6 SHIELDING EVALUATION

6.1 Review Objective

For certificate of compliance (CoC) applications, the objective of the U.S. Nuclear Regulatory Commission (NRC) shielding review is to ensure that the design features relied on for shielding provide adequate protection against direct radiation from the dry storage system (DSS) contents. The shielding features should limit the direct radiation dose to the operating staff and members of the public so that the total dose (i.e., due to direct radiation and any effluents or releases) remains within regulatory requirements during design-basis normal operating, off-normal (aka anticipated occurrences), and accident conditions (all of which are referred to as design-basis conditions in many locations in this chapter of the Standard Review Plan (SRP)). The review seeks to ensure that the shielding design is adequately defined and evaluated to support the evaluation of the following:

- the DSS's compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 72.236(d)—the DSS has shielding and confinement features sufficient to meet the requirements in 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS," and 10 CFR 72.106, "Controlled Area of an ISFSI or MRS."

- the occupational doses from operations with the DSS and adequate consideration of "as low as is reasonably achievable" (ALARA) in the DSS design and operations

The NRC staff conduct an assessment of compliance with these requirements and criteria in its radiation protection review (see Chapter 10B, "Radiation Protection Evaluation for Spent Fuel Dry Storage Systems," of this SRP).

For specific license applications, the objective of the NRC shielding review is to determine whether the shielding design features of the dry storage facility (DSF), whether an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), meet the NRC criteria for protection against direct radiation from the material to be stored. In particular, this evaluation should establish the validity of dose rate estimates made in the applicant's safety analysis report (SAR). These estimates are in turn used in the radiation protection review (described in Chapter 10A, "Radiation Protection Evaluation for Dry Storage Facilities," of this SRP) to determine (1) compliance with regulatory limits for allowable doses, and (2) conformance with criteria for maintaining ALARA with respect to radiation exposures. The scope for a specific license review is limited to evaluating the shielding for the radiation sources from spent nuclear fuel (SNF), reactor-related greater-than-Class-C (GTCC) waste, or high-level radioactive waste (HLW) to be stored. SRP Chapter 10A and Chapter 13, "Waste Management Evaluation," address other radiation sources at the ISFSI or MRS for which shielding may be required.

6.2 Applicability

This chapter applies to the review of applications for specific licenses for an ISFSI and MRS, categorized as a DSF. It also applies to the review of applications for a CoC for a DSS for use at a general license ISFSI. Sections, paragraphs, or tables that apply only to specific license applications have "**(SL)**" in the heading and apply to all relevant facility design features, operations, and contents. This includes any reactor-related GTCC waste and HLW (for MRSs only) as well as SNF to be stored at the facility and facility structures, systems, and components

(SSCs) and features in addition to the storage containers to be used at the facility. Sections, paragraphs, or tables that apply only to CoC applications have "**(CoC)**" in the heading and apply only to the DSS design features, operations, and contents, which are limited to SNF and the associated radioactive materials (referred to as nonfuel hardware (NFH)). A subsection without an identifier applies to both types of applications; however, the scope of review differs for the two application types.

6.3 Areas of Review

This chapter addresses the following areas of review:

- shielding design description
  - design criteria
  - design features
- radiation source definition
  - initial enrichment
  - computer codes for radiation source definition
  - gamma sources
  - neutron sources
  - other parameters affecting the source term
- shielding model specification
  - configuration of shielding and source
  - material properties
- shielding analyses
  - computer codes
  - flux-to-dose-rate conversion
  - dose rates
  - confirmatory analyses
- consideration of reactor-related GTCC waste storage **(SL)**
- supplementary information

6.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," that are relevant to the review areas addressed by this chapter. The NRC reviewer should refer to the exact language in the regulations. Table 6-1a provides the relevant regulatory requirements for a specific license review. Table 6-1b matches the relevant regulatory requirements for the areas of review covered in this chapter for a CoC. The NRC staff reviewer should verify the association of regulatory requirements with the areas of review presented in the tables to ensure that no requirements are overlooked as a result of unique design features.

Table 6-1a Relationship of Regulations and Areas of Review (SL)

Areas of Review	10 CFR Part 72 Regulations						
	72.24	72.104 (a)	72.106 (b)	72.120	72.122 (b)(2) (i),(c),(e)	72.126 (a)(6)	72.128 (a)(2)
Shielding Design Description	(b)(c)(e)	●	●	(a)(b)(c)	●	●	●

Radiation Source Definition	(c)	●	●	(b)(c)			
Shielding Model Specification	(b)(c)(e)	●	●	(b)(c)	●	●	●
Shielding Analyses	(m)(e)	●	●	(b)(c)	●	●	●
Consideration of Reactor-Related GTCC Waste Storage	(b)(c)(e)	●	●	(a)(b)(c)	●	●	●

Areas of Review	10 CFR Part 20 Regulations		
	20.1201 (a)(1)(2)	20.1301(a)(b)	20.1302(b)
Shielding Design Description	●	●	
Radiation Source Definition	●	●	
Shielding Model Specification	●	●	
Shielding Analyses	●	●	●
Consideration of Reactor-Related GTCC Waste Storage	●	●	●

Table 6-1b Relationship of Regulations and Areas of Review (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.104(a)[A]	72.106(b)[A]	72.236
Shielding Design Description	●	●	(b)(d)(g)
Radiation Source Definition	●	●	(a)
Shielding Model Specification	●	●	(d)(g)
Shielding Analyses	●	●	(d)(g)

A This requirement applies to CoCs and CoC applications through the requirement in 10 CFR 72.236(d).

The regulations in 10 CFR Part 72 require that SNF (including NFH), reactor-related GTCC waste and HLW storage and handling systems be designed with adequate shielding to provide sufficient radiation protection under normal, off-normal, and accident conditions. The SAR should describe the design principles and functional features of the SSCs important to safety that are relied on for shielding in sufficient detail to allow the NRC staff to thoroughly evaluate their effectiveness. It is the responsibility of the applicant to analyze such SSCs with the objective of assessing the impact of direct radiation doses and effluent releases to the environment on public health and safety.[1] The NRC reviewer should verify the applicant's evaluations through review of the applicant's model and, as needed, through confirmatory analyses or independent modeling analysis. In addition, SSCs important to safety should be designed to withstand the effects of both credible accidents and severe natural phenomena without impairing their capability to perform their safety functions. While only applicable to licenses, 10 CFR 72.122(b) and (c) provide a list of the kinds of conditions for which a DSS should be designed and evaluated in a CoC application.

(CoC) Several technical and licensing factors should be considered during the shielding evaluation. First, 10 CFR Part 72 specifies regulatory dose limits in terms of annual doses for normal conditions and total dose from accident conditions. These limits apply to individuals located at or beyond the controlled area boundary of a DSF. The regulations do not specify dose rate limits for DSS surfaces nor at set distances from DSS surfaces, unlike the package dose rate

[1] For CoC applications, as noted in other sections of this guidance, the general licensee is responsible for the ultimate assessment of these impacts for its use of the DSS design in an approved CoC.

limits in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." Therefore, responsibility for determining compliance with the dose limits in 10 CFR 72.104(a) and 10 CFR 72.106(b) ultimately rests with the general licensee that uses the DSS at its ISFSI (see 10 CFR 72.212, "Conditions of general license issued under § 72.210," which places this responsibility with the general licensee; compliance is verified by inspection). This is because compliance with these kinds of limits considers factors that are specific to the general licensee's site. These factors include the geometric arrangement of DSS arrays, topography, distances to the controlled area boundary, distances to dose receptors, exposure times of dose receptors, actual SNF loading patterns in each DSS, and dose contributions from other surrounding fuel-cycle facilities. Because the SAR is only for a DSS design that is intended to be usable by general licensees, the SAR analyses cannot fully address these factors for sites at which the DSS might be used. This does not mean, however, that compliance with the requirements in 10 CFR 72.104 and 10 CFR 72.106 is the sole responsibility of the licensee. As stated in 10 CFR 72.234(a), the certificate holder and applicant for a certificate must ensure that the design, fabrication, testing, and maintenance of a DSS comply with the requirements in 10 CFR 72.236. This includes 10 CFR 72.236(d), which requires the CoC applicant to demonstrate that the DSS shielding, together with the DSS confinement, is sufficient to meet the requirements in 10 CFR 72.104 and 10 CFR 72.106. Given the site-specific factors that do bear upon compliance with those requirements, the typical acceptance criteria for DSS shielding define standard analyses for single DSSs, and a generic array of DSSs, to demonstrate a sufficient shielding design and compliance with 10 CFR 72.236(d).

(CoC) In general, the DSS shielding evaluation should provide reasonable assurance that the proposed design fulfills the following acceptance criteria:

- The radiation shielding features of the proposed DSS must be sufficient for it to meet the radiation dose requirements in 10 CFR 72.104. The applicant demonstrates this by providing the following:

  - a shielding analysis of the surrounding dose rates that contribute to offsite doses at appropriate distances (for a single storage overpack and transfer cask (for a canister-based DSS) or a single cask (for a non-canister-based DSS) with bounding fuel source terms at various overpack and transfer cask, or cask, locations) for normal conditions and anticipated occurrences (that is, off-normal conditions)

  - a shielding analysis of a single DSS and a generic array of DSSs at appropriate distances

- DSS contents and design features important to and relied on for shielding are adequately described for evaluating shielding effects and dose rates. Dose rates are evaluated for an adequate number of appropriate locations around the DSS for different operations configurations to enable evaluation of occupational dose estimates and evaluation of ALARA.

- Radiation shielding features must be sufficient for the design to meet the requirements in 10 CFR 72.106. The applicant demonstrates this by calculating dose rates and doses at appropriate distances for different accident conditions for appropriate DSS configurations and appropriate assumptions regarding accidents (e.g., duration, including time to recover from or repair the effects of the accidents).

- The proposed shielding features should enable a general licensee that uses the DSS to meet the regulatory requirements prescribed in 10 CFR Part 20, “Standards for Protection Against Radiation.”

- Appropriate distances for the foregoing criteria are distances that are consistent with, or bounding for, the distances to the controlled area boundaries of potential DSS users. The minimum distance to the controlled area boundary is 100 meters (328 feet).

(SL) As described in the guidance for CoC applications, 10 CFR Part 72 only specifies dose limits for individuals located at or beyond the controlled area boundary; it does not specify dose rate limits for storage containers such as DSSs. Demonstration of compliance with the limits necessarily considers factors associated with the facility’s site. For specific license applications, the site and its surroundings are known. Thus, site factors should be considered as part of the applicant’s analysis, or the applicant should provide an analysis that is bounding for its site and describe how the analysis is bounding. The contents to be stored at the site are limited in characteristics and in quantity. Thus, the analysis should be bounding for the characteristics of what is to be stored at the site and should account for the maximum quantity to be stored at the site. This means the analysis should account for the number, configuration(s), and size(s) of the array(s) that will be employed at the site. Additionally, the SAR should describe the locations of members of the public (e.g., residences, places of work, and public access facilities or areas) and projections of changes known for the site (see Chapter 2, “Site Characteristics Evaluation for Dry Storage Facilities,” of this SRP). The application should also include a description of the controlled area and any restricted areas on the facility site. Site topography is also fixed for the site. These factors should be appropriately accounted for in the analysis or bounded by the analysis.

(SL) In the radiation protection chapter (see SRP Chapter 10A), the SAR should include evaluations that demonstrate that the facility design and operations meet, or will meet, the requirements in 10 CFR 72.104 and 72.106 and 10 CFR Part 20. The shielding analysis should be adequate to support that demonstration. This includes providing dose rates for (1) the storage container (e.g., DSS) surfaces and near the containers, (2) the surfaces and vicinity of other facility SSCs used to handle or transfer the material stored at the site, and (3) locations around the facility, including within restricted areas and in facility buildings and structures where facility personnel will be, or may be, located. The dose rates should include dose rates for different phases of operations and container configurations and should address the effects of different conditions (normal, off-normal, accident, which include natural phenomena) during these operations. The dose rate estimates should be sufficient in number and location to support evaluation of occupational doses and incorporation of ALARA as well as doses to members of the public.

The acceptance criteria also help to ensure the dose rates associated with the DSS or DSF are reasonable and acceptable. The acceptance criteria also help to ensure that the methods used to calculate the dose rates are appropriate and acceptable in terms of the methods’ use to demonstrate the DSS’s or DSF’s SSCs, as described in the application, fulfill the shielding safety function. The staff should be aware of the potential for further use of these methods and may therefore need to place additional emphasis on appropriate acceptance criteria related to methods. Such a review, however, still does not constitute approval of the methods outside of their use to demonstrate that the DSS or DSF, as described in the application, meets the shielding requirements.

In order to ensure that the shielding design of the DSS or DSF meets the regulatory requirements as defined in 10 CFR Part 72, the applicant should also include information in the SAR regarding the technical specifications that are necessary for the DSS or DSF to meet the dose limits at the controlled area boundary (see SRP Chapter 17, "Technical Specifications Evaluation"). The requirements to be included in technical specifications are described in 10 CFR 72.44(c). While only applicable to specific licenses, the information in 10 CFR 72.44(c) can be useful in determining the information needed in CoC conditions, including those referred to as technical specifications, to ensure compliance with 10 CFR 72.234(a) and 10 CFR 72.236.

6.4.1 Shielding Design Description

(SL) For a specific license, 10 CFR 72.126, "Criteria for radiological protection," and 10 CFR 72.128, "Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling," require that the applicant describe the storage and handling systems requiring shielding. The SAR must provide design criteria and descriptions of design features relied on for shielding for facility features and facility SSCs that are used to store, handle, or transfer the material to be stored at the facility in accordance with 10 CFR 72.24(b) and 10 CFR 72.24(c).

6.4.1.1 *Design Criteria*

The requirements in 10 CFR 72.104 and 10 CFR 72.106 provide dose limits for the members of the public around a DSF site (i.e., offsite). The SAR chapter on principal design criteria should specify the criteria that have been used as a basis for protection against direct radiation. Design criteria should include the identification of maximum dose rates and should also be specified for occupancy areas and correlated with occupancy duration and distance to radiation sources.

The design should consider the ALARA principle. For CoC applications, the NRC reviewer should note that it is the responsibility of the general licensee using the DSS design to develop detailed procedures that incorporate the ALARA objectives of its site-specific radiation protection program. However, the DSS design should reflect appropriate consideration of ALARA to the extent practical. For specific license applications, the SAR should include sufficient information to demonstrate incorporation of ALARA into the facility design, including facility layout, and operation procedures. SRP Chapters 10A and 10B (radiation protection) provide further information on ALARA considerations that apply to the respective reviews.

(SL) In addition to the limits in 10 CFR Part 72, 10 CFR 20.1201, "Occupational dose limits for adults," and 10 CFR 20.1301, "Dose limits for individual members of the public," prescribe additional dose limits for personnel and for members of the public, respectively.

6.4.1.2 *Design Features*

The SAR should describe the material and geometric properties of all design features relied on to reduce direct radiation dose rates and may consider the following:

- self-shielding provided by the radioactive material being stored
- shielding provided by the structural and nonstructural materials forming the DSS or DSF SSCs (e.g., a SNF cask, overpack, or transfer cask)
- neutron capture provided by borated materials incorporated into the DSS or DSF SSCs

- shielding provided by the temporary placement of water into DSS or DSF SSCs (e.g., into SNF canister and transfer cask) during loading and unloading procedures

- shielding provided by temporary placement of equipment and portable shields on and around the DSS or DSF SSCs during loading and unloading procedures (for DSSs, this means only those items that are part of the DSS design)

- shielding provided by natural or human-made, engineered (e.g., berms or shield walls) barriers between the radioactive material and the area beyond the controlled area boundary; human-made, or engineered features, used for this purpose (i.e., to ensure compliance with regulatory dose limits such as 10 CFR 72.104(a)) should be classified as important to safety at the appropriate category. Such features are most likely not part of DSS designs and analyses, though they may be.

(SL) The guidance in the preceding list applies to all DSF SSCs, not just the SSCs associated with the storage containers used at the DSF. The following includes some examples of these other DSF SSCs to which the preceding guidance list applies:

- shielding provided by pool or other site facility SSCs, including interior and exterior walls
- shielding to reduce dose to personnel in site facilities such as the administrative building

The SAR should describe the geometric arrangement of shielding and include illustrations that identify the spatial relationships among sources, shielding, and design dose rate locations. For specific license applications, this description should include scaled layout and arrangement drawings of the facility that show the locations of all sources and facility SSCs and features. The SAR should clearly indicate the physical dimensions of sources and shielding materials. The SAR should also identify penetrations, voids, or irregular geometries that provide potential paths for gamma or neutron streaming. Any submitted drawings should clearly identify these potential streaming paths. The SAR should describe design features used to minimize streaming through these penetrations.

The SAR should adequately describe the material properties and specifications, including composition of the items relied on for shielding. This information is particularly needed for nonstandard or proprietary materials such as proprietary polymer-based neutron shielding. The SAR should include appropriate references for the nonstandard or proprietary materials. Additionally, the technical design (or engineering) drawings should include material specifications important to the performance of the shield materials. These specifications include items such as the industry standard for the specifications of the lead gamma shielding and the minimum mass density, hydrogen composition, and boron composition of polymer-based neutron shielding.

The SAR should clearly state any differences in shielding features (material properties, geometry, and dimensional changes) for normal, off-normal, and accident conditions. These differences may be from effects such as physical impacts and material property changes caused by temperature effects. The SAR descriptions for the different conditions should consider different operating configurations that, though temporary, affect how the different conditions may affect the DSS or DSF shielding features. For example, a DSS or DSF design that relies on soil providing shielding for the storage containers should address the impacts of normal, off-normal, and accident conditions for excavation (to expand the storage array) next to operating (i.e., loaded) storage containers.

6.4.2 Radiation Source Definition

The SAR should describe the radioactive contents to be stored. For CoC applications, the allowable contents are limited to SNF and any NFH to be stored with the SNF. That description should include the condition of the SNF (e.g., undamaged, damaged). For specific license applications, the contents may also include solid reactor-related GTCC waste and, for MRSs, HLW. The SAR should include an adequate description of these items, including the physical and chemical form(s), radionuclide content, and geometric configuration(s).

The SAR should describe each type of contained radiation source used as a basis for the shielding design calculations. The source terms should be described in a format that is compatible with the shielding calculation input. For SNF, the source terms in particles per second per metric ton of uranium (MTU) (or metric ton heavy metal (MTHM) for mixed-oxide (MOX) SNF) or per assembly (e.g., neutron per second per MTU (n/s/MTU), gamma per second per assembly (γ/s/assembly)) or, for gammas, million (mega) electron volts per second (MeV/s) per MTU (or MTHM for MOX SNF) or per assembly (i.e., MeV/s/MTU or MeV/s/assembly) should be described in the form of either a group structure or a continuous function of energy. For assembly hardware and NFH, the source can be described in terms of the nuclide(s) in the hardware and the activity (in curies or becquerels) of the nuclide(s). For reactor-related GTCC waste and HLW contents in specific license applications, the SAR should specify the isotopic composition and photon yields and, as appropriate, neutron yields for each constituent in the waste.

The SAR should clearly present the data used as input for calculating the radiation source terms and include the bases for the parameter values selected for the input. This includes any material property, physical dimension, and irradiation history values that differ from the actual properties of the radioactive contents or are derived from assumptions (e.g., assumed down time between irradiation cycles for SNF). The applicant should show that the selected input values result in appropriate or conservative results. The energy group structure from the source term calculation should correspond to that of the cross-section set of the shielding calculation. In addition, the SAR should specify the computer methodology or database application used to compute source term strength.

The SAR should include a discussion of energetic radiations created by nuclear reactions such as (n, γ) in the materials and the contents of the DSS or DSF SSCs. The SAR should also provide source-term descriptions for induced radioactivity and the bases (assumptions and analytical methods) used for their estimation. For example, high-energy (approximately 6.7-MeV) gammas may be generated by the (n, γ) reaction of thermalized neutrons and the iron in the steel shell that is typically used to contain liquid or polymer-based neutron shields. Alternatively, the SAR may describe the bases for excluding induced radioactivity source terms.

6.4.2.1 *Gamma Sources*

The SAR should specify gamma source terms for both SNF and activated materials. Most hardware source terms will be from cobalt-60; however, some NFH may include other activated nuclides that should be evaluated (e.g., materials containing hafnium or silver-indium-cadmium). For reactor-related GTCC waste and HLW contents in specific license applications, the isotopic composition and photon yields for each constituent should be specified. A tabulated form of the radiological characteristics is acceptable.

6.4.2.2 *Neutron Sources*

The SAR should also describe the neutron source terms, both total strength and spectrum, for the SNF and for neutron sources and neutron source assemblies (NSAs) included as NFH contents. The description should also include the bases used to determine the source terms. The SAR should also describe how the analysis addresses neutrons from subcritical multiplication. For reactor-related GTCC waste and HLW contents in specific license applications, similar information should be included in the SAR for neutron sources in these wastes, if applicable. The neutron source term for these wastes may be specified in terms of the constituent radionuclides with their respective neutron yields and spectra. Alternatively, contents limits in the license conditions or technical specifications may limit these wastes such that they have a negligible neutron source. In that case, the SAR should describe how the contents specifications result in a negligible neutron source from these wastes and thus neutron source information is not needed for them.

6.4.3 Shielding Model Specification

The SAR should identify the models used in the analysis and include information on materials and arrangements of sources and design features included in the models. As described in Sections 6.4.3.1 and 6.4.3.2 below, the SAR should clearly present the data used in the analyses, identifying differences between actual properties and modeled properties of SSCs and features and of material to be stored and justifying the acceptability of those differences, whether they are from simplifications or assumptions or other reasons.

6.4.3.1 *Configuration of Shielding and Source*

The SAR should include descriptions of how the sources and DSS or DSF design SSCs and features are included in the analysis models. The SAR should justify how the models adequately include the sources and DSS or DSF design SSCs and features. The SAR should also justify any simplifications of features in the model and, for features that are not represented in the models, the acceptability of not including these features in the models. The analysis should include models that represent the source and design feature configurations that are appropriate for the different stages of operations (e.g., storage at the pad, loading, draining, and drying) and are appropriate for normal, off-normal, and accident conditions for the different stages of operations. The models should consider the information in Sections 6.4.1 and 6.4.2 of this SRP and, for SNF contents, the condition of the SNF (e.g., undamaged, damaged, debris). The analysis models should also include appropriate or bounding physical distribution(s) of the source term(s). See the section of Chapter 8, “Materials Evaluation,” of this SRP that discusses the condition of SNF.

6.4.3.2 *Material Properties*

The SAR should describe how materials specifications and properties for the DSS or DSF contents and design features are included in the models. The SAR should justify that the materials properties in the models are adequate, bounding, or otherwise appropriate for representing the materials that comprise the DSS or DSF contents, SSCs and design features for different configurations (e.g., damaged SNF vs. undamaged SNF), conditions (i.e., normal, off-normal, accident conditions) and operations configurations (e.g., draining, drying, storage at the pad), considering the information in Sections 6.4.1 and 6.4.2 of this SRP chapter.

6.4.4 Shielding Analyses

The SAR should describe the computer codes, including version; computational models; data; and assumptions with their bases used in evaluating shielding effectiveness. It should provide dose rate estimates for areas of concern, as described near the beginning of the shielding evaluation acceptance criteria.

6.4.4.1 *Computer Codes*

The SAR should identify the computer codes used in the shielding evaluation, including codes for calculating the source term descriptions identified in Section 6.4.2 above and codes for calculating dose rates, and reference the appropriate documentation. For each computer code used, the SAR should provide test problem solutions that demonstrate substantial similarity to solutions from other sources (e.g., hand calculations, published literature results). The SAR should provide a summary that compares the test problem solutions in either graphical or numeric form. However, these solutions may be referenced and need not be submitted in the SAR if the references are widely available or have been previously submitted to the NRC for the same computer code and version.

The SAR should address calculational error (i.e., standard error) and uncertainties in computer codes for both radiological and thermal source terms. Because validation data are relatively limited for burnups above 45 gigawatt days/MTU (i.e., high burnup fuel), the SAR should numerically specify radiological and thermal source term uncertainties for high burnup fuels.

The SAR should determine whether and how source term values with uncertainties should be applied to the shielding analysis. The applicant may do this by making adjustments to the source term or by compensating in other aspects of the shielding analysis. In this determination, the SAR may consider the following:

- other conservative assumptions and design margins in the analysis
- the maximum fuel assembly heat loads for the design basis fuel, burnup, enrichment, and cooling time
- the maximum gamma and neutron dose rates (including relative contributions to total dose rates)
- any measurable dose rate limitations proposed in the technical specifications
- the gamma and neutron sources corresponding to the design basis decay heat limit

The applicant should calculate dose rates with a code that is capable of handling the geometries and configurations of the DSS or DSF design features and SSCs and the contents (i.e., SNF, reactor-related GTCC waste, or HLW) during the different stages of storage operations for normal, off-normal, and accident conditions. This includes storage containers (e.g., DSS) that have axial or radial variations in features relied on for shielding, inlet and outlet vents, and other features that can be streaming paths and, for a DSF, variations in facility features that can affect dose rates. This also includes configurations of contents that result in variations in the physical distribution of the contents' source term, which can also affect dose rates. The SAR should include a description of the shielding code that is sufficient to justify that it is adequate to determine dose rates for the DSS or DSF, considering the DSS's or DSF's design SSCs and features that affect shielding.

The SAR should include representative computer code input files for the different types of calculations done to support the shielding analysis.

6.4.4.2 *Flux-to-Dose-Rate Conversion*

The SAR should state the flux-to-dose-rate conversion used in the shielding analysis, including conversions that are done by a computer code using its own data library, and the basis for using that conversion(s). The SAR should include a table that shows the one-to-one conversion factor for each energy group of the source term spectra. The NRC accepts the flux-to-dose-rate conversion factors in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose Conversion Factors."

6.4.4.3 *Dose Rates*

(CoC) The SAR evaluation of shielding effectiveness should include calculated or estimated dose rates in representative areas around the DSS and at appropriate distances from the DSS. The SAR should clearly indicate the locations on and around the DSSs and the distances from the DSSs for which dose rate calculations have been performed. The selected locations should be adequate to support determination of occupational dose estimates and doses to members of the public described in Chapter 10B of this SRP, demonstrating consideration of the following:

- locations on or in the immediate vicinity of DSS surfaces and at appropriate distances from the DSS where workers will perform operations during loading, retrieval, handling, maintenance, and surveillance activities

- locations of DSS features and surfaces with potentially elevated dose rates or streaming paths such as (labyrinthine) air flow passages; the SAR should include dose-rate estimates for these areas (e.g., air inlets and outlets)

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary (minimum distance to the boundary must be at least 100 meters (328 feet) in accordance with 10 CFR 72.106(b)); locations should be sufficient to develop dose-to-distance curves for a single DSS and a sample array of DSSs for 10 CFR 72.104 evaluations

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary from accidents for 10 CFR 72.106 evaluations

- potential use of some dose rates as limits in the technical specifications

(CoC) Dose rates should be calculated for the variety of DSS configurations that exist at different stages of DSS operations (e.g., storage at the ISFSI pad, DSS loading, DSS welding). Also, dose rates should be calculated for normal conditions, anticipated occurrences, and accidents and natural phenomena to enable evaluation of the doses for each of these conditions.

(CoC) For canister-based systems, the system includes a transfer cask and a storage overpack. Thus, for these DSSs, the various conditions for the different DSS configurations include the transfer cask and overpack. The overpack is a passive, engineered SSC that provides the necessary radiation shielding during storage on the DSF pad. As of the publication of this SRP, overpack designs have included vertical concrete or metal silos, concrete modules, and designs for vertical storage systems that rely on engineered fill and the surrounding soil as the “overpack.” For DSSs with the latter kind of overpack, dose rate analyses should also address normal, off-normal, and accident conditions with excavation (to expand the storage system array) next to loaded systems. This information will support any needed technical specification to limit the proximity of excavation to loaded systems. Transfer casks may also include or make use of supplemental shielding that is necessary for personnel to be able to perform some operations involving the loaded transfer cask (i.e., the third type of supplemental shielding described in the term’s definition in the SRP glossary). The applicant should consider configurations with and without this supplemental shielding in the dose rate analyses for the transfer cask. For non-canister-based systems, all configurations and conditions will involve a single cask, which is used for all operations.

(SL) The SAR evaluation of shielding effectiveness should include calculated or estimated dose rates in representative areas around the storage containers (e.g., SNF container, GTCC waste container) and at appropriate distances from the storage containers and at appropriate locations within, at, and beyond the controlled area boundary. The SAR should clearly indicate the locations on and around the containers and the distances from the containers for which dose rate calculations have been performed. The SAR should clearly indicate the locations within the facility (e.g., within the restricted area, in areas of container-handling buildings, and administrative buildings) and locations at and beyond the controlled area boundary for which dose rates were calculated. The selected locations should be adequate to support determination of occupational dose estimates and doses to members of the public described in Chapter 10A of this SRP, demonstrating consideration of the following:

- locations on or in the immediate vicinity of container surfaces and at appropriate distances from the container and the surfaces of and appropriate distances from facility SSCs used to handle, transfer, or store the containers where workers will perform operations during loading, retrieval, handling, maintenance, and surveillance activities

- locations of container features and surfaces with potentially elevated dose rates or streaming paths such as (labyrinthine) air flow passages; the SAR should include dose rate estimates for these areas (e.g., air inlets and outlets)

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary (minimum distance to the boundary must be at least 100 meters (328 feet) (see 10 CFR 72.106(b))); locations should be sufficient to evaluate doses for members of the public and should include residences, businesses and other places of work, recreational facilities and areas, and other public access facilities and areas around the DSF for 10 CFR 72.104 evaluations

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary from accidents for 10 CFR 72.106 evaluations

- Locations where personnel will be working to support DSF operations (e.g., administrative buildings)

- Locations of public access facilities and areas, including throughways (e.g., roads, highways, waterways, railways) that traverse through the controlled area

- Facility layout and locations of personnel performing DSF operations related to that layout (e.g., surveillance or maintenance conducted on a storage container within an array of containers at a single storage pad, surveillance of containers on one pad from locations surrounded by other pads for a multi-pad facility)

- Potential use of some dose rates as limits in the technical specifications

(SL) The SAR should include calculated dose rates for the variety of container configurations that exist at different stages of storage operations (e.g., storage at the DSF pad, container loading, container welding). Further, dose rates should be calculated for normal conditions, anticipated occurrences, and accidents and natural phenomena to enable evaluation of the doses for each of these conditions. The preceding CoC discussion related to canister-based systems and non-canister-based systems should also be considered, as applicable, for the storage containers to be used at the DSF. Additionally, any supplemental shielding (e.g., berms or shield walls) included in the estimates to demonstrate compliance with dose limits should be classified as important to safety at the appropriate category.

6.4.5 Consideration of Reactor-Related GTCC Waste Storage (SL)

(SL) As described in the preceding sections, an applicant that proposes to store reactor-related GTCC waste at its DSF should ensure that the shielding analysis includes the reactor-related GTCC waste. The applicant should further ensure that the SAR includes all appropriate information to support that analysis. This includes a description of the forms and compositions of different types of reactor-related GTCC waste (e.g., steel core baffle plates), the characterization of the radionuclides and their activities, the total amount of reactor-related GTCC waste to be stored at the facility, and a description of the SSCs, including the containers, used to handle, transfer, and store the reactor-related GTCC waste. The SAR should clearly state that the reactor-related GTCC waste is in solid form since only solid reactor-related GTCC waste may be stored under 10 CFR Part 72. The results of the shielding analysis should include dose rates that can be used to estimate occupational doses for operations for the reactor-related GTCC waste, including the different configurations of SSCs at the different operations stages. The shielding analysis results should include dose rates that include the impacts of reactor-related GTCC waste storage operations for evaluating the doses to members of the public for normal, off-normal, and accident conditions from DSF operations. These dose rates are used in the radiation protection evaluations (see Chapter 10A) to demonstrate facility design and operations meet, or will meet, the requirements in 10 CFR 72.104, 10 CFR 72.106, and 10 CFR Part 20.

(SL) There are multiple ways for the shielding analysis to address reactor-related GTCC waste. First, as may be done for analyses for SNF or HLW contents, the applicant may simply perform dose rate calculations for each type of reactor-related GTCC waste, or the applicant may choose to perform dose rates for a bounding reactor-related GTCC waste type. For this second option, the applicant should demonstrate that the selected waste type results in bounding dose rates for all the reactor-related GTCC waste types to be stored at the DSF. Such a demonstration would include the waste characterization, including the physical distribution of the radionuclides within the waste and any changes to that distribution resulting from the different conditions of operations. Additionally, the applicant may choose to demonstrate that the dose rates for reactor-related GTCC waste are bounded by the dose rates for the SNF or HLW to be stored at the DSF and apply the SNF or HLW dose rates to the reactor-related GTCC waste. If the reactor-related GTCC waste is handled and stored in the same containers as the SNF or HLW, then demonstrating that the bounding reactor-related GTCC waste source term is bounded by the SNF or HLW source term in total strength and across the energy spectra may be sufficient. A final option is that, in the case that the radiation protection evaluation indicates significant margins to the limits in 10 CFR 72.104 for analysis with just the SNF and HLW, as applicable, the applicant may choose to demonstrate that dose rates from GTCC waste are insignificant in comparison with the SNF or HLW dose rates. In this instance, the DSF dose rates would not need to include the reactor-related GTCC waste contribution. This last option only applies to the normal and off-normal conditions dose rates analysis for the 10 CFR 72.104 evaluation. For any one of these options, the SAR should include the appropriate information to support the selected analysis approach.

6.5 Review Procedures

Figures 6-1a and 6-1b show the interrelationship between the shielding evaluation and the other areas of review described in this SRP for specific license and CoC applications, respectively.

Coordinate with the technical specifications reviewer (SRP Chapter 17) to ensure that the license and CoC conditions and technical specifications adequately capture those items that (1) for a DSF, are necessary for the DSF to meet the regulatory dose limits, or (2) for a DSS, are necessary for the DSS to function to enable general licensees that use it to meet the regulatory dose limits. Make a determination that descriptions of the DSS or DSF in the SAR provide the information needed to evaluate the DSS or DSF shielding in the context of its proposed use and operations.

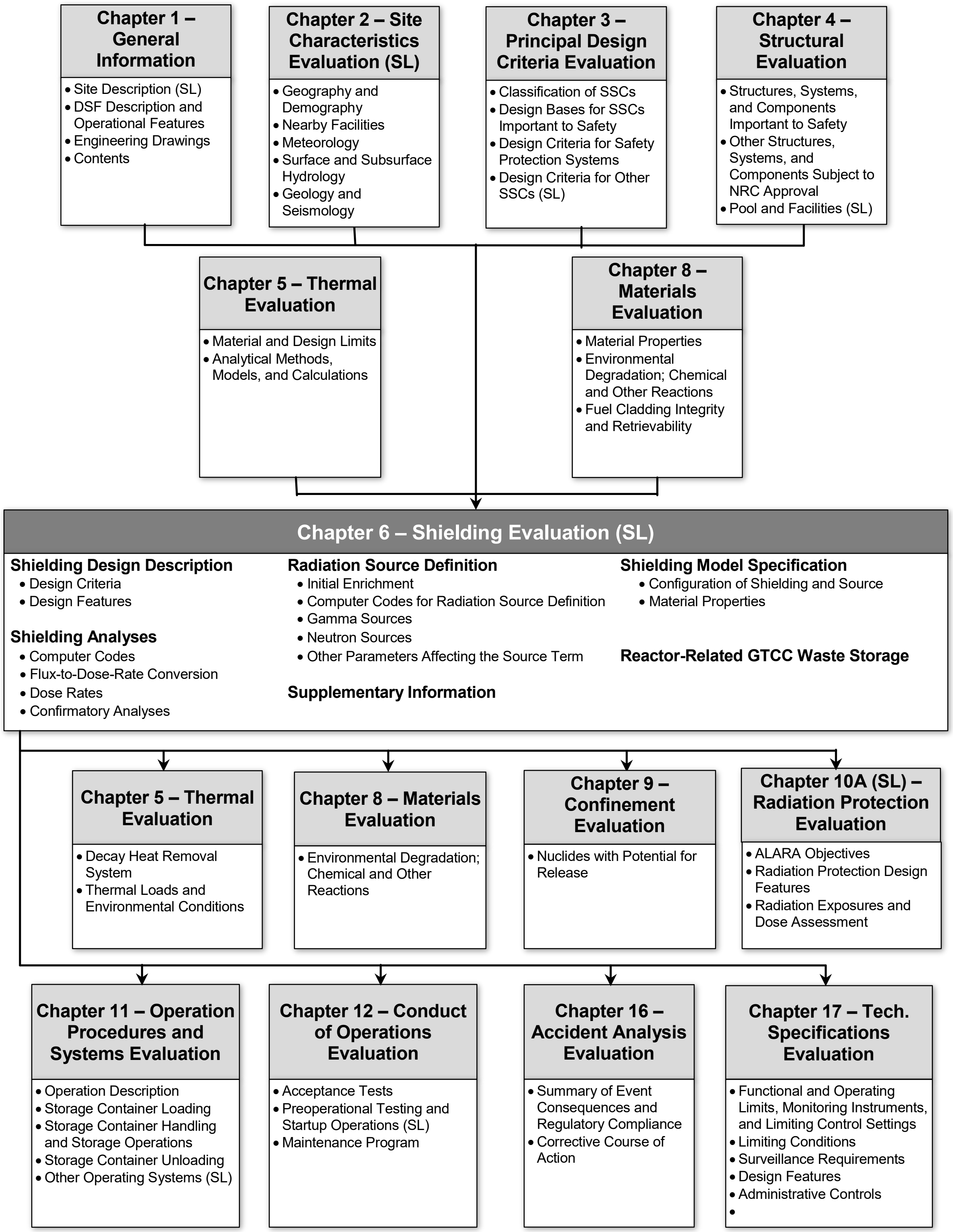

Figure 6-1a Overview of Shielding Evaluation of Specific License Applications for a DSF (SL)

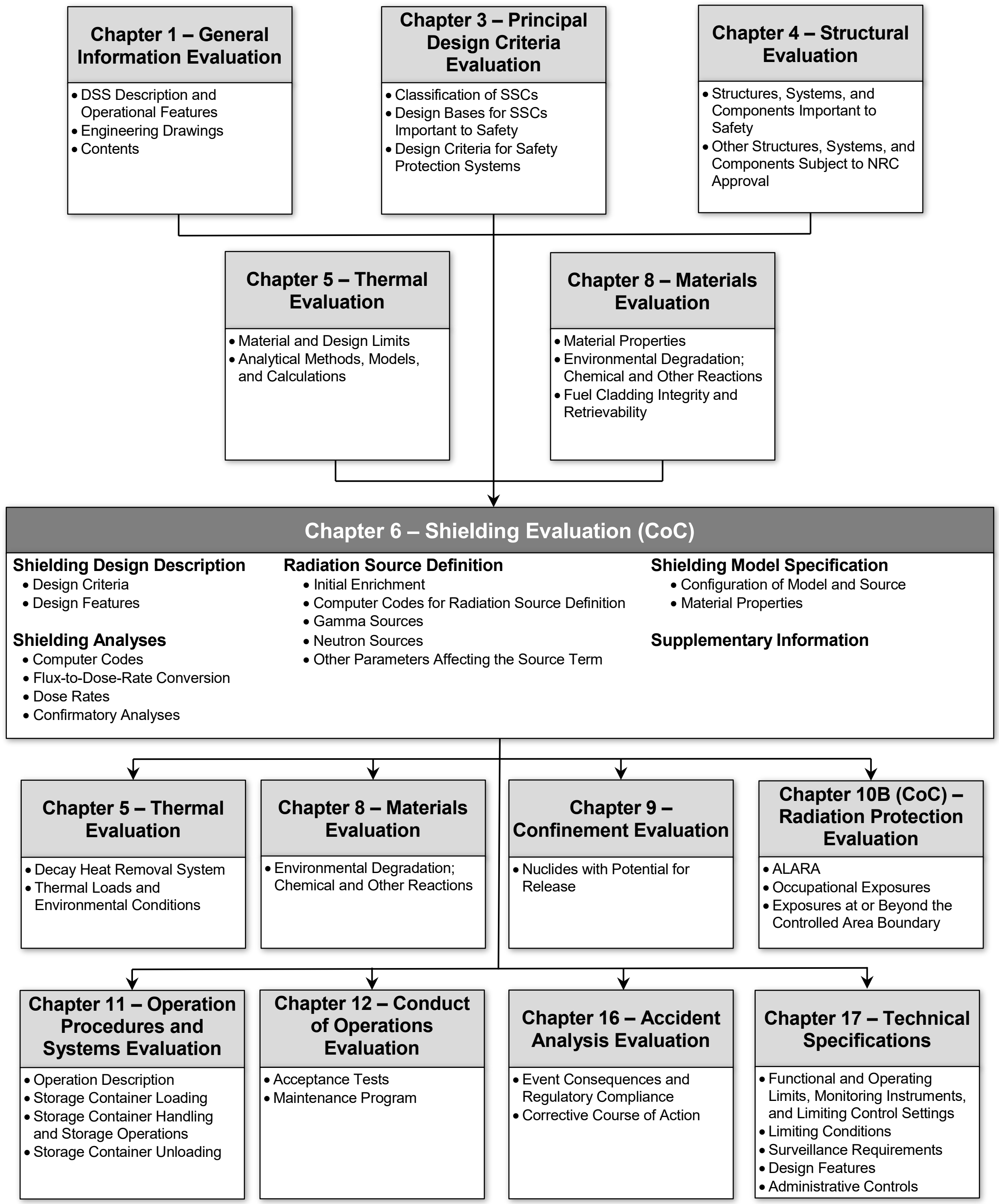

Figure 6-1b Overview of Shielding Evaluation of Applications for a DSS (CoC)

6.5.1 Shielding Design Description

6.5.1.1 *Design Criteria*

Verify that applicant has used specific design criteria and that the SAR describes the criteria as the basis for the shielding design and protection against direct radiation. These criteria may include specification of appropriate maximum dose rates for the variety of storage container (e.g., DSSs) configurations during storage operations for important and relevant container features. For specific license applications, these dose rate criteria may also include DSF SSCs involved in the handling, transfer, and storage of SNF, reactor-related GTCC waste, and HLW. Dose rates at the container surface and in the vicinity of a loaded container may vary during the different stages of storage operations (i.e., loading and unloading, activities to prepare for storage or unloading such as canister welding, canister opening, transferring to and from the storage pad, and activities conducted while the container is at the DSF storage pad).

While 10 CFR Part 72 establishes dose limits for DSFs, it does not impose specific dose rate limits on individual storage containers. The NRC has accepted SNF DSS (cask or storage overpack) storage surface dose rates from 20 to 450 millirem per hour in evaluations for previous CoC applications. For canister-based DSSs, these dose rates apply only to the storage overpack. Surface dose rates for transfer casks for these DSSs are noticeably higher. The surface dose rates for the majority of the transfer casks have not exceeded 2 rem per hour. Some instances with higher transfer cask dose rates have been accepted with technical specifications or conditions in the CoC in addition to the dose rate limit conditions, which are usually established for transfer casks. Coordinate with the radiation protection reviewer (for example, see Chapter 10A, Section 10A.5.2.3, or Chapter 10B, Section 10B.5.1, of this SRP) and technical specifications reviewer (Chapter 17 of this SRP) to determine the necessary license or CoC conditions and technical specifications for the DSS or DSF storage containers, including both the storage overpack and the transfer cask for canister-based designs.

Acceptable dose rates depend on a number of factors, including both the transfer cask and storage overpack for canister-based storage container designs. These factors include (1) the geometry of the storage array, (2) the time workers will routinely spend in the storage array for activities such as monitoring or maintenance, (3) the proximity to other areas frequently occupied by workers, (4) the proximity to the controlled area boundary or other public access areas, (5) the need for unique operation techniques (e.g., remote operations using remote optical systems to perform actions), (6) recovery from off-normal events requiring actions and proximity to SSCs significantly different from normal operations, and (7) limitations or other requirements imposed in the technical specifications for operations with the storage container design. At least some of these factors are specific to individual licensee sites and so are most directly applicable to specific license applications. However, for CoC applications, consider reasonable expectations and estimates for these factors and the implications for a licensee's ability to meet regulatory dose limits as appropriate in determining the acceptability of the storage containers' dose rates. This includes the dose rates for both the transfer cask and the storage overpack of canister-based designs for the different operations configurations for normal, off-normal and accident conditions. Coordinate with the radiation protection reviewer (see Chapter 10A of this SRP for specific license applications and Chapter 10B of this SRP for CoC applications) to evaluate the acceptability of the dose rates.

Coordinate with the reviewer of Chapter 3, "Principal Design Criteria Evaluation," of this SRP and review any additional shielding-related criteria. Refer to Chapter 11, "Operation Procedures and Systems Evaluation," of this SRP to consider any expected operating procedures that would require being close to the storage container, such as equipment that should be monitored or serviced frequently. Also, review the evaluated dose rates at the side of the same storage container to ensure that ALARA principles are either engineered into the design or evoked by specific operating procedures in the chapter of the SAR on operating procedures.

6.5.1.2 *Design Features*

Read the general description of the DSS or DSF presented in the general description chapter of the SAR, as well as any additional information provided in the shielding evaluation chapter. Review the text descriptions as well as the drawings, figures, and tables that describe the DSS or DSF SSCs and features that are relied on for shielding, or for which dose rates should be calculated, to confirm they are sufficiently detailed to allow the staff to perform an indepth evaluation. This includes any unique features or SSCs that are not commonly associated with DSS or DSF design, such as additional, or supplemental, shielding items that are necessary to enable personnel to perform some storage operations (i.e., are necessary beyond just ALARA). Confirm that the descriptions and drawings clearly identify the geometric arrangements of DSS or DSF SSCs and features and physical dimensions. Confirm that the SAR describes the differences in the configuration of the DSS or DSF SSCs and features for normal, off-normal, and accident conditions. Ensure that the information in the SAR addresses the various stages of operations for the identified conditions for all proposed contents (i.e., including any reactor-related GTCC waste and HLW to be stored at the DSF for specific licenses). For SSCs and features for which scenarios may exist that remove or expose material relied on for shielding that otherwise remains in place or unexposed (e.g., excavation near loaded storage containers that rely on the surrounding soil for shielding), ensure that the SAR addresses the effects of the conditions during such scenarios.[2]

(SL) Assess whether the SAR adequately describes the spatial relationship between sources, shielding, and the design dose rate area(s). Consider that the design of shielding can be oriented either on the radiation sources or a point to be protected. The layout of an ISFSI or an MRS typically creates the potential for direct radiation exposure of the offsite population in all directions. As a result, shielding is typically oriented on the sources, which is the most effective positioning of shielding.

Review the SAR material composition descriptions of SSCs and features relied on for shielding. Ensure that the descriptions identify and describe all materials taken into consideration in determining shielding requirements. These include the following:

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[2] Note that design features descriptions are important for ensuring compliance with regulatory dose limits, including the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b). For canister-based storage container designs, these limits apply to the loaded transfer cask as well. The limits apply regardless of the storage container's location (whether in a structure under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," or on the DSF storage pad). This position is consistent with the November 16, 2006, rulemaking's definition of the boundary between 10 CFR Part 50 and 10 CFR Part 72 for criticality safety (see Volume 71 of the Federal Register, page 66648 (71 FR 66648)). Also note that for CoC applications, 10 CFR 72.236(d) places responsibility for designing a DSS to meet 10 CFR 72.104 and 10 CFR 72.106 with the DSS designer (the CoC applicant) and does not contemplate that this requirement regarding the DSS design can be passed to the general licensee through its 10 CFR 72.212 evaluation or 10 CFR Part 50 programs.

- materials that have other functions but their mass also provides shielding (especially gamma shielding by structural materials, gamma and neutron shielding by concrete and pool water, and building and barrier materials for DSFs)

- materials especially selected and positioned for gamma shielding, such as lead

- materials especially selected and positioned for neutron shielding, such as water, concrete, and proprietary shielding materials

Confirm that the material specifications for nonstandard materials (e.g., proprietary neutron shield materials) include appropriate references for the material's properties that are relevant to and are included in the shielding analysis. Consult with the materials evaluation and thermal evaluation reviewers (Chapters 8 and 5, respectively, of this SRP) to identify and understand the material specifications for nonstandard materials in the design. Confirm that the technical design (or engineering) drawings include material specifications important to the performance of the shield materials such as those identified in Section 6.4.1.2 of this SRP chapter, and consider whether any of these specifications should be included in the CoC or license technical specifications.

Also consult with the materials and thermal reviewers to identify and understand the impacts of normal, off-normal, and accident conditions on the properties and behavior of the DSS or DSF SSCs and features relevant to the shielding evaluation for the different stages of operations. These properties and behavior include temperature sensitivities to elevated temperatures, which may cause reduced neutron shield efficacy from the loss of bound or free water in concrete or other hydrogenous shielding materials, as well as impacts of accumulated radiation exposure. Coordinate with the materials reviewer to obtain reasonable assurance that any degradation that may occur will not impact the safe performance of the shielding materials for the term proposed in the CoC or specific license application. Confirm that the SAR includes appropriate tests with adequate acceptance criteria to ensure that components such as lead gamma shielding and neutron shielding are fabricated correctly and perform as designed and will maintain their performance for the proposed CoC or license storage term.

As part of the DSS or DSF shielding design review, also consider the items identified in Section 6.4.1.2 above, as applicable. ANSI/ANS 6.4.2, "Specification for Radiation Shielding Materials," includes information that may be useful to consider as part of this review.

6.5.2 Radiation Source Definition

Verify that all potential radiation sources have been correctly identified and quantified, even if analysis shows that they produce negligible contributions to dose.

Burnup, cooling time, initial uranium loading, and initial enrichment are parameters that affect the total source term of SNF. Examine the description of the design-basis fuel in the chapter of the SAR on principal design criteria to verify that the applicant calculated the bounding source term. Confirm that the applicant examined all designs and burnup conditions for the SNF to be stored in the DSS or at the DSF to ensure that the bounding fuel type and parameter values are used. Devote particular attention to the combined effects of gamma and neutron source terms as a function of fuel burnup, cooling times, and enrichment. In many cases, there is no single specific enrichment-burnup-cooling time combination that bounds all potential storage container loadings (see the analysis presented in NUREG/CR-6716, "Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks," issued March 2001).

Ensure that the SAR specifies cooling times and enrichments as minimum values and burnups as maximum values. For enrichments and burnups, it is acceptable for the values to be assembly average minimum and assembly average maximum values, 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. This in turn results in higher gamma and particularly neutron sources. However, the impact is insignificant for natural uranium blankets shorter than 15 centimeters (6 inches). Variations in fuel assembly type play a secondary role for pressurized-water reactor (PWR) fuel. For boiling-water reactor (BWR) fuel, void fractions and channel sizes may affect the strengths of neutron and gamma sources. Ensure that the SAR describes the condition of the SNF contents (e.g., undamaged, damaged); this information plays a role in determining the adequacy of the analysis models' representation of the physical distribution of the radiation source (see Section 6.5.3.1 below).

Pay special attention to proposed SNF contents that include MOX or thoria. Ensure that the source terms calculated for this kind of SNF properly account for unique aspects of these fuel materials, including nuclides and nuclide quantities from fuel irradiation and from natural decay of the fuel materials. For short post-irradiation cooling times, consider whether or not the source term analysis needs to account for longer times because of possible dose rate increases with time that may result from buildup of nuclides with significant radiations (e.g., Tl-208 in thoria-bearing fuel) at times longer than the analyzed post-irradiation cooling times.

NFH can contribute significantly to SNF storage container dose rates, either locally or overall depending on irradiation history and any limits regarding allowable numbers and locations within the storage containers. Thus, for storage containers that include NFH with the SNF assemblies, ensure that the SAR properly identifies the types of NFH to be stored with the assemblies. Ensure that the SAR includes the parameters needed to determine the source terms for the NFH. These parameters include the burnup (or irradiation exposure); cooling time; component materials, masses, and cobalt impurity levels in different axial zones; the neutron flux factors in the different axial zones; and neutron source types and source strengths (if NSAs are included). Be aware that some NFH may have other materials that, when activated, also can be significant sources (e.g., hafnium, silver-indium-cadmium). Ensure that the SAR addresses these materials and the NFH containing them. Also, some NFH types may have multiple configurations that can affect material amounts in different axial zones. For example, thimble plug devices may also have water displacement or absorber rods. Ensure that the NFH descriptions in the SAR appropriately account for these variations. Ensure that the design-basis NFH source term is based on a saturation value for activation of cobalt impurities or on cobalt activation from a specified maximum burnup and minimum cool time. Consider other activation products, as appropriate, as noted previously. Review the source term from the assembly hardware (e.g., upper and lower nozzles) following applicable guidance for source terms from activated NFH.

(SL) In addition to the SNF and NFH, other radiation sources at a specific license DSF may include the following:

- solid reactor-related GTCC waste
- HLW in a form ready to be stored and other activated materials to be stored with the HLW

(SL) Verify that the SAR provides the physical and chemical form, source geometry, radionuclide content, and estimated curie value and bases for estimation for each source type (i.e., the reactor-related GTCC waste, HLW, and other radioactive material referred to above). The radionuclide inventory and quantity of that inventory in each shielded container define the gamma and neutron sources for that material. The other properties of the material will be useful in defining the distribution of the radionuclide inventory within the material and how that could change under different conditions (e.g., normal vs. accident condition configurations).

Verify that the shielding analysis in the SAR uses parameter values that bound the parameter values that define the allowable SNF and NFH contents and, for specific licenses, the reactor-related GTCC waste and HLW contents in the technical specifications. The technical specification parameters for defining the SNF allowed for storage in the DSS or at the DSF should include the combination(s) of maximum burnup, minimum enrichment, and minimum cooling time that is bounded by the parameters used to define the source terms in the shielding analysis. If the applicant proposes technical specifications that limit the SNF in other ways (e.g., by decay heat only), verify the applicant's justification for that approach and that the radiation source terms used in the shielding analysis are bounding for the variety of SNF that meets the proposed technical specification limit. Verify that the applicant's justification and analysis account for the effects of uncertainty in the methods for determining a SNF assembly meets the limits on the radiation source terms (and thus the dose rates). For example, while decay heat and radiation source terms relate to each other, the relationship is such that a significant variety of burnup, enrichment, and cooling time combinations can result in a given amount of decay heat. Further, the combinations resulting in the same decay heat can vary among types and designs of fuel assemblies. Also, for a DSS, licensees using the DSS may determine their assemblies' decay heat using a different method than the applicant used, which is a source of uncertainty for the radiation source term that should be addressed. Appropriate definition of and evaluation of the source terms for the allowable contents of a DSS or DSF is an important part of the analyses for demonstrating compliance with regulatory requirements, which for a DSS includes, as discussed elsewhere in this SRP chapter, meeting the requirements in 10 CFR 72.236(d).

6.5.2.1 *Initial Enrichment*

The specifications in the chapter of the SAR on principal design criteria should indicate the maximum fuel enrichment used in the criticality analysis. For shielding evaluations, however, the neutron source term increases considerably with lower initial enrichment for a given burnup. As described in Section 3.4.1.2, "Enrichment," of NUREG/CR-6716, as the initial enrichment decreases, the fuel is exposed to a larger neutron fluence to achieve the same burnup. The larger neutron fluence generates a larger actinide content, which results in a larger neutron source term and secondary gamma source term, as illustrated in NUREG/CR-6716, Section 3.4.1.2. Therefore, confirm that the SAR specifies the minimum initial enrichment as one of the parameter limits for the SNF contents, or justifies the use of a neutron source term, in the shielding analysis, that specifically bounds the neutron sources for fuel assemblies to be placed in the storage containers, both in total source strength and strength across the energy spectrum. Because average initial enrichments typically increase with increasing burnup within the SNF population, the latter option may be used if the applicant uses low enrichments that bound the historical enrichments for fuels at the proposed burnups. However, do not attempt to use specific source terms as the bases for establishing SNF contents limits because these are not readily inspectable parameters. The fuel assembly minimum initial enrichment, maximum burnup, and minimum cooling time are more appropriate for use as loading controls and limits.

6.5.2.2 *Computer Codes for Radiation Source Definition*

Verify that the applicant determines the source terms using a computer code, such as ORIGEN-S (e.g., as a SAS2 sequence of Oak Ridge National Laboratory's "SCALE" computer code package), that is well benchmarked and recognized and widely used by the 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 and the Department of Energy, Office of Civilian Radioactive Waste Management, Characteristics Database 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 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, as described previously in this SRP.

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 the SAR includes verification that the chosen cross section library is appropriate for the fuel specifications being considered. Many libraries are not appropriate for a burnup exceeding 45,000 MWd/MTU because validation data are limited at high burnups.

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. As part of this determination, consider the factors described in Section 6.4.4.1 of this SRP chapter. For example, adjustments to source term values or calculation bases or other aspects of the shielding analysis may be necessary to compensate for uncertainties in the source-term calculations for fuel with high burnups. An acceptable approach to address calculation errors and uncertainties is to establish a bounding value(s) with justified conservatisms.

When reviewing the source term calculations, also consider the factor 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 is limited at high burnups. Several NRC-sponsored studies (i.e., ORNL/TM-13315, "Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel"; ORNL/TM-13317, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel"; 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; NUREG/CR-6798, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor," issued January 2003) provide additional information on high-burnup source-term issues.

Coordinate with the thermal reviewer to determine the need to evaluate the applicant's calculation of 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. Regulatory Guide 3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage

Installation," describes a few such methods. Verify that the SAR adequately describes the calculation method and that the method is appropriate for and correctly used to determine the decay heat for the radioactive contents to be stored in the DSS or DSF. Ensure that the analysis also appropriately identifies and accounts for uncertainties in the decay heat analysis.

6.5.2.3 *Gamma Sources*

Verify that the applicant specified gamma source terms as a function of energy for both the SNF and activated hardware (both assembly hardware and NFH), and, for DSF license applications, any reactor-related GTCC waste and HLW to be stored at the facility. 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 0.8 to 2.5 MeV will contribute significantly to the dose rate through typical types of DSS shielding; thus, regrouping outside this range is of a lesser importance for DSSs. Consider the importance of other gamma energies to dose rates for storage containers with shielding that differs from the typical DSS shielding and, for DSFs, for shielding for other SSCs for which dose rates should be calculated. Determine whether the source terms are specified per assembly, per total assemblies, per metric ton, or, for specific licenses, on some appropriate basis for any reactor-related GTCC waste and HLW. Ensure that the total source is correctly used in the shielding evaluation.

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. 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) in which the materials are irradiated. Thus, verify that the SAR identifies the materials that comprise the assembly hardware and NFH to be stored with the assemblies.

Verify that the SAR describes the masses of the materials that are located within each assembly axial zone. Ensure that the SAR 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 SAR 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 SAR, which should be consistent with or reasonably bounding for operations practices for those items.

Ensure that the SAR 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," show that, for at least some assembly types fabricated before that time, cobalt levels may be as high as 1500 ppm in Inconel and 2100 ppm in steel. Thus, ensure that the SAR analysis uses cobalt impurity levels that are appropriate for the fuel assemblies and NFH to be stored in the DSS or DSF storage containers, given the age of the assemblies and NFH (based on their burnups and cooling times).

The nature of the flux changes in magnitude and spectrum in regions outside of the fuel region. Thus, ensure that the SAR 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 described in Section 3.3.2, "Hardware Regional Activation" of NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages," issued May 2003. 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 SAR 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 dose rates 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 dose rates presented in the shielding evaluation. Ensure that the source term analysis addresses all appropriate NFH items that are included in the proposed DSS or DSF contents, comparing the items identified in the source term analysis with those items listed in the DSS or DSF contents descriptions in the appropriate SAR chapters.

Depending on the storage container design(s), neutron interactions may result in the production of high-energy gammas near the container surface. If this source term is not treated by the shielding analysis computer code, verify that it is determined and its contribution to dose rates is addressed by other appropriate means.

Support the confinement review, as needed, by verifying the quantities of certain nuclides (e.g., krypton-85, tritium, and iodine-129) the applicant used to analyze doses from the release of radioactive material during design-basis conditions (i.e., normal, off-normal, and accident conditions). Confer with the confinement reviewer to determine the need to verify these nuclide quantities.

6.5.2.4 *Neutron Sources*

Verify that the SAR expresses the neutron source term as a function of energy. 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 may be a necessary basis for defining the allowed contents. Verify that the assumed minimum enrichments bound all assemblies the application proposes for storage. Specific limits are needed for inclusion in the CoC or license, as applicable. Lower-enriched fuel, irradiated to the same burnup as higher-enriched fuel, produces a higher neutron source. Therefore, verify that the SAR chapter on technical specifications and operational controls and limits specifies the minimum initial enrichment as an operating control and limit. Alternatively, ensure that the applicant specifically justified the use of a neutron source term, in the shielding analysis, that bounds the neutron sources for the SNF assemblies to be stored. An applicant may demonstrate that the assumed enrichment(s) bounds the proposed fuel population except for possible outliers in the SNF population. This is acceptable if the SAR specifically requires verification of the minimum enrichment with the values in the final SAR, and if there are specific dose rate limits in the technical specifications. The applicant and the NRC staff should not attempt to establish specific source terms as the operating controls and limits for SNF storage container (e.g., DSS) use.

Ensure that the SAR adequately describes the neutron source, both source strength and spectrum, for NSAs included in the NFH to be stored with the spent fuel 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 spent fuel assembly. It is these sources that can contribute significantly to the neutron source term in the SNF storage container and so should be included in the shielding evaluation. Secondary sources include antimony-beryllium (SbBe) and others 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 storage container.

Ensure that the SAR adequately addresses contributions to the neutron source from subcritical multiplication since this contribution is not included in the results of depletion codes like SCALE's TRITON and SAS2H or CASMO. 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. The applicant may use such factors when the shielding model properties are such that the model would be critical or near critical (e.g., a flooded SNF container with the SNF modeled as 5-weight-percent enriched fresh fuel). Ensure that the applicant justifies the appropriateness of the selected factor(s).

(SL) The reactor-related GTCC waste and HLW to be stored at the DSF may also include neutron sources, depending on the specification of the wastes. Thus, follow the preceding guidance, as appropriate, when evaluating the neutron source terms for these wastes, considering the criteria given in Section 6.4.2.2 of this SRP chapter.

6.5.2.5 *Other Parameters Affecting the Source Term*

Ensure that the SAR 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, NUREG/CR-6701, and NUREG/CR-6798) discuss the potential effects of other parameters not typically included in technical specifications (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 DSS dose rates 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 dose rate profile of a BWR spent fuel assembly. The dose rate 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.

6.5.3 Shielding Model Specification

Verify that the applicant adequately described the models that were used in the shielding evaluation for storage under normal, off-normal, and accident conditions. For example, if a DSS transfer cask has an external neutron shield, the SAR should reflect whether the cask would be damaged by a tipover accident or by a tornado missile or it would be degraded in a fire. Ensure that the applicant has assumed that liquid, polyesters, or other resin neutron shields are not present after an accident, unless justification is made that they remain intact. Confirm with the structural (SRP Chapter 4), thermal (SRP Chapter 5), and other reviewers, as appropriate, that the treatment of the DSS or DSF features and SSCs in the shielding analysis is consistent with or bounding for the expected operation configurations and the impacts of normal, off-normal, and accident conditions for those operation configurations. Coordinate with these reviewers to ensure the applicant has analyzed the impacts of all appropriate normal, off-normal and accident conditions, including any conditions that may be unique to the DSS or DSF design and operations. Examples include analyses of accident events with excavation adjacent to DSSs that rely on the soil for shielding and dropping onto DSS SSCs (e.g., the transfer cask) of separate shielding devices (also to be considered as part of the DSS design) that are needed to allow personnel to perform some of the DSS operations. Confirm that the shielding assumptions made in dose rate calculations, for both occupational workers and the public, are consistent with the design criteria and design drawings for the DSS or DSF. Ensure that, for DSF license applications, the analysis models address all facility SSCs and features that affect shielding. ANSI/ANS 6.4, "Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," includes information that may be useful to consider as part of the review of the model specifications (this SRP section) and the analysis (Section 6.5.4) for concrete DSS or DSF SSCs.

6.5.3.1 *Configuration of Shielding and Source*

Examine the sketches or figures and descriptions that indicate how the DSS or DSF SSCs and features important, or credited, for shielding are modeled. Ensure that the sketches or figures clearly indicate the geometric arrangement(s) and physical dimensions of the DSS or DSF SSCs and features. Verify that the models are consistent with the DSS or DSF design, including dimensions and materials that are consistent with those specified in the DSS or DSF drawings presented in the general information evaluation chapter of the SAR. Verify that the SAR accounts for voids, streaming paths, and irregular geometries or otherwise treats them in a conservative manner. Verify that the models address the configurations of DSS or DSF features and SSCs during the different operations stages, including any conditions that, though temporary, may affect how different conditions impact the DSS or DSF shielding features (e.g., excavation adjacent to DSSs that rely on soil for shielding) and dose rates for these different conditions. In addition, verify that the applicant clearly stated the differences, if any, between normal, off-normal, and accident conditions.

Verify that the applicant properly modeled the source term locations. For SNF storage containers, this involves properly locating the SNF source within the envelope of the assembly's fuel zone and locating any assembly hardware and NFH sources within the proper assembly zones where this hardware and NFH source may be present. Also, verify that the applicant properly modeled the physical distribution and the material properties of the sources. In many cases, the fuel assembly materials may be homogenized within the fuel region to facilitate the shielding calculations. Watch for cases when homogenization may not be appropriate. For example, homogenization should not be used in neutron dose calculations when significant neutron multiplication can result from moderated neutrons (i.e., when significant amounts of moderating materials are present such as when the SNF container is flooded). In some, particularly early, applications, fuel and basket material homogenization may have been used; however, with improved analytical capabilities, this practice should be discouraged. If homogenization is used, ensure it is not used for configurations where significant streaming could occur between basket components or significant neutron multiplication is expected. Confirm also that the models account for any possible shifts in the position of the contents for different design-basis conditions.

If the applicant has requested storage of damaged fuel assemblies, ensure that the SAR adequately describes the proposed damage assemblies. If the fuel assemblies are damaged to the extent that reconfiguration of the fuel into a geometry different from intact fuel assemblies has occurred (e.g., fuel debris) or can occur, ensure that the SAR provides appropriate materials, geometry, and other necessary parameter specifications to calculate dose rates for normal, off-normal, and accident conditions.

SNF typically has a cosine-shaped burnup profile along its axial length. If axial peaking appears to be significant, verify that the applicant has appropriately accounted for the condition. Typically, fuel gamma source terms vary proportionally with axial burnup. Fuel neutron source terms vary exponentially by a power of 4.12 with burnup (NUREG/CR-6802), which can be applied to the axial variation in burnup. In addition, the structural support regions (e.g., top and bottom end hardware and plenum regions) of the assembly should be correctly positioned relative to the SNF. The materials in these regions may be individually homogenized. Guidance regarding homogenization in the fuel region applies to the assembly hardware regions. Generally, at least three source regions (i.e., fuel and top and bottom assembly hardware) are necessary. Some storage containers may also employ fuel spacers to maintain the axial position of the SNF inside the container.

Verify that the SAR shows or adequately describes the detector locations selected for the various dose rate calculations. Ensure that these locations are representative of all locations relevant to radiation protection issues, including for site personnel and members of the public. Pay particular attention to dose rates from streaming paths to which occupational workers would be exposed (e.g., at vent and drain port covers, lid bolts, air vents). Ensure that the applicant has noted shielding end points (such as lead in the storage container wall in relation to the assembly hardware and use of fuel spacers to center the fuel). Sections 6.4.4.3 and 6.5.4.3 of this chapter provide additional information regarding the selection of detector locations for dose rate calculations for both CoC and specific license analyses.

(SL) Ensure that the models address the design-basis conditions (i.e., normal, off-normal, and accident conditions) for the different stages of storage operations and all the proposed types of contents (any reactor-related GTCC waste and HLW, as well as SNF) in an appropriate or bounding manner. This includes storage container array configurations and maximum quantities of stored materials on the facility’s storage pad(s), contents locations and orientations within containers, and cautions against homogenization of contents with container internals where that is not appropriate. This also includes other DSF SSCs, as appropriate and necessary, in addition to the storage containers.

6.5.3.2 *Material Properties*

Review the descriptions of the materials and their compositions and densities that are used in the models. Verify that the material compositions and densities are consistent with the description of design features and SSCs and the contents given in the SAR as they are geometrically represented in the models. This includes materials for SSCs and features that have other functions but that also provide shielding as well as SSCs and features that are specifically for shielding. Ensure that the materials properties in the models are consistent with or bounding for the effects on the materials properties of the different design-basis conditions (normal, off-normal, accidents). These effects include any degradation from aging, high temperature, accumulated radiation exposure, and manufacturing tolerances. Many shielding computer codes allow the densities to be input directly in grams per cubic centimeter. If densities are input into the models in atoms per barn-centimeter, pay particular attention to the conversion.

6.5.4 Shielding Analyses

6.5.4.1 *Computer Codes*

Evaluate the computer codes or programs used for the shielding analysis. There are several recognized computer codes that are widely used for shielding analysis. These include computer codes that use Monte Carlo, deterministic transport, and point-kernel techniques. The point-kernel technique is generally appropriate only for gammas since storage containers, including DSSs, and SSCs used in operations typically do not contain sufficient hydrogenous material to apply removal cross sections for neutrons.

It is important to assess whether the number of dimensions of the computer code being applied for the shielding analysis is appropriate for the dose rates being calculated. Typically, the NRC does not accept the use of one-dimensional codes for calculations other than shielding designs with simple cylindrical geometries. At the least, a two-dimensional calculation is generally necessary. One-dimensional computer codes provide little information about off-axis locations and streaming paths that may be significant to determining occupational exposure. Even a two-dimensional calculation may not be adequate for determining any streaming paths if the modeled configuration is not properly established. These considerations in applying a particular computer code also apply to the computation of dose rates at the axial ends of storage containers. In some cases, the applicant will use the flux output from a deep-penetration shielding code as input to a large distance, skyshine code. Verify that the use and interface of these codes are appropriate and done correctly.

Ensure that the codes used in the analysis have the capability to account for the effects of radiation interactions that impact dose rates. Also ensure that the applicant has used the codes to appropriately account for the impacts of those interactions. This includes gammas produced by (n, γ) reactions in the DSS or DSF SSCs and contents and subcritical multiplication. For example, for models that may be critical or near critical (e.g., assuming 5 weight percent fresh fuel for the SNF contents in a flooded storage container), use of code features to track fission may not be appropriate to account for subcritical multiplication effects on dose rates. This is because the calculation may not properly converge or finish, except due to any time limits set in the input file. In such a case, the results would not adequately represent the dose rates for the analyzed DSS or DSF SSCs and contents. For this scenario, subcritical multiplication should be addressed in another manner.

The Electric Power Research Institute (EPRI) has published a valuable primer on shielding computer codes and analysis techniques (Broadhead 1995). Computer codes that have commonly been used in CoC and specific license applications include MCNP and SCALE. Codes that have been used or may be useful include the following (grouped by code type):

- Monte Carlo codes: MORSE, MONACO/MAVRIC, MCBEND, SCALE, MCNP
- Discrete Ordinates codes: DORT, ANISN, DANTSYS, DOORS 3.2
- Point Kernel codes: QAD-CGGP, RANKERN

- Others: SKYSHINE-II, STREAMING

The NRC recognizes that there are other codes available that may also be useful for DSS or DSF shielding analyses. These codes come from a variety of sources, including government organizations and commercial vendors. Note that the previous use of these codes (in approved CoC, license, or amendment applications) does not constitute generic NRC approval of these codes.

Regardless of the code(s) used in the SAR analysis, confirm that the applicant has justified the applicability and appropriateness of the particular code(s) for the SAR analysis. The extent of the justification may vary, with codes that are well established, have a broad user base, and have capabilities to handle complex problems needing less justification than a proprietary code or a code that is limited in its capabilities. Confirm that the applicant used a computer code version that is demonstrated to be adequate for the analysis and is valid for the particular computational platform used to perform the analysis. Computer codes are periodically updated to be compatible with the latest operating system, correct errors found in previous versions, or incorporate updated methods. Therefore, consider whether additional confirmatory assessments and review are needed to validate the shielding predictions by an applicant that uses older or unsupported codes or code versions. This consideration should include a recognition that the applicant may use these codes later as the CoC holder or licensee to evaluate changes to the DSS or DSF design or operations under 10 CFR 72.48, “Changes, tests, and experiments,” and the associated implications.

Verify that the SAR describes each of the numerical models of the computer codes used in the shielding evaluation. For each computer code used, ensure that an approved, validated, and verified version of the computer code is being applied by verifying that the SAR provides the following information:

- author, source, and dated version
- description of the numerical model applied in the computer code and the extent and limitation of its application
- either (1) the evaluation of computer code solutions to a series of test problems, demonstrating substantial similarity to solutions obtained from hand calculations, analytical results published in the literature, acceptable experimental tests, a similar computer code, or benchmark problems; or (2) the specification of publicly available references for commonly used and well-established codes (e.g., SCALE and MCNP) that demonstrate validation

Examine the solution comparisons provided in the SAR and determine whether satisfactory agreement of computer and test solutions (or resolution of deviations) is evident. Ideally (though not a requirement), the applicant should have validated the computer code used for evaluation of shielded storage containers with actual dose rate measurements from similar or prototypical SNF or, for specific license applications, GTCC waste or HLW storage containers.

Be aware that 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 dose rates for different source terms by simply 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 dose rates. 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 dose rates for the different stages of operations for the design-basis conditions (i.e., normal, off-normal, and accident conditions) at the locations necessary to evaluate personnel and public doses as discussed in Sections 6.4.4.3 and 6.5.4.3 of this chapter.

6.5.4.2 *Flux-to-Dose-Rate Conversion*

Review the flux-to-dose-rate conversions used in the applicant's shielding analysis and confirm that they are acceptable for the purposes for which the dose rates are used, including demonstration of compliance with regulatory dose limits, estimating occupational doses during operations, and serving as the basis for any dose rate limits in the CoC or license technical specifications, as applicable. The computer code used in the analysis may have data libraries for different conversions and options to perform these conversions automatically or require (or have an option) that conversion factors be manually included in the input file. Whichever option is used, confirm that the SAR clearly identifies the conversion factors used to determine dose rates.

While there are different conversion factors available for use, the NRC has only accepted the use of the ANSI/ANS 6.1.1-1977 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 requirements in 10 CFR Part 72 include two sets of dose limits to individual members of the public located at or beyond the controlled area boundary, annual dose limits for normal operations and anticipated occurrences in 10 CFR 72.104(a), and accident dose limits in 10 CFR 72.106(b).

The limits in 10 CFR 72.106(b) incorporate the methodology of 10 CFR Part 20, which incorporates the methodology from the International Commission on Radiological Protection (ICRP)-26, “Recommendations of the International Commission on Radiological Protection,” and dose calculation methods of ICRP-30, “Limits for Intakes of Radionuclides by Workers.” The limits in 10 CFR 72.104(a) are based on the methodology from ICRP-2, “Report of Committee II on Permissible Dose for Internal Radiation,” to maintain compatibility with the Environmental Protection Agency’s regulation in 40 CFR 191.03(a), which is applicable to 10 CFR Part 72 storage operations (see 63 FR 54559; October 13, 1998).

The ICRP issued a series of ICRP-30 reports that provide the means to derive doses under the dosimetry concept of ICRP-26. The dose calculation methods in the revised 10 CFR Part 20, and relevant for the 10 CFR 72.106(b) limits, do not quantify doses in terms of doses to the whole body and individual, critical organs like is done under the ICRP-2 methodology. Instead, the dose is quantified as a risk-equivalent dose that considers the relative risks of different tissues, expressed as organ or tissue weighting factors (tabulated in 10 CFR 20.1003, “Definitions”). In this manner, doses absorbed by the whole body and individual organs or tissues can be summed into a single quantity relating to risk. This method negates the need to keep track of two sets of doses, one for the whole body and another for a series of organs, as is done under the ICRP-2 methodology.

The conversion factors in the 1977 revision of ANSI/ANS 6.1.1 are derived from methodologies that are consistent with the ICRP-2 and so are appropriate for determining compliance with the limits in 10 CFR 72.104(a). For 10 CFR 72.106(b) limits, though from a different methodology, the conversion factors from the 1977 revision of the standard result in conservative dose rates versus factors derived from the methodology incorporated into 10 CFR 72.106(b) and so are acceptable for evaluating compliance with that requirement.

The 1977 ANSI/ANS 6.1.1 conversion factors are also accepted because they result in dose rates (given as dose-equivalent) that can be readily compared against dose rates measured with appropriate monitoring equipment and techniques for converting instrument readings into meaningful results. The methodology in ICRP-26 introduced dosimetry units of effective dose-equivalent, which is not a measurable quantity, at least without the aid of more sophisticated measurement techniques. Thus, dose rates determined with the 1977 ANSI/ANS 6.1.1 conversion factors are appropriate to use as a basis for dose rate limits in the CoC and license technical specifications, compliance with which is determined by measurement.

While a later revision of ANSI/ANS 6.1.1 (the 1991 revision) was issued, the conversion factors in that revision are based on determination of effective dose-equivalent. Thus, their applicability and usefulness for demonstrating compliance with 10 CFR 72.104(a) limits and for developing dose rate limits in technical specifications carries the concerns of the dosimetry bases identified above. Furthermore, while the 1991 conversion factors were intended to replace the 1977 factors, there were some issues. One basic issue is that in 1985, a recommendation was made in ICRP-45, “Quantitative Bases for Developing a Unified Index of Harm,” to double the neutron quality factors. The 1991 conversion factors, which account for body shielding, have the effect of reducing predicted neutron dose rates by about a factor of two. Had the ICRP-45 recommendation been implemented, dose rates calculated with the 1991 conversion factors and the new quality factors would have been comparable to the dose rates calculated with the current quality factors and the 1977 conversion factors (though, because the calculated dose quantities are different, a direct comparison does not have much meaning). However, the ICRP-45 recommendation was never adopted, given that the standard was later withdrawn. So, calculating dose rates with the 1991 conversion factors would result in predicted neutron dose rates that are reduced by a factor of two. If at some later time the ICRP-45 recommendation were adopted, that could mean issues with compliance with regulatory dose limits and any dose rate limits in CoC or license technical specifications. Thus, there is no regulatory advantage to use the 1991 revision of the standard, and the NRC staff has determined that it should not be used in analyses to demonstrate compliance with regulatory limits or to establish technical specifications dose rate limits.

6.5.4.3 *Dose Rates*

On the basis of experience, comparison to similar systems, or scoping calculations, make an initial assessment of whether the dose rates appear reasonable and whether their variation with location is consistent with the geometry and characteristics of the DSS or DSF contents and design features for the different configurations that exist at different operations stages for the different design-basis conditions. The models used for these calculations should be consistent with the expected condition of the DSS or DSF SSCs and features for the design-basis conditions (normal, off-normal, accident). The following guidance pertains to the selection of points at which the dose rates should be calculated.

For normal and off-normal conditions, ensure that the applicant indicated the dose rate at all locations accessible to occupational personnel during storage container loading, transfer to the DSF storage pad, and maintenance and surveillance operations. Generally, these locations include points at or near various DSS or DSF components and in the immediate vicinity of the storage container and distances from the storage container that are reasonable for the types of activities, including surveillance and maintenance, to be performed during operations, considering the likely locations of personnel involved in the system operations and activities. Examples of locations include inlet and outlet vent areas, trunnion areas, maximum dose rate locations for an SNF storage container's side and top surfaces, the canister-to-transfer cask or overpack (as applicable) gap region, top (including maximum dose rate spot) and upper radial surfaces of the canister, and the bottom of a DSS's transfer cask. Additional examples include locations of changes in shielding such as radial surface locations above and below the axial extent of radial neutron shielding and openings in the transfer cask lid as well as areas on the lid. For rectangular-shaped SSCs such as storage modules and overpacks, ensure that the locations include maximum dose rate spots on each side and on the top. Verify that the applicant calculated the dose rates at a distance of 1 meter (3.28 feet) from these locations because they typically contribute to occupational exposures.

Dose rate analyses should address potential configuration changes of the contents (e.g., reconfiguration of damaged fuel within a damaged-fuel can), if applicable, to support demonstration that the container or fuel (or both) meets the dose limits of normal, off-normal, and accident conditions of storage. The shielding analysis should assume a worst-case or bounding configuration of the contents (e.g., the canned fuel).

Verify that the dose rate estimates have appropriately considered the following:

- conservatism of simplifying assumptions and assertions that non-conservative assumptions are more than compensated for by conservative assumptions

- streaming path dose rates that include failure to offset penetrations in SSCs such as storage container lids for venting, draining, drying

- analyzed configurations consistent with or bounding for anticipated or expected configurations (e.g., water levels in canisters during welding of canister lid or canister decontamination)

- potential negative effects of radiation scattering in DSS or DSF SSCs that increase dose rates in accessible areas near the storage container

- local “hot spots” from gaps or significantly reduced shielding around the source, considering all solid angles

(CoC) Regulations in 10 CFR 72.236(d) require that the application for a DSS design demonstrate that the shielding and confinement features of the DSS are sufficient to meet the requirements in 10 CFR 72.104 and 10 CFR 72.106. Compliance with this part is evaluated as part of the radiation protection review (see Chapter 10B of this SRP).

(CoC) Ensure that the applicant calculated dose rates at appropriate and sufficient distances from the DSS. For 10 CFR 72.104 evaluations, this includes calculations for a single DSS and a sample array(s) of DSSs on a storage pad. The DSS array is typically a 2 x 10 DSS arrangement or some other array that is representative of how the system will or may be used at a DSF. For canister-based systems, ensure the calculations include the transfer cask for 10 CFR 72.106 analyses. Calculations with the transfer cask for 10 CFR 72.104 analyses may also be needed depending on the transfer cask characteristics and operations descriptions. Examples of when such calculations should be provided and evaluated for transfer casks include when dose rates at 100 meters (328 feet) from the cask indicate that transfer cask operations could result in nonnegligible, or possibly significant, doses at that distance for the estimated duration of normal operations or for an anticipated occurrence of reasonably expected time duration (e.g., crane malfunction during cask movement and associated recovery actions).[3] Coordinate with the radiation protection reviewer to determine if these calculations are needed. For both 10 CFR 72.104 and 10 CFR 72.106 analyses, ensure that the applicant calculated the dose rates at distances starting at 100 meters from the DSS and the DSS array. For 10 CFR 72.106 analyses, calculations at 100 meters have typically been sufficient to support demonstration of compliance with the regulatory limit. For 10 CFR 72.104 analyses, dose rates are typically needed at multiple distances, beginning at 100 meters.

(CoC) It is important to note that a general licensee is permitted to use distance or additional engineering features such as berms, or both, to mitigate doses to real individuals near the site. If such features are used in the DSS SAR dose rate calculations, verify that they are included in the descriptions of the DSS and their use is included in the CoC as a condition of DSS use. In addition, verify that the SAR determines the degree to which the normal condition dose rates could change for the identified off-normal conditions.

(SL) In addition to the dose rate location and estimate considerations listed above, ensure that the dose rate locations and estimates include surfaces and appropriate distances from all relevant DSF SSCs involved in the handling, transfer, or storage of radioactive materials to be stored at the site. Also ensure that the dose rate locations and estimates include other relevant site locations where facility personnel and other individuals (e.g., shippers bringing material on site) may be located, and which are needed for the radiation protection evaluation (SRP Chapter 10A) of occupational and public doses. This includes evaluation of situations such as a work station

[3] See Footnote 2 on page 6-18.

that is shielded from multiple sources of radiation. For such situations, check the solid angles about that station for potential gaps or other sources of elevated dose rates.

(SL) Consult with the radiation protection reviewer (SRP Chapter 10A) who will use the dose rate estimates (in addition to other information) to determine whether appropriately detailed SAR calculations (dose rates and collective dose estimates) show that the radiation shielding features are sufficient to meet the requirements in 10 CFR 72.104, 10 CFR 72.106, and ALARA objectives. As noted in Section 6.4.4.3 of this SRP, any supplemental shielding or feature (e.g., berms) included in the calculations to demonstrate compliance with the regulatory dose limits should be classified as important to safety.

6.5.4.4 *Confirmatory Analyses*

Perform confirmatory calculations, as necessary, to verify the results of the applicant's shielding analysis. Independently evaluate the dose rates in the vicinity of the DSS or DSF SSCs and features for normal, off-normal, and accident conditions for the different configurations at the different operations stages. In determining the level of effort appropriate for these calculations, consider the following factors:

- the degree of sophistication in the SAR analysis
- a comparison of SAR dose rates with those of similar DSS or DSF SSCs that have previously been reviewed, if applicable
- the amount of conservatism applied in the applicant's analysis
- the typical variation in dose rates expected between different computer codes and cross-section sets
- the fact that actual dose rates will be monitored and practices employed by the licensee to limit or minimize doses in accordance with the requirements in 10 CFR Part 20
- the restrictions to be placed on the DSS or DSF operations or the limits to be placed on dose rates, as documented in the CoC or license, including any technical specifications
- the applicant's experience in using the methods and computer codes in previous submittals
- the use of computational methods or computer codes that are new or that have been used in previously reviewed CoC or specific license applications
- the inclusion in the design of any significant departures from previous DSS or DSF SSC and feature designs (e.g., unusual shield geometry, new types of materials, or different source terms) or operations

Coordinate with the radiation protection reviewer in determining the need for, and level of effort to expend in, performing confirmatory calculations. At a minimum, examine the representative input files submitted in, or with, the SAR. Verify that the data for the DSS or DSF design features and contents are properly entered into the code, including proper dimensions, material properties, gamma and neutron source terms, and distributions of the sources. Verify that the applicant uses a cross section library that is appropriate for the shielding analysis, including the use of any coupled cross sections in instances where the code is used to evaluate secondary sources through modeling of the radiation interactions in the DSS or DSF shielding materials. Ensure that the applicant correctly uses appropriate code options and features to enable accurate calculations, including for secondary source contributions and neutrons from subcritical multiplication.

If a more detailed review is required (e.g., the applicant used a new shielding computer code not used in a previously approved CoC or license application, the design is unusual, dose rates are significantly high vs. other reviewed DSSs or DSFs), independently confirm the dose rates to ensure that the SAR results are reasonable and conservative. As previously noted, the use of a simple computer code for neutron calculations often does not provide results with sufficient accuracy and confidence. An extensive and more detailed evaluation may be necessary if large uncertainties are suspected. To the degree possible, the use of a different shielding computer code with a different analytical technique and cross-section set from that used in the SAR analysis will usually provide a more independent evaluation.

EPRI has published a good reference (Broadhead 1995) regarding the treatment of uncertainty in thick-shielded cask analyses.

Coordinate with the thermal and confinement reviews 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. Refer to the literature regarding these codes for information about the calculation uncertainties. For example, for SCALE, this information is included in various Oak Ridge National Laboratory technical reports and NUREG/CRs (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”).

(SL) In addition to the preceding guidance, consider the following in determining the appropriate level of effort:

- margin of safety in the SAR analyses
- use of the results in developing projected doses
- magnitude of estimated doses (occupational and for members of the public) under normal, off-normal, and accident conditions, as applicable, considering all radiation sources

6.5.5 Consideration of Reactor-Related GTCC Waste Storage (SL)

(SL) Review the applicant’s approach to addressing storage of solid reactor-related GTCC waste at the DSF, considering the requirements described in Section 6.4.5 of this SRP. Confirm that the applicant clearly describes the analysis approach for the reactor-related GTCC waste and the basis for that approach. Evaluate the acceptability of the approach, considering the contents, SSCs, and design features (including the storage containers), and operations descriptions. Ensure the descriptions in the SAR are adequate for the reactor-related GTCC waste, appropriately applying the guidance in the preceding review sections. For evaluating approaches that use dose rates from SNF or HLW storage to bound and represent reactor-related GTCC waste dose rates, compare the descriptions for reactor-related GTCC waste with the information for the SNF or HLW and confirm that the information supports the adequacy of the approach.

Confirm that the SAR analysis addresses all operations configurations (e.g., loading, container closure, storage at the storage pad) and all design-basis conditions (normal, off-normal, accident). Ensure that the analysis provides dose rate information that can be used in the radiation protection evaluations (see SRP Chapter 10A) to demonstrate facility design and operations meet, or will meet, the requirements in 10 CFR 72.104 and 10 CFR 72.106 and the requirements in 10 CFR Part 20, and that the storage of reactor-related GTCC waste will not have an adverse effect on the safe storage of SNF and HLW.

6.5.6 Supplementary Information

Review supplemental information, which can include copies of applicable references (especially if a reference is not generally available to the reviewer), computer code descriptions, input and output files, and any other information that the applicant deems necessary. Request any additional information needed to complete the review process.

6.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the applicable regulatory requirements in Section 6.4 of this SRP. 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, as separately provided for CoCs and specific licenses:

Certificate of Compliance **(CoC)**

F6.1 The SAR provides specifications of the spent fuel contents to be stored in the [DSS designation] in sufficient detail that adequately defines the allowed contents and allows evaluation of the DSS shielding design for the proposed contents. The SAR includes analyses that are adequately bounding for the radiation source terms associated with the proposed contents' specifications. (10 CFR 72.236(a))

F6.2 The SAR describes the structures, systems, and components (SSCs) important to safety that are relied on for shielding in sufficient detail to allow evaluation of their effectiveness for the proposed term of storage. [The reviewer should cite specific drawings that are used to define the SSCs relied on for shielding.] (10 CFR 72.236(b) and 10 CFR 72.236(g))

F6.3 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of the [DSS designation's] compliance with 10 CFR 72.236(d). This evaluation is described in the radiation protection review (SRP Chapter 10B).

F6.4 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of consideration of ALARA in the [DSS designation's] design and evaluation of occupational doses. This evaluation is described in the radiation protection review (SRP Chapter 10B).

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

> In summary, the staff has reasonable assurance that the design features relied on for shielding for the [DSS designation] have been adequately identified and evaluated. The evaluation includes appropriate shielding analyses for the configurations that exist during the different stages of storage operations, including the impacts of normal, off-normal, and accident conditions. The evaluation includes dose rate results that are adequate to support evaluation of the [DSS designation]'s compliance with the radiation protection requirements in 10 CFR 72.236(d), the occupational doses estimated to result from storage operations using the [DSS designation], and the adequate consideration and incorporation of ALARA principles into the [DSS designation] design and operations. The staff reached this finding on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, the statements and representations in the SAR, and the staff's confirmatory analyses.

<u>Specific License **(SL)**</u>

F6.5 The SAR provides specifications of the radioactive materials to be stored at the proposed DSF in sufficient detail that adequately defines the allowed materials and allows evaluation of the DSF shielding design for the proposed materials. The SAR includes analyses that are adequately bounding for the radiation source terms associated with the proposed materials' specifications. (10 CFR 72.24(c), 10 CFR 72.120(b) and 10 CFR 72.120(c))

F.6.6 The SAR describes the DSF structures, systems, and components (SSCs), including those that are important to safety that are relied on for shielding, in sufficient detail to allow evaluation of their effectiveness for the proposed license term. [The reviewer should cite specific drawings that are used to define the SSCs relied on for shielding.] The descriptions include design criteria and design bases for the design, fabrication, construction, and performance requirements of SSCs important to safety. (10 CFR 72.24(b) and 10 CFR 72.24(c), 10 CFR 72.120(a–c))

F6.7 The DSF design includes SSCs and features to shield personnel from radiation exposure to meet 10 CFR 72.126(a)(6) and for radiation protection under normal and accident conditions to meet 10 CFR 72.128(a)(2). Evaluation of the suitability of the shielding to perform these functions is described in the radiation protection review (SRP Chapter 10A).

F6.8 The SAR provides reasonable and appropriate information, including dose rates, to allow evaluation of the DSF's compliance with 10 CFR 72.24(e). This evaluation is described in the radiation protection review (SRP Chapter 10A).

F6.9 The SAR provides reasonable and appropriate information, including dose rates, to enable performance of the evaluations required in 10 CFR 72.24(m) and to allow evaluation of the DSF's ability to meet the radiation protection requirements for members of the public in

10 CFR 72.104. 10 CFR 72.106 and 10 CFR Part 20. This information includes impacts to shielding and dose rates to support evaluations of compliance with the requirements in 10 CFR 72.122(b)(2)(i), 10 CFR 72.122(c), and 10 CFR 72.122(e) as well. These evaluations are described in the radiation protection review (SRP Chapter 10A).

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

> In summary, the staff has reasonable assurance that the design features relied on for shielding for the DSF have been adequately identified and evaluated. The evaluation includes appropriate shielding analyses for the configurations of DSF SSCs and features that exist during the different stages of storage operations, including the impacts of normal, off-normal, and accident conditions. The evaluation includes dose rate results that are adequate to support evaluation of the DSF's ability to meet the radiation protection requirements in 10 CFR 72.104, 10 CFR 72.106, and 10 CFR Part 20, including doses to members of the public and occupational doses estimated to result from DSF operations, and the adequate consideration and incorporation of ALARA principles into the DSF design and operations. The staff reached this finding on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, the statements and representations in the SAR, and the staff's confirmatory analyses.

6.7 References

10 CFR Part 20, "Standards for Protection Against Radiation."

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

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

40 CFR Part 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes." Subpart A, Environmental Standards for Management and Storage.

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.1.1, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," 1977.

ANSI/ANS 6.1.1, "Neutron and Gamma-Ray Fluence-to-Dose Factors," 1991.

ANSI/ANS 6.4, "Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," 2006.

ANSI/ANS 6.4.2, "Specification for Radiation Shielding Materials," 2006.

Broadhead, B.L. "Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments," EPRI TR-104329, Electric Power Research Institute, Palo Alto, CA, June 1995.

DeHart, M.D. and O.W. Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, Oak Ridge National Laboratory, September 1996.

Hermann, O.W. and M.D. DeHart, “Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel,” ORNL/TM-13315, Oak Ridge National Laboratory, September 1998.

International Commission on Radiological Protection (ICRP) Publication 2, “Report of Committee II on Permissible Dose for Internal Radiation,” 1959.

ICRP Publication 26, “Recommendations of the International Commission on Radiological Protection,” *Annals of the ICRP*, Vol. 1, Issue 3, 1977.

ICRP Publication 30 (Part 1), “Limits for Intakes of Radionuclides by Workers,” *Annals of the ICRP*, Vol 2, Issue 3-4, 1979. (Part 2) Vol 4, Issue 3-4, 1980. (Part 3) Vol 6, Issue 2-3, 1981. (Part 4), Vol 19, Issue 4, 1988. Supplement to Part 1 Vol 3, Issue 1-4, 1979. Supplement to Part 2, Vol 5, Issue 1-6, 1981. Supplement A to Part 3 Vol 7, Issue 1-3, 1982. Supp B to Part 3, Vol 8, Issue 1-3, 1982.

ICRP Publication 45, “Quantitative Bases for Developing a Unified Index of Harm,” *Annals of the ICRP*, Vol. 15, Issue 3, 1985.

Luksic, A. “Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal,” PNL-6906, Pacific Northwest Laboratory, June 1989.

NUREG/CR-5625, “Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data,” ORNL-6698, Oak Ridge National Laboratory, July 1994.

NUREG/CR-6700, “Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel,” ORNL/TM-2000/284, Oak Ridge National Laboratory, January 2001.

NUREG/CR-6701, “Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel,” ORNL/TM-2000/277, Oak Ridge National Laboratory, January 2001.

NUREG/CR-6716, “Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks,” ORNL/TM-2000/385, Oak Ridge National Laboratory, March 2001.

NUREG/CR-6798, “Isotopic Analysis of High-Burnup PWR Spent Fuel Samples from the Takahama-3 Reactor,” ORNL/TM-2001/259, Oak Ridge National Laboratory, January 2003.

NUREG/CR-6802, “Recommendations for Shielding Evaluations for Transport and Storage Packages,” ORNL/TM-2002/31, Oak Ridge National Laboratory, May 2003.

Regulatory Guide 3.54, “Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation.”

U.S. Nuclear Regulatory Commission, “Minor Revision of Design Basis Accident Dose Limits for Independent Spent Fuel Storage and Monitored Retrievable Storage Installations,” *Federal Register*, Vol. 63, No. 197, October 13, 1998, pp. 54559–54562.

NRC, “Criticality Control of Fuel Within Dry Storage Casks or Transportation Packages in a Spent Fuel Pool,” *Federal Register*, Vol. 71, No. 221, November 16, 2006, pp 66648–66657.