Ensure that the contents specifications and source-term calculations for SNF that include MOX or thoria properly account for unique aspects of these fuel materials. These aspects include contributions from nuclides produced from fuel irradiation and from natural decay of fuel materials and buildup of nuclides with significant radiations at longer cooling times for fuel with short decay times (e.g., Tl-208 in thoria-bearing fuel).

For commercial SNF packages, also ensure that the application contains specific information concerning reactor operations that affect the SNF source term. Several NRC technical reports (specifically, NUREG/CR-6716, but also NUREG/CR-6700, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," issued January 2001; NUREG/CR-6701, "Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel," issued January 2001; and NUREG/CR-6798, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor," issued January 2003) discuss the potential effects of other parameters not typically included in the CoC conditions for commercial SNF package contents limits (e.g., moderator soluble boron concentrations, maximum poison loading, minimum moderator density (for BWR fuels), and maximum specific power). For example, the net impact of moderator density on package radiation levels is expected to be low for PWR fuels. However, be aware that the axial variation in moderator density in BWR cores can have a measurable effect on the axial variation of radiation levels for a BWR SNF assembly. The radiation levels may increase near the top of the assemblies where the moderator density was the lowest. This is particularly important for neutron sources because reduced moderator density will harden neutron spectrum and hence induce more actinide production.

For setting commercial SNF contents limits in the CoC, ensure the application uses proper parameters and specifications that are readily inspectable and with which a package user can easily determine compliance. Several of the parameters described above fit this purpose (e.g., minimum enrichment, maximum burnup, minimum decay time, maximum uranium mass). However, specific gamma and neutron source terms do not and so should not be used in the CoC to describe the allowable SNF contents.

For research SNF packages, ensure that the application adequately describes these SNF contents. Some items for commercial SNF also apply to research SNF. These specifications include maximum burnup, minimum enrichments (or fissile material specifications), assembly hardware, fuel condition, and appropriate assembly physical parameters (e.g., plate-type fuel, dimensions). The CoC description of the contents should include those parameters important for defining the source terms for the research SNF.

5.4.2.1 Source-term calculation methods

Ensure that the applicant has accurately determined the source terms associated with the proposed package contents and has used appropriate methods for the determination. This may involve the use of published data sources, which may be useful for contents of source packages with limited numbers of radionuclides present in the package, or the use of computer codes. The International Commission on Radiological Protection (ICRP) Publication 38, “Radionuclide Transformations—Energy and Intensity of Emissions,” is an example of such data source, though more recent data sources (e.g., ICRP Publication 107, “Nuclear Decay Data for Dosimetric Calculations”) are available. Depending upon the shielding code used to calculate the package radiation levels, the code may have source information built in already. This is the case for the MicroShield® code,[3] which allows selection of the radionuclides present in the source and the capability to specify the quantity (in curies or becquerels).

The SCALE code system’s ORIGEN-ARP module also has the capability to calculate the source terms from commercial SNF contents as well as specific radionuclides and other source materials (e.g., (α,n) neutron sources). The code can provide results in a variety of forms, including an energy spectrum with total source strength. For commercial SNF calculations, ORIGEN-ARP provides more of a rough estimate for source terms since it interpolates on libraries generated for specific assembly types with set characteristics for the ranges of enrichment, burnup, and decay time values used to generate those libraries. Other modules and sequences in the SCALE code system have been developed to calculate SNF source terms, including for research SNF, and provide more flexibility and user control over the assembly parameters for calculating them. These include ORIGEN-S, SAS2H and, in more recent versions of SCALE, TRITON.

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[3] The MicroShield code was developed by Grove Software, 4925 Boonsboro Road #257, Lynchburg, Virginia, 24503, http://www.radiationsoftware.com.

For applications that use published data sources, ensure that the data source has a strong pedigree; that is, the source is published by a well-known and trusted entity and the data have been properly validated and are publicly available. Ensure that the applicant has used the correct source term data from the published source in the shielding analyses. Also, confirm that the applicant has included the data for radionuclides that may also be present that are decay products of the proposed contents and that the contents description addresses the decay products. In various cases, the decay products may have significant impacts on and even be the dominant contributor to the package radiation levels. In such cases, ensure that the applicant has addressed and correctly determined the source term for an appropriate decay time that will maximize the radiation levels from the parent radionuclide and daughter radionuclides. The capability for this determination may also be included in the shielding code as well, as is the case for some versions of MicroShield.

For applications that use computer codes to determine source terms, verify that the applicant used a computer code, such as ORIGEN-S, that is well benchmarked and recognized and widely used by industry. If a vendor proprietary code is used, check the code validation and verification records and procedures, preferably with sample testing problems. Although easy to use, use of ORIGEN-2 (including ORIGEN-2.1) and the U.S. Department of Energy, Office of Civilian Radioactive Waste Management (OCRWM) Characteristics Database (TRW 1992) should be discouraged. Both have energy group structure limitations. For example, for ORIGEN-2, many libraries are not appropriate for burnups exceeding 33,000 MWd/MTU. Also, ORIGEN-2 and the OCRWM database are no longer maintained by the original developer and are based on outdated data that may contain errors. If the applicant uses a computer code that is designed for reactor analyses (e.g., CASMO) for source-term calculation, ensure that the code has been used in such a way that the calculations yield appropriate results to use as source terms in the shielding analysis. This includes appropriate consideration of unique aspects of any proposed SNF contents that include MOX or thoria.

Ensure that the applicant has provided appropriate descriptive information, including validation and verification status, and reference documentation. Determine whether the computer code is suitable for determining the source terms and if it has been correctly used. Pay particular attention to “Area of Applicability” to verify whether the application falls into the parameter ranges for which the code is validated. Determine whether the computer code is appropriately applied and that for SNF packages, the application includes verification that the chosen cross-section library is appropriate for the fuel specifications being considered. For example, many libraries are not appropriate for a commercial SNF burnup exceeding 45,000 MWd/MTU because validation data are limited at high burnups. If the applicant has used the code outside its validated parameter ranges, ensure that the applicant has adequately justified the acceptability of such use, including addressing uncertainties in the analysis results that result from this use.

Verify that the applicant has adequately addressed calculational error and uncertainties of the computer codes used to determine the radiological and thermal source terms for the shielding analyses for SNF packages (and for other packages, if appropriate). As part of this determination, consider factors such as other conservative assumptions and design margins in the analysis and maximum assembly heat loads for the design basis combination (or combinations) of fuel, burnup, enrichment, and cooling time. For example, adjustments to source-term values or calculation bases or other aspects of the shielding analysis or reduced decay heat or other parameter limits (versus low burnup fuel) may be necessary to compensate for uncertainties in the source-term calculations for commercial fuel with high burnups. An acceptable approach to address calculation errors and uncertainties is to establish a bounding value (or values) with justified conservatisms.

When reviewing the commercial SNF source-term calculations, also consider that nuclide importance changes in high-burnup fuels as a function of burnup and cooling time. The data for benchmarking the calculations and computer codes are limited at high burnups. Several NRC-sponsored studies (e.g., ORNL/TM-13315, "Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel" issued September 1998; ORNL/TM-13317, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel" issued September 1996; NUREG/CR-6700; NUREG/CR-6701; and NUREG/CR-6798) provide additional information on high-burnup source-term issues.

Ensure that the application describes the source terms in a format that is compatible with the shielding calculation input, including energy spectrum structure where applicable. For some packages or some package contents and for some shielding codes, the nuclide and its activity may be sufficient. In other cases, this may require specification of radiation type, energy spectrum, and total emission rate in particles per second per some unit basis (e.g., neutron/sec per assembly for SNF). Also, ensure that the application addresses any secondary radiations produced by reactions within the package contents or the package components. This includes gammas produced by (n,γ) reactions or neutrons produced by subcritical multiplication or (α,n) reactions. For package contents with significant β emitters, particularly when the package can be used to ship such contents without significant γ-emitting nuclides present in the contents in significant quantities, this also includes bremsstrahlung. When bremsstrahlung should be accounted for, ensure the applicant has used an appropriate method for estimating the source. One such method is included in "Introduction to Health Physics" (Cember 1996).

Coordinate with the thermal reviewer to determine the need to evaluate the applicant's calculation of the package contents' decay heat. Often, the same codes used to determine radiation source terms can also be used to calculate decay heat. Other methods are also available for determining decay heat for SNF. RG 3.54, Revision 2, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," describes a few such methods. Verify that the application adequately describes the calculation method and that the method is appropriate for and correctly used to determine the decay heat for the package contents. Ensure that the analysis also appropriately identifies and accounts for uncertainties in the decay heat analysis, as appropriate.

Perform independent calculations to confirm the applicant's calculated radiation source terms and decay heat levels, as appropriate. Perform independent calculations, as needed, to confirm that the applicant has properly determined the bounding source terms for the package contents. Support the containment review, as needed, by verifying the quantities of certain nuclides (e.g., krypton-85, tritium, and iodine-129) the applicant used to analyze releases of radioactive material during normal conditions of transport and hypothetical accident conditions. Confer with the containment reviewer to determine the need to verify these nuclide quantities.

5.4.2.2 Gamma sources

Based on the specified package contents, verify that the applicant calculated the maximum gamma source strength and spectra by an appropriate method (e.g., standard computer codes and hand calculations) for all appropriate contents. This includes all source terms that result in maximum radiation levels at different package surface and distance locations and for the different types of package contents for packages with multiple types of contents (e.g., SNF, NFH, greater-than-Class-C waste). Ensure that the application includes source-term contributions from radioactive decay products if they result in higher radiation levels than the contents without decay, as described in Section 5.4.2.1 of this SRP chapter. In evaluating the contents' source terms, note that for MOX SNF, the gamma source can be significantly larger than for LEU SNF (see Appendix C to this SRP).

For gamma source terms that are calculated with computer codes, review the key parameters described in the application or listed in the input file. When neutron sources are present, verify that the production of secondary gamma (e.g., from (n,γ) reactions in shielding material) is either calculated as part of the shielding evaluation (see Section 5.4.4 below) or otherwise appropriately included in the source term. Confirm that the results of the source-term calculations are presented as a listing of gamma fluences or fluence rates, for example, gamma or million electric volts (MeV) per second, as a function of energy. The energy group structure of the source term (or terms) should be consistent with the group structure input requirements of the shielding analysis code. If the energy group structure from the source-term calculation differs from that of the cross-section set of the shielding calculation, the applicant may need to regroup the photons. Regrouping can be accomplished by using the nuclide activities from the source-term calculation as input to a simple decay computer code with a variable group structure. Some applicants will convert from one structure to another using simple interpolation. In general, only gammas with energies from approximately 0.4 to 3.0 MeV will contribute significantly to the radiation levels for typical types of package shielding; thus, regrouping outside this range is usually of lesser importance. However, look for cases when other gamma energies may also be significant to package radiation levels and ensure these gammas are also appropriately handled.

Ensure that the application provides activity (or mass) and total inventory of radionuclides that contribute significantly to the source term as supporting information. Also, determine whether the source terms are specified in terms of total package contents or other appropriate contents quantities (e.g., for SNF, in terms of per assembly, per total assemblies, or per MTU). Ensure that the application correctly uses the source-term information (e.g., the total source strength and spectra).

For SNF packages, be aware that determining the source terms for fuel-assembly hardware and NFH is generally not as straightforward as for the SNF. The source term is primarily from the cobalt contained in the hardware, particularly in the steel and Inconel components, though other activation products should be considered as well, as appropriate. For some NFH, activation of other components, such as hafnium in hafnium-absorber assemblies and the silver-indium-cadmium material in some control-rod assemblies, can also produce a significant gamma source. The strength and physical distribution of the hardware source term depends upon factors such as the mass of the materials, the level of cobalt impurity in the steel and Inconel components, and the axial region of the fuel assembly (i.e., top nozzle or upper end-fitting, upper plenum, fuel, lower plenum, bottom nozzle or lower end-fitting) and the associated neutron flux in which the materials are irradiated. Thus, verify that the application identifies the materials that comprise the assembly hardware and NFH to be stored with the assemblies.

Verify that the application describes the masses of the materials that are located within each assembly axial zone. Ensure that the application includes the masses of the assembly components for steel-clad assemblies or assemblies with steel guide and instrument tubes. For NFH, such as control-rod assemblies, ensure that the application describes the basis for the masses of the components listed for each axial region. The activation of these items is dependent upon the operation practices of the different reactors. Many may be operated with these items positioned just above the fuel region or slightly inserted into the fuel region. Thus, only the lower ends of these items are irradiated, and the activation will be based on the appropriate flux factors for the axial regions in which the items were located. Ensure that the masses listed in each axial region are consistent with the extent of insertion into the assembly described in the application, which should be consistent with or reasonably bounding for operations practices for those items.

Ensure that the application identifies the cobalt impurity level used in the source-term calculation and describes the basis for that assumption. Various analyses have used impurity levels of about 800 to 1,000 parts per million (ppm), which is bounding for steel components of assemblies and NFH manufactured since the late 1980s. Data contained in PNL-6906, “Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal,” issued June 1989, show that, for at least some assembly types fabricated before that time, cobalt levels may be as high as 1,500 ppm in Inconel and 2,100 ppm in steel. Thus, ensure that the application analysis uses cobalt impurity levels that are appropriate for the fuel assemblies and NFH to be transported in the package, given the age of the assemblies and NFH (based on their burnups and cooling times). If a lower cobalt impurity is assumed, ensure that appropriate references are provided.

The nature of the flux changes in magnitude and spectrum in regions outside of the fuel region. Thus, ensure that the application analysis adequately accounts for the impact of these changes on hardware irradiation in these other axial regions. This may be done by the use of scaling factors such as those described in NUREG/CR-6802, Section 3.3.2, “Hardware Regional Activation.” Additionally, ensure that the hardware source term includes the contributions of materials such as hafnium and silver-indium-cadmium for those NFH items that include these materials. While the application may describe the source from cobalt in terms of curies, the source terms for these other materials likely will be described in terms of their energy spectrum.

The impacts on radiation levels from the activated assembly hardware and NFH can be significant. The effort devoted to reviewing this analysis should be based on the contribution of these source terms to the radiation levels presented in the shielding evaluation. Ensure that the source-term analysis addresses all appropriate NFH items that are included in the proposed package SNF contents, comparing the items identified in the source-term analysis with those items listed in the contents descriptions in the General Information and Shielding Evaluation sections of the application.

5.4.2.3 Neutron sources

Evaluate the method used to determine all neutron-source terms described in the application. Verify that the method considers, as appropriate, neutrons from spontaneous fissions and from (α,n) reactions. Verify that the contribution from both of these sources are separately identified, along with the actinides or light nuclei significant for these processes, as appropriate for the package contents. If the application assumes that either source-term contributions is negligible, confirm that the applicant provided an appropriate justification for their omissions. Verify that the production of neutrons from subcritical multiplication is either calculated as part of the shielding evaluation (see Section 5.4.4 below) or otherwise appropriately included and described in the basis of the source terms.

Confirm that the results of the source-term calculations are presented as a listing (or listings) of total neutron strengths and spectra (i.e., neutrons per second as a function of energy) for all appropriate contents. This includes all source terms that result in maximum radiation levels at different package surface and distance locations and for the different types of package contents for packages with multiple types of contents (e.g., SNF, neutron-source assemblies (NSAs), other neutron-emitting radioactive materials). Also, determine whether the application specifies the source terms in terms of total package contents or other appropriate contents quantities (e.g., for SNF, in terms of per assembly, per total assemblies, or per MTU). Ensure that the source-term information (e.g., the total source strength and spectra) is correctly used in the Shielding Evaluation section of the application. The energy group structure of the source term(s) should be consistent with the group structure input requirements of the shielding analysis code.

For SNF packages, the SNF neutron source will generally result from both spontaneous fission and alpha-n reactions in the fuel. Depending on the method used to calculate these source terms, the applicant may need to define the energy group structure separately. This is often accomplished by selecting the nuclide with the largest contribution to spontaneous fission (e.g., curium-244) and using that spectrum for all neutrons, since the contribution from alpha-neutron reactions is generally small. For SNF with cooling times less than 5 years, confirm that the analysis addresses the spectra of curium-242 and californium-252.

The specification of a minimum initial enrichment is a necessary basis for defining the allowed SNF contents. Verify that the assumed minimum enrichments bound all assemblies the applicant proposes for transport in the package. Lower-enriched fuel, irradiated to the same burnup as higher-enriched fuel, produces a higher neutron source. Therefore, verify that the application specifies the minimum initial enrichment, and ensure the CoC contents limits include appropriate minimum enrichment limits.

Ensure that the applicant adequately described the neutron source, both source strength and spectrum, for NSAs included in the NFH to be transported with the SNF assemblies. NSAs are divided into two main categories: primary and secondary sources. Primary sources include polonium-beryllium (PoBe), americium-beryllium (AmBe), and other sources that generate neutrons though (α,n) reactions or spontaneous fission. Some of these sources have significantly long half-lives and can contribute a neutron source equivalent to the source of a SNF assembly. It is these sources that can contribute significantly to the neutron-source term in the package and so should be included in the shielding evaluation. Secondary sources include antimony-beryllium (SbBe) and other sources that generate neutrons through γ-n reactions. These sources typically have very short half-lives and need to be "charged" through neutron activation of the heavier element in the source material. Thus, secondary neutron sources usually contribute negligibly to the neutron-source term in the SNF package.

With regard to the contributions to the neutron source from subcritical multiplication in SNF packages, note that the results of depletion codes like SCALE's TRITON and SAS2H or CASMO do not include this contribution. This source can often be addressed through the use of proper options in the input to the shielding code or use of appropriate factors by which the neutron source is increased when input into the shielding code. Ensure that the applicant justified the appropriateness of the selected method, including the input options and parameters in the shielding code (e.g., conservative assumptions of fissile content) or the factor (or factors) used to increase the source.

In reviewing the neutron-source specifications for MOX SNF, consider the information in Appendix C to this SRP, which indicates the neutron source may be more important relative to the gamma source for MOX SNF, with neutron emission rates significantly larger than for LEU SNF. Additionally, the (α,n) contribution is more significant and may dominate the spontaneous fission contribution to the neutron source. Therefore, the determination of the neutron-source term and the source energy group structure should account for the contributions from both of these neutron sources. In reviewing MOX SNF, consider and account for the differences in the neutron energies, spectral distributions, and emission rates versus LEU SNF to ensure the applicant has properly calculated and described the MOX SNF neutron-source terms.

5.4.3 Shielding Model and Model Specifications

Coordinate with the structural, thermal, and materials reviewers to determine the effects the evaluations for normal conditions of transport and the tests for hypothetical accident conditions have on the packaging and its contents. For example, the package might have impact limiters or an external neutron shield that could be damaged or destroyed during the structural and thermal tests of 10 CFR 71.73. Also, the package may have a personnel barrier. This barrier may be present for normal conditions of transport but is not designed to survive the hypothetical accident conditions. Verify that the models and modelling assumptions used in the shielding calculations are consistent with the effects for the respective conditions.

5.4.3.1 Configuration of source and shielding

Examine the sketches or figures and sample input files, if provided, in the application to evaluate the applicant's shielding models. Verify that the dimensions and materials properties of the contents, radioactive sources in the contents, and the packaging components used in the shielding models are consistent with those specified in the package drawings and contents descriptions presented in the General Information section of the application.

Verify that the dimensions and material properties of the packaging components used in the models are those that maximize the package radiation levels. For example, the dimensions should be at the conservative end of their tolerance range, or they should be set such that the package shielding is minimized in a realistic manner. If the latter option is chosen, ensure that the applicant has adequately justified that the selected model dimensions result in the minimum shielding performance of the package. Ensure that voids, streaming paths, and irregular geometries are included in the model or otherwise treated conservatively in the model. These items include such things as any gaps between lids and flanges and between lead shielding and surrounding steel components that can exist based on packaging component dimensions, including tolerances, and locations of changes in package dimensions and shielding properties such as locations beyond the axial or radial extent of neutron- or gamma-shield components. Also ensure that the models include the effects of the normal conditions of transport evaluations and the hypothetical accident conditions tests for analyses versus the appropriate radiation level limits for these conditions. These effects may include loss of neutron shielding, lead slump, loss of impact limiters, crushing or deformation of packaging components, and puncture of packaging components for hypothetical accident conditions and the release or unscrewing of internal container lids for normal conditions of transport.

Verify that the dimensions and other properties of the package contents and sources used in the models are those that maximize the package radiation levels. If the package contents can be positioned at varying locations, have varying densities or compositions, or have varying source distributions, ensure that the locations, properties, and source distributions of the contents used in the evaluation are those that result in maximum expected external radiation levels. For example, the contents and source configuration that maximizes radiation levels on the side of the package might not be the same configuration that maximizes the radiation level on the top or bottom. Ensure that the application includes any changes in contents and source configurations (e.g., displacement or redistribution of the contents and sources, movement of contents and sources out of inner containers for containers when the lids release or unscrew, compaction of contents and sources) resulting under normal conditions of transport and hypothetical accident conditions, as appropriate.

The requirements in 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) state that package effectiveness should not be substantially reduced and external radiation levels should not be significantly increased for a package evaluated under the normal conditions of transport. In terms of the shielding evaluation, these requirements may be considered as met for shielding evaluations where the applicant includes the impacts of the normal conditions of transport evaluations in the models used to evaluate compliance with the 10 CFR 71.47 radiation-level limits. For exclusive-use shipments in which the analysis is based on the radiation levels stated in 10 CFR 71.47(b), confirm that the application includes the dimensions of the transport vehicle and the package location on the vehicle, as appropriate.

For commercial SNF packages, the verification of the package contents and sources described above includes verifying that the application properly models the contents, source-term locations, and the structural support regions of the fuel assemblies. Generally, the SNF contents model should include at least three source regions (the fuel region and top and bottom assembly hardware regions). Within the SNF region, the fuel materials may generally be assumed to be homogeneous in facilitating shielding calculations. In some cases, the presence of basket material may be homogenized as well. In either case, determine whether homogenization is not appropriate or improperly modeled, such as when it distorts the neutron multiplication rate or when radiation streaming can occur between basket components.

Because of uneven burnup profiles, a uniform source distribution is generally conservative for the top and bottom radiation level points. However, this may not be appropriate for the axial center unless the neutron and gamma source strengths are appropriately adjusted. Typically, fuel gamma source terms vary proportionally with axial burnup, and fuel neutron-source terms vary exponentially by a power of 4.12 with burnup (NUREG/CR-6802). These effects can be applied to the axial variation in burnup. If axial peaking appears to be significant, verify that the applicant's analysis has appropriately treated this phenomenon, including the effects on the gamma- and neutron-source terms. Ensure that the assembly structural support regions (e.g., top and bottom end hardware and plenum regions) are correctly positioned relative to the SNF. These regions may be individually homogenized.

If the proposed commercial SNF contents include damaged fuel, ensure that the contents models appropriately represent the possible configurations of the damaged fuel that maximize package radiation levels. Because damaged fuel may not retain the structural configuration of an assembly or may also be defined to include fuel debris, the models should include compaction of the damaged fuel contents and the associated source terms, identifying the amount of compaction of the source that maximizes package radiation levels. While compaction concentrates the source terms from the fuel, it also results in denser material, which in turn results in increased self-shielding by the contents. Thus, the bounding degree of compaction may not be the full amount of compaction that is physically possible. Also, ensure the models include fuel material in assembly regions that for undamaged assemblies normally only contain assembly hardware material since, with damaged fuel and fuel debris, fuel material can move into these areas. Additionally, ensure that the models include movement of the damaged fuel contents consistent with what the package would allow (e.g., within a damaged fuel can, if used) to maximize the package radiation levels for the different package surface locations. For example, the models should place the compacted source and contents (i) as close as possible to the base of the package to maximize radiation levels at the package base; (ii) at the package side surfaces below the axial extent of any gamma or neutron shielding on the side of the package; and (iii) as close as possible to the top of the package to maximize the radiation levels at the respective bottom, side, and top areas of the package, including areas where packaging shielding varies along those package surfaces.

For commercial SNF packages that include high-burnup fuel contents (i.e., SNF with burnup exceeding 45 GWd/MTU), work with the materials, structural, and thermal reviewers to understand the approach taken for addressing these contents and to understand the implications for the fuel's behavior under normal conditions of transport and hypothetical accident conditions. Based on this coordination, identify and ensure the applicant's models address the impacts of these conditions on the high-burnup fuel's configuration. The shielding analysis should address credible and bounding reconfigurations of the fuel. Depending upon the applicant's approach and the outcomes of the materials, structural, and thermal reviews, analysis with fuel reconfiguration may be necessary to support the certification basis (also referred to as the licensing basis) for the package or may be needed as a defense-in-depth measure. Ensure that the application and the results of the review clearly indicate the purpose of the reconfiguration analysis (either as part of the certification basis or as defense-in-depth). Since the staff's understanding and knowledge regarding the behavior of high-burnup fuel continues to evolve, work with the other reviewers, particularly the materials reviewer, to understand the latest guidance that applies to evaluations of high-burnup fuel.

For research SNF, apply the preceding guidance as applicable and appropriate to ensure the applicant's analyses adequately consider the possible configurations of the research SNF and its associated source terms within the package.

5.4.3.2 Material properties

Verify that the applicant described and used appropriate material properties (e.g., composition, mass densities, and atom densities) in the shielding models for all packaging components, package contents, and the conveyance (if applicable). For nonstandard materials or other uncommon materials such as polymer-based neutron shields, foams, plastics, and other hydrocarbons, ensure that the applicant provided relevant references documenting the materials' properties. Ensure that the shielding model uses the material properties that minimize the shielding effectiveness of these materials (e.g., minimum density, minimum hydrogen content, minimum boron-10 content).

Most computer programs used for shielding calculations allow the analyst to specify either mass densities in grams-per-cubic-centimeter or atom densities in atoms-per-barn-centimeter. Consider whether either mass density or atom densities alone is sufficient for certain types of materials. Note that the use of atom densities can be subject to errors. Therefore, if used, confirm that the applicant calculated correct atom densities and correctly input these densities into the analysis models.

Work with the materials and the acceptance tests and maintenance program reviewers to ensure that the composition and fabrication of the nonstandard and uncommon materials are properly controlled in achieving the specified properties that are relied on for shielding (e.g., compositions, densities, dimensional properties). Such controls may also be needed for shielding materials such as poured lead shields. This also includes appropriate controls and tests for neutron absorbers that are also relied upon in the shielding evaluation. In this context, verify that specific information on control measures and appropriate shielding effectiveness tests is included in the Acceptance Tests and Maintenance Program section of the application (see Sections 5.4.1.1, 7.4.6, and 9.4.1.7 of this SRP). For cases where neutron absorbers are credited, also work with the criticality reviewer to ensure that the application includes appropriate qualification and testing of the absorbers. Also work with these reviewers to assess if any shielding properties could degrade during the service life of the packaging and to confirm that adequate controls and tests are in place to ensure the long-term effectiveness of such shielding materials (see Sections 7.4.6 and 9.4.2 of this SRP).

Work with the materials and thermal reviewers to ensure that the application describes the effects of temperature and radiation on packaging materials. Work with the materials and thermal reviewers to understand the effects of the normal conditions of transport evaluations and hypothetical accident conditions tests on the properties of the package components and contents material, including changes in composition and density. For example, elevated temperatures may reduce hydrogen content through loss of bound or free water in hydrogenous shielding materials or degradation of polymer materials. Ensure that materials properties in the shielding models appropriately or conservatively include these effects (i.e., the effects of temperature, radiation, and the different conditions' evaluations and tests). Certain effects are not acceptable. For example, temperature-sensitive materials credited in the shielding evaluation should not be subject to temperatures at or above their design limitations during normal or accident conditions. Melting of lead shielding is also not acceptable. Also, these materials' properties should not degrade during the package's service life (e.g., degradation of foam, dehydration of hydrogenous materials, cracking of the neutron shield).

Typically, nonstandard or uncommon materials such as polymer-based neutron shields are neglected in the models for hypothetical accident conditions. This is because of the effects of tests such as the puncture test and thermal test. However, if the applicant's analysis takes some credit for these materials in these models, ensure the credit bounds or is conservative for the impacts of the tests for these conditions. This includes ensuring that the applicant has provided information that describes the impacts of the tests for these conditions on the materials' properties and working with the materials and thermal reviewers to confirm the validity and applicability of the information to describe the materials' properties under these conditions.

If the shielding model considers a homogenous source region rather than a detailed heterogeneous model of the contents (e.g., homogeneous fuel region for SNF versus explicit model of fuel rods with pellets and cladding), confirm that such an approach is justified, and verify that the homogenized mass densities are correct for normal conditions of transport and hypothetical accident conditions. Because an accurate, effective density of homogenized source terms is important in characterizing self-shielding, perform a confirmatory calculation of this homogenized density.

5.4.4 Shielding Evaluation

5.4.4.1 Methods

Ensure that the methods used for the shielding evaluation are appropriate for evaluating the radiation levels of the package. The methods should be adequate to effectively represent and evaluate the material properties, geometries and configurations of the packaging components and package contents, and the contents' radiation source-term properties (e.g., radiation types, energies, spectra, and secondary sources such as from (n,γ) reactions in the packaging materials). Verify that the methods are also adequate to effectively represent and evaluate the effects of the normal conditions of transport evaluations and the hypothetical accident conditions tests. Generally, more complex methods are necessary to adequately evaluate packages with more complex component and contents geometries and materials properties and more complex sources. However, simpler methods may also be acceptable for a complex package if the applicant used the methods in a manner that is bounding for the package.

Evaluation methods may not always involve computer codes. Depending upon the package, simple hand calculations may be sufficient. Additionally, in lieu of an analytical calculation, the package evaluation may involve radiation measurements on a prototype package, with a description of the measurement method and the results provided in the application's Shielding Evaluation section of the application. RIS 2013-04 also includes information that may be useful to consider in evaluating the applicant's shielding evaluation method.

If the applicant chooses to evaluate the package using radiation measurements, ensure the application includes an adequate description of the measurement methods and provides adequate details of the results to demonstrate compliance with the limits in 10 CFR 71.47 and 10 CFR 71.51(a). Verify that the information in the application is sufficient to demonstrate that the applicant has used measurement equipment and techniques that are appropriate for the types of radiation and the radiation energy and spectrum of the package contents and that the equipment produces reliable results (e.g., the detector calibration is valid). Depending on the technique and equipment and the strength of the source used in the measurements, correction factors may also be necessary to adjust the results of the measurements to ensure they demonstrate compliance with the limits for the proposed contents limits. These correction factors may include geometric adjustments to ensure that the result is for the package surface as well as scaling factors for use of sources with source strengths that are less than the proposed package limits. The International Atomic Energy Agency's Safety Guide TS-G-1.1, "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material," paragraph 233.5 and Table 1, and NUREG/CR-5569, "Health Physics Positions Data Base," HPPOS-013, "Averaging of Radiation Levels Over the Detector Probe Area," issued February 1994, contain useful information regarding detector size and measurement correction factors and averaging of radiation levels over the detector probe area. Ensure that the applicant's evaluation includes measurements for comparison against the regulatory limits that are for prototype packages that are in the as-fabricated condition and for prototype packages that have been evaluated and tested for the appropriate conditions (i.e., normal conditions of transport and hypothetical accident conditions). Ensure that the description of the analysis demonstrates that the measurement results are the maximum radiation levels at any point on the package surface and at the regulatory distances from the package.

If the applicant evaluated the package using hand calculations, ensure that the application includes adequate information to describe the calculation method and the results. The information should be adequate to demonstrate that the applicant correctly identified the locations and configurations of the package for which package radiation levels are maximized. Ensure the description also describes the data and the sources of the data used in the analysis, including source spectra, source emission rates, attenuation properties of the source materials and packaging components credited in the analysis, buildup factors for those materials, and production of secondary radiation in the packaging materials (if applicable). Confirm that the data come from validated sources and that the applicant has used appropriate data in the analysis. For analyses with multiple shielding materials, confirm that the applicant has appropriately or conservatively accounted for the buildup and attenuation of radiation through multiple materials.

A variety of computer codes are available that have been and may be used for shielding analyses. The codes may use Monte Carlo transport, deterministic transport, or point-kernel techniques for problem solutions. The point-kernel technique is generally appropriate only for gammas since transportation packagings typically do not contain sufficient hydrogenous material to apply removal cross sections for point-kernel neutron calculations. Shielding codes that have typically been used or may be used in package analyses include MicroShield, SCALE (e.g., SAS4, MONACO/MAVRIC), MCBEND, and MCNP. MicroShield is a one-dimensional point-kernel code that applicants have used for source packages and other similar packages. The remaining codes have been and can be used for more complex package designs, as well as simple package designs.

For a shielding analysis that uses computer programs or codes, ensure that the application identifies the codes used, including the versions, and provides a brief description of the code to justify that it is appropriate for analyzing the package radiation levels. For older code versions, additional justification may be necessary, particularly if the applicant's use of the older code version extends beyond the ranges of parameters for which that version of the code was validated or that version of the code the developer no longer supports. If the applicant used proprietary computer codes or those not well established (e.g., the codes are not widely used or recognized codes), ensure that the applicant has included a detailed description of the code, including the methods the code uses and the limitations and capabilities of the code.

Ensure that the applicant has demonstrated that the computer codes and versions used in the analysis are adequate for the analysis and valid for the particular computational platform used to perform the analysis through benchmarking and validation of the versions of the codes used. The applicant should provide appropriate references for the code as well as benchmark and validation data for the code. For a well-established code, such as MCNP and SCALE, applicant may instead specify widely available references or references that have been previously submitted to the NRC for the same code and code version. Otherwise, check that the application includes test problem solutions that demonstrate substantial similarity to solutions from other sources and benchmark that code's capability to perform calculations for the proposed package.

Verify that the applicant used a code appropriate for the package design. Packages with complex geometries and configurations, such as streaming paths and irregular or nonsymmetric geometries, generally require a code with a two-dimensional or three-dimensional calculation capability. One-dimensional codes provide little information about off-axis locations and streaming paths. Even for radiation levels at the end of the package, one-dimensional codes require a buckling correction that must be justified since merely using the packaging cavity diameter may underestimate actual radiation exposure rates (i.e., overestimate the radial leakage). Even a two-dimensional calculation may not be adequate for determining any streaming paths if the modeled configuration is not properly established.

Confirm that the code's cross-section library is applicable for shielding calculations. Confirm that a coupled cross-section set is used and that the code has been executed in a manner that accounts for secondary sources (e.g., subcritical multiplication, secondary gamma production), unless the evaluation has independently determined source terms for these secondary sources (e.g., in the source-term calculations described in Section 5.4.2 above). Confirm that radionuclide libraries, decay schemes, neutron and gamma yields, and spectra are valid and appropriate and are documented in the application, as applicable for the analysis method and computer code.

Additionally, particularly for commercial SNF packages, applicants often use transport or point-kernel methods to calculate neutron and gamma response functions [unit of (mrem/hr)/(source particle/s/cm^2)]. This technique, also known as the response function method, enables an applicant to quickly determine radiation levels for different source terms by multiplying the source terms by the response functions instead of running a separate transport calculation for each source term. It is based on the premise that, all else being equal (e.g., source particle type, energy, origin; detector location; material and geometric properties of the system), an increase in the source strength results in a corresponding increase in package radiation levels. For analyses that employ this response function technique, verify the following:

- The applicant calculated a response function for each particle type and for each energy bin in the particle type's energy spectrum.
- The response functions are used only for the shielding and source configuration (geometric and material properties) for which the response functions were calculated.
- The source properties (material and geometric) are appropriate or conservative for the contents for which the functions were calculated.
- The response functions are used only for the detector location for which the functions were calculated.
- The calculations for determining the response functions are well converged and appropriately account for any errors and uncertainties resulting from calculation or use of the response functions.

Thus, multiple sets of response functions may be needed to support the shielding analysis. This includes separate sets of response functions for differences in shielding properties (material or geometric), for differences in source properties (material or geometric), and for different detector locations. Ensure that the applicant has determined a sufficient number of sets of response functions to analyze and determine the maximum radiation levels at the package surfaces and the distances from the package specified in the regulations.

5.4.4.2 Code input and output data

Verify that the application identifies key input data for the shielding evaluations that use computer codes. The key input data will depend on the type of code (e.g., point-kernel, deterministic, or Monte Carlo) as well as the code itself. In addition to data describing the source terms and the materials and dimensions of the package contents and the packaging components identified above, key input data may also include data such as convergence criteria, mesh size, neutrons per generation, number of generations, and conversion factors to convert radiation fluence rates to radiation levels. Note that codes such as MicroShield may have input data limitations with regard to materials specifications and handling buildup across multiple materials. Thus, confirm that the applicant selected input parameters in a way that is conservative for these aspects of the package.

Ensure that the application includes a set of representative output files (or key sections of specific files, including input data) for each type of calculation performed in the shielding analyses. Ensure that proper convergence is achieved and that the calculated radiation levels from the output files agree with those reported in the text and tabulations and demonstrate compliance with 10 CFR Part 71 radiation limits.

For the other, noncomputer code evaluation methods, ensure that the application identifies the data and parameters for those methods and the results of those evaluations as discussed in the method description in Section 5.4.4.1 above.

5.4.4.3 *Fluence-rate-to-radiation-level conversion factors*

Ensure that the evaluation properly converts gamma and neutron fluence rates, as applicable to the package, to radiation levels. Verify the accuracy of the conversion factors, which should be tabulated as a function of the energy group structure used in shielding calculations. Ensure that the application includes supporting information and documentation for these tabulations.

While a variety of conversion factors are available for use in shielding analyses, the NRC only accepts the use of the American National Standards Institute/American Nuclear Society (ANSI/ANS) 6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," conversion factors. The basis for this acceptance is explained below. Thus, unless adequately justified, confirm that the applicant used these conversion factors in its analysis. The justification should include close correspondence with the accepted conversion factors and appropriateness for the application (e.g., conversion factors are based on the same methodology as is incorporated into the limit, or usefulness for demonstration of compliance by measurement).

The radiation level limits in 10 CFR Part 71 are in terms of dose equivalent and apply to the package surfaces and specific distances from those surfaces and not to doses to individuals. Furthermore, the package user demonstrates compliance with these limits at the time of shipment [to meet 10 CFR 71.87(j)] by measurement. The conversion factors in ANSI/ANS 6.1.1-1977 are appropriate because they convert the fluence rate to radiation levels that are in terms of dose equivalent.

Conversion factors, such as those in the 1991 version of ANSI/ANS 6.1.1 are based on significantly different models and result in radiation levels that are in terms of effective dose equivalent. This quantity (effective dose equivalent) and the model are based on impact to organs in the body, as can be seen in the definitions available for this quantity (e.g., see 10 CFR 20.1003, "Definitions"). In addition to being a different quantity than specified in the regulations, effective dose equivalent is not a measurable quantity and is specific to doses to individuals. Thus, use of conversion factors that yield results in terms of effective dose equivalent is not appropriate to demonstrate compliance with the 10 CFR Part 71 radiation limits.

Other problems arise with other conversion factors such as the ANSI/ANS 6.1.1-1991 standard's factors. While direct comparison is not appropriate because the quantities are different, the radiation levels calculated with conversion factors like those in ANSI/ANS 6.1.1-1991 underestimate radiation levels versus those calculated with factors such as those in ANSI/ANS 6.1.1-1977. This is a result of the shielding provided by other body tissues between the source and the target organs in the models that are the basis of the 1991 version factors. In addition, ICRP Publication 45 (1985) recommends that the quality factors for neutrons be scaled up uniformly by a factor of two, which counteracts the neutron dose rate reduction effected by the body shielding the target organs. However, nothing has been done to address the neutron quality factors; thus, use of the conversion factors from the 1991 version of the standard significantly under-predicts neutron radiation levels. While the 1991 version of the standard has been withdrawn (as well as the 1977 version), given the preceding considerations, the NRC accepts the use of the 1977 version of the standard.

Note that some versions of some codes, such as MicroShield, use conversion factors that are more like the ANSI/ANS 6.1.1-1991 standard factors and may not have an option for using the accepted factors. As described above, this will result in underestimates of package radiation levels. Verify that the application addresses this. One approach to address this is for the applicant to calculate appropriate adjustment factors and apply these factors to the radiation level results from the code. For codes that also show the fluence rates at the detector locations, another option is for the applicant to use the fluence-rate results and manually perform the conversion to radiation levels using the ANSI/ANS 6.1.1-1977 conversion factors.

5.4.4.4 External radiation levels

Confirm that the external radiation levels under normal conditions of transport and hypothetical accident conditions agree with the summary tables in the application and the discussion in Section 5.4.1.2 of this SRP chapter. Confirm that the radiation levels meet the limits of 10 CFR 71.47(a) or 10 CFR 71.47(b), as appropriate, and 10 CFR 71.51(a)(2). Verify that all radiation level point locations shown in the shielding analyses include all locations prescribed in 10 CFR 71.47(a) or 71.47(b) and in 71.51 (a)(2).

Verify that the analyses, whether calculations or measurements on a package prototype, demonstrate that the applicant has selected the locations of maximum expected package radiation levels. Note that maximum levels might not occur at the midpoint of a package surface or parallel plane. Radiation peaking often occurs near the axial or radial edges of package neutron- and gamma-shielding components and impact limiters and at or near locations of voids and other streaming paths and other irregular package component geometries. Therefore, ensure that the analyses in the application appropriately considered and evaluated these aspects of the package in identifying locations of maximum radiation levels. Ensure that the external radiation levels are reasonable and that their variations with locations over external surfaces of the package are consistent with the geometry and shielding characteristics of the package and the locations of the source terms of the contents that are used in the different calculations. Also, verify that the analyses appropriately consider the conservatism of simplifying assumptions and support assertions that nonconservative assumptions are more than compensated for by conservative assumptions.

In evaluating package surface radiation levels, ensure the applicant correctly identified the package surfaces and analyzed the radiation levels for the package surfaces and at the correct distances from the package surfaces. This is fairly straightforward for packages that have uniform, simple surfaces. In the case of packages with complex configurations or geometries, the package surface can vary significantly.

Figure 5-3 illustrates what constitutes the package surface and the appropriate radiation level limits for package surfaces for packages with nonuniform, complex surfaces. The images in the figure are a cutaway view (quarter symmetry) of the packages and only show the outer edge of the package surface (i.e., no detail is provided to distinguish different components such as neutron shielding, impact limiters, or the outer shell of the package). The top image in the figure is for an exclusive-use shipment that uses a personnel barrier that extends only between the impact limiters on the package. As can be seen in Figure 5-3, a package may have features that do not extend over the entire surface, so the surface location changes. Or, in the case of closely spaced fins, where the spacing makes it impractical to see or access the package’s true surface between the fins, the package surface for radiation limit compliance purposes may be

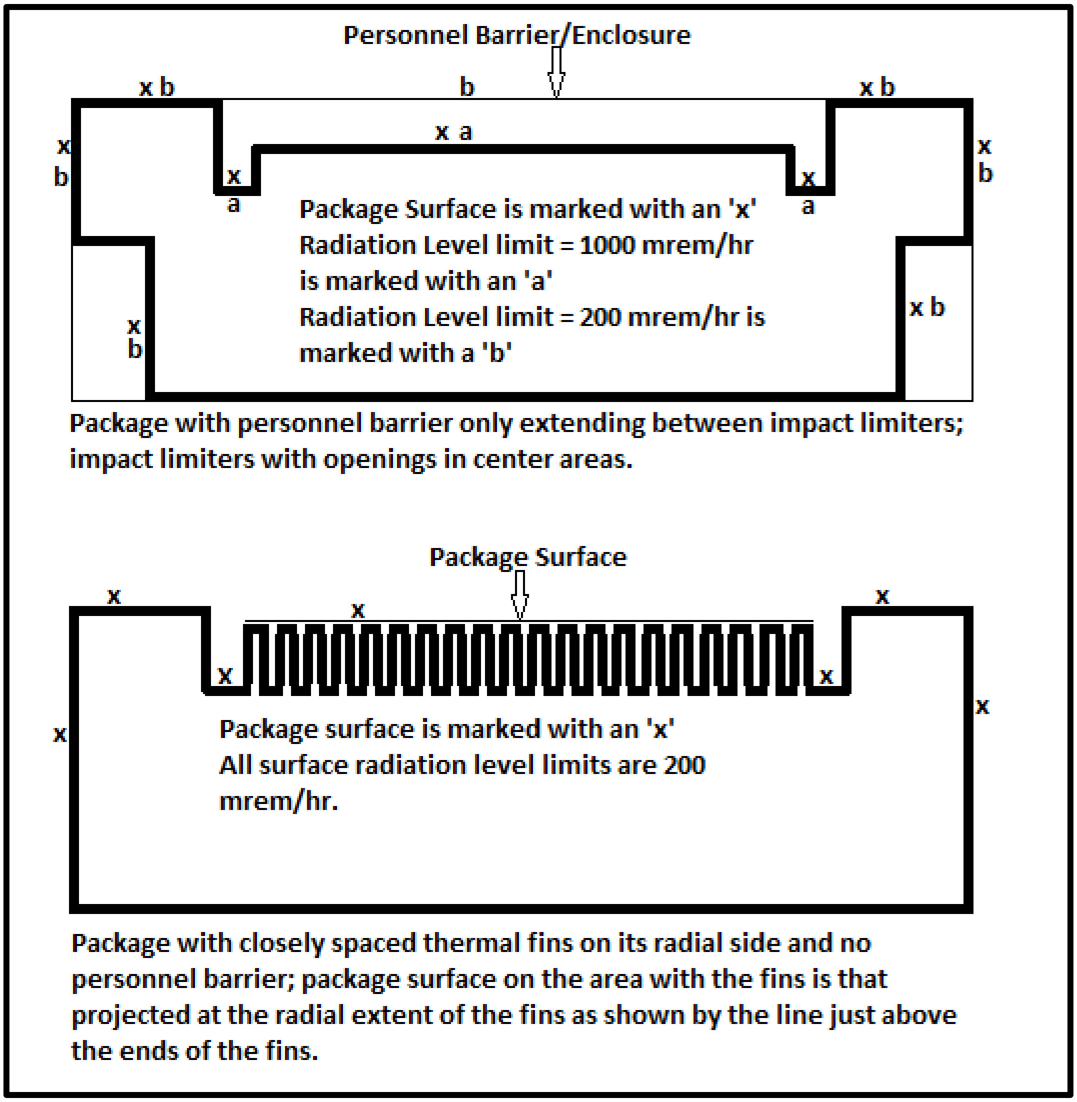

Figure 5-3 Cutaway images depicting package surfaces and radiation-level limits for packages with complex surfaces

the plane projected by the outer edges of the fins. As can also be seen in the figure, in instances where a package may have a personnel barrier that only extends over portions of the package surface (e.g., between the impact limiters on the package's radial side), the higher surface radiation limit for packages in exclusive-use shipments only applies to the package surfaces covered by the personnel barrier. The basic rule is that the 2-mSv/hr (200-mrem/hr) limit applies to any exposed, or accessible, package surfaces. For purposes of 10 CFR 71.47(b), only those parts of the package shown in the drawings and that have been demonstrated to remain in place under the normal conditions of transport evaluations (in 10 CFR 71.71) may be considered to be the external surface of the package.

Confirm that the application addresses damage to the shielding under normal conditions of transport and hypothetical accident conditions. Verify that any damage under normal conditions of transport (under 10 CFR 71.71) does not result in a significant increase in external radiation levels, as required by 10 CFR 71.43(f) and 10 CFR 71.51(a)(1). Ensure that the application includes an explanation of any increase and a justification as to why the increase is not significant. As stated earlier in this SRP chapter, the NRC has often accepted analyses of packages modeled with damage from the 10 CFR 71.71 evaluations that show compliance with the 10 CFR 71.47 limits as adequate demonstrations of compliance with 10 CFR 71.43(f) and 10 CFR 71.51(a)(1), as well. With regard to hypothetical accident conditions, note that some shielding components, such as external neutron shielding, may not be designed to remain in place or may sustain significant enough damage so that they cannot be credited or relied on under these conditions. Also, personnel barriers and enclosures cannot be credited for hypothetical accident conditions, as these also are not designed to survive these conditions, and the limits are for radiation levels at 1 meter [40 inches] from the package's surfaces.

Confirm that the applicant's evaluation provides radiation levels for the contents and source terms that result in maximum radiation levels for the different package surfaces. As described previously, the same contents and source terms may not be bounding for all package surfaces or for all conditions. The shielding characteristics at different locations on the package and the impacts of the evaluation and test conditions will influence what source terms are bounding at which package surface locations and under which conditions. For example, SNF contents with a more dominant neutron-source term may be bounding for package surfaces located away from the package's neutron shielding or in hypothetical accident conditions when the neutron shielding is lost, but SNF contents with a more dominant gamma source term may be bounding otherwise.

Confirm that the applicant's evaluation addresses potential shifting of the package contents and redistribution of the source terms that are possible based on the package design, conditions incident to transport, and the impacts of the normal conditions of transport evaluations and the hypothetical accident conditions tests. The contents and source terms should be shifted so as to maximize the radiation levels associated with the package as designed and for the types of damage sustained from the different condition evaluations and tests. This also includes any kind of credible and bounding reconfigurations of the contents such as for loose particulates or debris. Similarly, ensure that the applicant's evaluation addresses this for any high-burnup fuel in a SNF package, consistent with the applicant's approach to high-burnup fuel as modified by the materials, structural, and thermal reviews.

In determining maximum external radiation levels, radiation levels may be averaged over the cross-sectional area of a radiation probe, with an appropriate size for such types of measurements (see HPPOS-013 in NUREG/CR-5569). For the applicant's analysis of package radiation levels, ensure the tally or detector sizes are appropriate for the contents configurations allowed in the package and the axial or radial variation of the package features relevant to shielding performance. For example, for package features such as streaming paths or voids or localized damage from the normal conditions of transport evaluations or the hypothetical accident conditions tests, ensure that the applicant selected tally or detector sizes such that radiation levels associated with such features or damage are not averaged with radiation levels for package areas around the features or damage. Also, ensure the applicant did not otherwise apply averaging to reduce the radiation levels attributed to such features or damage.

Also, if transport is by exclusive use (as is typical for commercial SNF), the application may also include an evaluation for radiation levels in normally occupied vehicle locations to address 10 CFR 71.47(b)(4). As required in that paragraph, the radiation level limit for these locations is 2 mrem/hr unless the vehicle occupants wear dosimetry devices under a radiation protection program in conformance with 10 CFR 20.1502. If included, ensure this evaluation and the results are consistent with the analysis and results for the analyses against the other limits in 10 CFR 71.47(b). Note, however, that determination of the need for dosimetry for these locations is determined at the time of shipment and not by analyses in the application.

Though not an external radiation-level issue, some packaging components may be sensitive to radiation exposure or have thresholds of exposure to gamma or neutron radiation above which the components' material properties and performance degrades (e.g., polymer-based containment seals). Therefore, coordinate with the materials reviewer to determine the need to evaluate the applicant's calculation of the gamma radiation levels and neutron fluences the packaging components will experience. This evaluation involves determination of an appropriate time over which the exposure accumulates. The results of this evaluation may play an important role in determining the frequency with which such components are repaired or replaced as part of the maintenance programs described in the application and incorporated into the CoC by reference. Verify that the application adequately describes the calculation method and that the method is appropriate for and correctly used to determine the gamma and neutron exposures for the packaging components. Ensure that the analysis also appropriately identifies and accounts for uncertainties in the analysis, as appropriate.

5.4.4.5 Confirmatory analyses

Perform confirmatory analyses, as appropriate, of the shielding calculations reported in the application, to the extent necessary. A number of factors should be considered in determining the level of effort for such confirmatory analyses. These factors include the expected magnitude of radiation levels, the margins between the analyzed radiation levels and the regulatory limits, similarity with previously reviewed packages, thoroughness of the review of source terms and other input data, radiation contributions from difficult-to-measure neutrons, the complexity of the package design, the complexity and variety of the proposed package contents, the degree of sophistication of the applicant's analysis methods, the limitations of these methods and their potential impacts on results, the degree of conservatism in the applicant's analyses, the applicant's experience with these methods (as demonstrated in previous submittals), and the assumptions used in the analyses.

At a minimum, examine the applicant's input to the computer program used for the shielding analysis. For noncomputer code methods, examine the data the applicant used in that analysis and ensure the applicant's use and manipulations of that data are appropriate and correct. Verify the use of proper package dimensions, material properties and composition, contents and source specifications and distributions, cross-section sets (including couple cross-section sets where necessary), attenuation and buildup factors, parameters or other options to address subcritical neutron multiplication, and correct factors to convert fluence rates to radiation levels, as applicable to the package, its contents, and the analysis methods. Also, independently evaluate the use of the gamma- and neutron-source terms, as applicable to the package contents and the analysis methods.

If a more detailed evaluation is deemed necessary, independently evaluate projected radiation levels to ensure that the application results are reasonable and conservatively bounding. As previously noted, the use of a simple code for neutron calculations is often not appropriate. An extensive evaluation would be necessary if significant errors or large uncertainties are suspected or noted in the review. If feasible, use a different shielding code or other appropriate analysis method with different analytical techniques and cross-section set (or other necessary data, as applicable to the analysis method) from that of the application to conduct an independent evaluation and confirm the application results.

Coordinate with the thermal and containment reviewers to determine the need to independently confirm the estimated source terms (i.e., decay heat and radionuclide quantities) and their uncertainties for these reviews. The items can be calculated with the codes used to calculate radiation source terms or other appropriate methods. For calculations using computer codes, refer to the literature regarding these codes for information about the calculation uncertainties. For example, for SCALE, this information is included in various NRC-sponsored studies (e.g., ORNL/TM-13315; ORNL/TM-13317; and NUREG/CR-5625, "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data," issued July 1994). Also, coordinate with the materials reviewer to determine the need to independently confirm the estimated gamma radiation levels and neutron fluences for the packaging components, particularly those that are sensitive to radiation or have threshold levels above which the components may degrade from the radiation exposure.

5.4.5 Appendix

The applicant may provide some of the information described in the preceding sections in one or more appendices to the shielding section of the application (as opposed to the main body of that section). In such a case, confirm that the relevant appendices present all supporting information necessary to confirm that the package meets the radiation requirements in 10 CFR Part 71. This information includes, but is not limited to, a list of references, copies of applicable references that are not generally available, specifications and performance data for nonstandard packaging materials (e.g., polymer-based neutron shields), descriptions of source terms, radionuclide inventories, neutron and gamma energy spectra, descriptions of analytical methods (e.g., computer codes) or measurement methods, input and output files, results of test and sensitivity analyses, analytical method benchmarking and validation information, and other appropriate supplemental information.

5.5 Evaluation Findings

Prepare evaluation findings upon satisfaction of the regulatory requirements in Section 5.3 of this SRP chapter. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F5-1 The staff has reviewed the application and finds that it adequately describes the package contents and the package design features that affect shielding in compliance with 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b), and provides an evaluation of the package's shielding performance in compliance with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a). The descriptions of the packaging and the contents are adequate to allow for evaluation of the package's shielding performance. The evaluation is appropriate and bounding for the packaging and the package contents as described in the application.

F5-2 The staff has reviewed the application and finds that it demonstrates the package has been designed so that under the evaluations specified in 10 CFR 71.71 (normal conditions of transport), and in compliance with 10 CFR 71.43(f) and 10 CFR 71.51(a)(1), the external radiation levels do not significantly increase.

F5-3 The staff has reviewed the application and finds that it demonstrates that under the evaluations specified in 10 CFR 71.71 (normal conditions of transport), external radiation levels do not exceed the limits in 10 CFR 71.47(a) for nonexclusive-use shipments or 10 CFR 71.47(b) for exclusive-use shipments, as applicable.

F5-4 The staff has reviewed the application and finds that it demonstrates that under the tests specified in 10 CFR 71.73 (hypothetical accident conditions), external radiation levels do not exceed the limits in 10 CFR 71.51(a)(2).

F5-5 The staff has reviewed the application and finds that it identifies codes and standards used in the package's shielding design and in the shielding analyses, in compliance with 10 CFR 71.31(c).

F5-6 The staff has reviewed the application and finds that it includes operations descriptions, acceptance tests, and maintenance programs that will ensure that the package is fabricated, operated, and maintained in a manner consistent with the applicable shielding requirements of 10 CFR Part 71.

F5-7 [For packages intended to ship plutonium by air] The staff has reviewed the application and finds that it demonstrates that under the tests specified in 10 CFR 71.74 (accident conditions for air transport of plutonium) and 10 CFR 71.64(b)(2), the external radiation levels do not exceed the limits in 10 CFR 71.64(a)(1)(ii).

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

> Based on its review of the information and representations provided in the application and the staff's independent, confirmatory calculations, the staff has reasonable assurance that the proposed package design and contents satisfy the shielding requirements and the radiation level limits in 10 CFR Part 71. The staff also considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices, in reaching this finding.

5.6 References

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

49 CFR Part 173, "Subpart I—Class 7 (Radioactive) Materials."

Ade, B.J. "SCALE/TRITON Primer: A Primer for Light Water Reactor Lattice Physics Calculations" (NUREG/CR-7041, ORNL/TM-2011/21), Oak Ridge National Laboratory, Oak Ridge, TN, November 2012.

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NRC Information Notice 83-10, “Clarification of Several Aspects Relating to Use of NRC-Certified Transport Packages.” March 11, 1983.

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