5.0 SHIELDING EVALUATION

I. Review Objective

In this portion of the dry cask storage system (DCSS) review, the NRC evaluates the shielding features of the proposed cask system, as designed for an independent spent fuel storage installation (ISFSI). In conducting this review, the NRC staff seeks to ensure that the proposed shielding features provide adequate protection against direct radiation from the cask contents. The shielding features should limit the dose from direct radiation to the operating staff and members of the public, so that the dose remains within regulatory requirements during normal operating, off-normal, and design-basis accident (DBA) conditions.

II. Areas of Review

This chapter of the DCSS Standard Review Plan (SRP) provides guidance for use in evaluating the shielding features of the proposed cask system. As defined in Section V, "Review Procedures," a comprehensive shielding evaluation *may* encompass the following areas of review:

1. shielding design description
    a. design criteria
    b. design features
2. radiation source definition
    a. gamma source
    b. neutron source
3. shielding model specification
    a. configuration of shielding and source
    b. material properties
4. shielding analyses
    a. computer programs
    b. flux-to-dose-rate conversion
    c. dose rates
    d. independent calculations
5. supplementary information

As prescribed in 10 CFR Part 72[1], the regulatory requirements for doses at and beyond the controlled area boundary include both the direct dose and that from an estimated release of radionuclides to the atmosphere (based on the tested leak tightness of the confinement). Consequently, an overall assessment of the compliance of the proposed DCSS with these regulatory limits is contained in Chapter 10, "Radiation Protection," of this SRP.

III. Regulatory Requirements

10 CFR Part 72 requires that spent fuel radioactive waste storage and handling systems be designed with suitable shielding to provide adequate radiation protection under both normal and accident conditions. Consequently, the DCSS application must describe the shielding structures, systems, and components (SSCs) important to safety in sufficient detail to allow the NRC staff to thoroughly evaluate their effectiveness. It is the responsibility of the vendor, the facility owner, and the NRC staff to analyze such SSCs with the objective of assessing the impact of direct radiation doses on public health and safety.

In addition, SSCs important to safety must be designed to withstand the effects of both credible accidents and severe natural phenomena without impairing their capability to perform their safety functions. The applicable shielding requirements are identified, in part, in 10 CFR 72.24(c)(3), 72.24(d), 72.104(a), 72.106(b), 72.122(b), 72.122(c), 72.128(a)(2), and 72.236(d).

IV. Acceptance Criteria

The task of identifying dose rate limits for direct radiation from storage casks is complicated by three considerations. First, 10 CFR Part 72 states regulatory dose limits in terms of total absorbed doses rather than dose rates. Second, dose analyses must include potential sources of radiation other than direct radiation from spent fuel in the cask. Third, the regulatory requirements (listed below) for acceptable cask use at an ISFSI are site-specific and must be separately evaluated on a case-by-case basis. That is, these evaluations are performed as required for a site-specific license application or as required by 10 CFR 72.212 for a utility using a cask under the general license.

In general, the DCSS shielding evaluation seeks to ensure that the proposed design fulfills the following acceptance criteria:

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The “controlled area” is defined in 10 CFR 72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.

2. The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed DCSS are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.

3. Dose rates from the cask must be consistent with a well-established “as low as reasonably achievable” (ALARA) program for activities in and around the storage site.

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ.

5. The proposed shielding features must ensure that the DCSS meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20[2], Subparts C and D.

V. Review Procedures

1. Shielding Design Description

a. Design Criteria

Dose rates at the cask surface and in the vicinity of a loaded cask may vary during storage, transfer, and in-storage activities. Dose rates are defined as the total expected exposure to workers during ISFSI-related operations, or to members of the public who are assumed to be at the closest boundary of the controlled area (at least 100 meters from the storage cask). The NRC has accepted a range of surface dose rates, but doses calculated for workers and the public must comply with the criteria in 10 CFR Parts 20 and 72.

10 CFR Part 72 does not establish specific cask dose rate limits. Cask dose rates from 20 to 400 mrem/hour have been accepted in previous Part 72 evaluations. Acceptable dose rates depend on a number of factors, such as the geometry of the storage array, the time workers will routinely spend in the storage array for activities like monitoring or maintenance, the proximity to other areas frequently occupied by workers, and the proximity to the controlled area boundary or other public access areas.

Review the design criteria presented in Section 2 of the applicant’s safety analysis report (SAR), as well as any additional shielding-related criteria. Consider the proximity of the storage array to equipment that must be monitored or serviced frequently. Note that *high dose rates at the cask top or bottom may be acceptable if these areas are not routinely occupied during storage operations and if the expected exposure during cask transfer operations is controlled*. However, remember to evaluate the dose rates at the side of the same cask to ensure that ALARA principles are either engineered into the design or evoked by specific operating procedures.

b. Design Features

Review the general description of the cask presented in SAR Section 1, as well as any additional information provided in SAR Section 5. All drawings, figures, and tables describing shielding features must be sufficiently detailed to support an in-depth staff evaluation.

2. Radiation Source Definition

Examine the description of the design-basis fuel in SAR Section 2 to verify that the applicant calculated the source term on the basis of the fuel that will actually provide the bounding source. The SAR should examine all fuel designs and burnup conditions for which the cask system is to be certified to ensure that the bounding fuel type and values are used. The applicant should devote particular attention to the enrichment, burnup, and cooling times. Generally, the specifications in SAR Section 2 will indicate the maximum fuel enrichment used in the criticality analysis. For shielding evaluations, however, the neutron source term increases considerably with decreasing initial enrichment and constant burnup. Consequently, the SAR may either specify the minimum initial enrichment or establish the specific source terms as operating controls and limits for cask use.

Generally, the applicant will determine the source terms using ORIGEN-S[3] (e.g., as a SAS2 sequence of SCALE), ORIGEN2[4], or the U.S. Department of Energy (DOE) Characteristics Data Base[5]. Although the latter two are easy to use, both have energy group structure limitations, as discussed below. If the applicant has used ORIGEN2, verify that the chosen cross-section library is appropriate for the fuel being considered. Many libraries are not appropriate for a burnup that exceeds 33,000 MWd/MTU.

a. Gamma Source

Verify that the applicant specified gamma source terms as a function of energy for both the spent fuel and activated hardware. 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 code with a variable group structure. Some applicants will merely interpolate from one structure to the other. 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 shielding; thus, regrouping outside this range is of little consequence. Pay attention to whether the source terms are specified per assembly, per total assemblies, or per metric ton, and ensure that the total source is correctly used in the shielding evaluation.

Determining source terms for fuel assembly hardware is generally not as straightforward as for the spent fuel, especially if the applicant uses one of the ORIGEN codes. The effort devoted to reviewing this calculation should be appropriate to the contribution of these terms to the dose rates presented in the shielding evaluation. Also, note whether the cask is intended to contain special hardware, such as control assemblies or shrouds, and ensure that source terms from these components are included if applicable.

Depending on the cask design, neutron interactions may result in the production of energetic gammas being produced near the cask surface. If this source term was not treated by the shielding analysis code, verify that it has been determined by other appropriate means.

As part of the source term determination, the applicant must calculate the quantities of certain nuclides (e.g., Kr^{85}, H^3, and I^{129}) for use in analyzing doses from the release of radioactive material during postulated accidents in later sections of the SAR. If these calculations are presented in the shielding evaluation section, they should be reviewed at this time. Often, the applicant will tabulate all nuclides that are important to the direct radiation dose rate. This information can be used to resolve differences that may exist between the source terms derived by the applicant and those derived by the NRC staff reviewer.

b. Neutron Source

Verify that the neutron source term is expressed as a function of energy. The neutron source will generally result from both spontaneous fission and alpha-n reactions in the fuel. Depending on the method used to determine these source terms, the applicant may need to independently determine the energy group structure. This analysis is often accomplished by selecting the nuclide with the largest contribution to spontaneous fission (e.g., Cm^{244}) and using that spectrum for all neutrons, since the contribution from alpha-neutron reactions is generally small.

3. Shielding Model Specification

Verify that the SAR adequately describes the models that the applicant used in the shielding evaluation for storage under both normal and accident conditions. For example, if the cask has an external neutron shield, determine whether it would be damaged by a tipover accident or degraded in a fire. Coordinate this analysis with the structural and thermal reviews, as appropriate. 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.

a. Configuration of the Shielding and Source

Examine the sketches or figures that indicate how the shielding is modeled. Verify that the model dimensions and materials are consistent with those specified in the cask drawings presented in SAR Section 1. Ensure that voids, streaming paths, and irregular geometries are accounted for or otherwise treated in a conservative manner. In addition, the SAR must clearly state the differences, if any, between normal storage conditions and accident conditions.

Verify that the applicant properly modeled the source term locations for both spent fuel and structural support regions. In some cases, the fuel and basket 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). Similarly, homogenization should not be used in configurations where significant radiation streaming can occur between the basket components.

Because of a cosine burnup profile, a uniform source distribution is generally conservative for the top and bottom, but not for the axial center. If axial peaking appears to be significant, verify that the applicant has appropriately accounted for the condition. In addition, the structural support regions (e.g., top and bottom end pieces and plenum) of the assembly should be correctly positioned relative to the spent fuel. These support regions may be individually homogenized with the basket materials. Generally, however, at least three source regions (i.e., fuel and top/bottom assembly hardware) are necessary.

Verify that the SAR shows or adequately describes the locations selected for the various dose calculations. Ensure that these dose points are representative of all locations relevant to radiation protection issues. Pay particular attention to dose rates from streaming paths to which occupational workers would be exposed (e.g., at vent/drain port covers, lid bolts, etc.). Also, note the shielding end points, such as lead in the cask wall, in relation to the assembly hardware. See Section IV.4.c, below, for additional information regarding the selection of locations for dose calculations.

b. Material Properties

Verify that the SAR provides information concerning compositions and densities for all materials used in the calculational model. For nonstandard materials (such as neutron shields), SAR Section 9 must also reference the source of the data and indicate validation criteria. Many shielding codes allow the densities to be input directly in g/cm^3. If input is required in atoms/barn-cm, pay particular attention to the conversion.

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

4. Shielding Analyses

a. Computer Programs

Examine the computer program(s) used for the shielding analysis. These codes may include Monte Carlo, deterministic transport, or point-kernel techniques for problem solution. Some shielding codes available from the Radiation Safety Information Center[a] are listed below:

- TORT\DORT (three- and two-dimensional discrete-ordinate neutron/photon transport codes)

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[a] Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, Tennessee, 37831-6362.

- ONEDANT/TWODANT (one- and two-dimensional multigroup discrete-ordinate transport codes)
- MCNP (Monte Carlo n-particle transport code)
- ANISN (one-dimensional neutron attenuation code)
- SKYSHINE (air-scattering code)
- MORSE (Monte Carlo multigroup three-dimensional neutron and gamma transport computer code)
- QAD-CGGP (three-dimensional point kernel gamma transport shielding computer code)
- SCALE (a modular code system for performing standardized computer analyses for licensing evaluation)

The above computer programs are recognized and widely known in shielding analysis. However, their use does not constitute generic NRC approval, and the reviewer is cautioned that these codes can produce errors when used incorrectly. The applicant should have design control measures for ensuring the quality of computer programs.

A valuable primer on shielding codes and analysis techniques has been published by Oak Ridge National Laboratory[6].

For each program, verify the following information to demonstrate its applicability and validity:

i. The author, source, dated version, and facility.

ii. A description, and the extent and limitation of its application.

iii. The computer program solutions to a series of test problems, demonstrating substantial similarity to solutions obtained from any one of the following sources:
(a) hand calculations
(b) analytical results published in the literature
(c) acceptable experimental tests
(d) a similar program
(e) benchmark problems

In addition, verify that the applicant has prepared a summary comparison of the solutions obtained using sources (a) through (e), in either graphical or numeric form. These solutions may be referenced, and need not be submitted in the SAR, provided that the references are widely publicly available or have previously been submitted to the NRC and the information submitted under items I and ii remains unchanged.

Review the submitted computer solutions to the test problems, and compare them with the test solutions. Satisfactory agreement of computer and test solutions and/or resolution of deviations provides verification of the quality and adequacy of the computer programs to perform the calculations for which they were designed. Identify any deviations that have not previously been justified to the staff's satisfaction, and transmit the finding to the applicant with a request for additional technical justification regarding application of the code.

Determine whether the number of dimensions of the code is appropriate for the dose rates being calculated. Generally, at least a two-dimensional calculation is necessary. One-dimensional codes provide little information about off-axis locations and streaming paths that may be significant for determining occupational exposure. Even for dose rates at the end of the cask, one-dimensional codes require a buckling correction that is not particularly straightforward, since merely using the cask cavity diameter may underestimate the dose rate.

The SAR should include a representative computer code input file. As discussed in Section V.4.d below, if the reviewer is familiar with the code used in the SAR analysis, examining the input file can significantly expedite the review.

Verify that the information from the shielding model is properly entered into the code. Also, verify that the cross-section library used by the code is appropriate for use in analysis of cask shielding problems. If the applicant has not independently determined a source term for neutron-induced gamma radiation or subcritical multiplication of neutrons, ensure that a coupled cross-section set was used and that the applicant executed the code in a manner that accounts for these secondary source terms.

b. Flux-to-Dose-Rate Conversion

The shielding analysis code may perform flux-to-dose-rate conversion using its own data library. For the conversions, the NRC accepts the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 6.1.1-1977[7]. The 10 CFR Part 20 radiation protection requirements are based on fluence-to-dose conversions that are essentially the same as those defined by ANSI/ANS 6.1.1-1977, and are conservative relative to those of ANSI/ANS 6.1.1-1991. Neutron dose rates determined on the basis of conversions performed according to ANSI/ANS 6.1.1-1991 may be significantly lower than those determined on the basis of 10 CFR Part 20 or ANSI/ANS 6.1.1-1977.

c. 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 shielding characteristics of the cask system. The following guidance pertains to the selection of points at which the dose rates should be calculated.

For normal conditions, the SAR should indicate the dose rate at all locations accessible to occupational personnel during cask loading, transport to the ISFSI, and maintenance and surveillance operations. Generally these locations include points at or near various cask components and in the immediate vicinity of the cask.

Appropriately detailed calculations are necessary to show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.104 and 72.106. These calculations will need to assume a typical storage arrangement for the casks. Later, when a particular site is selected, calculations will be needed to show ultimate compliance of the spent fuel system. Site configurations not enveloped by the typical site layout assumed in the SAR must be treated in the written evaluations required under 72.212(b)(2) and (3) before the cask system is used. In addition, the SAR should determine the degree to which the normal condition dose rates could change for the identified off-normal conditions. The need for additional calculations should be indicated in the SER and in the conditions set forth in the Certificate of Compliance.

The NRC has previously accepted a calculated dose rate of 0.25 mSv/yr (25 mrem/yr) at the ISFSI controlled area boundary as sufficient evidence that the limit for exposure of the public will not be exceeded under normal conditions. The applicant should provide a discussion that would help determine whether a potential ISFSI would be within the dose rate envelope. These could involve identifying the minimum controlled area dimensions to ensure that the 0.25 mSv/yr dose is not exceeded. Alternatively, the presentation could provide the maximum number of casks that could be stored in an ISFSI, with the minimum distance of 100 meters between the stored fuel and the controlled area boundary (10 CFR Part 72.106(b)), or a suggestion that the licensee install berms, to meet the criterion of 0.25 mSv/yr (25 mrem/yr).

To demonstrate applicant compliance with these requirements, the NRC staff has accepted calculations in the SAR showing a dose rate less than 0.25 mSv/yr (25 mrem/yr) from one cask (or a representative array of casks) at an assumed distance to the controlled area boundary. Such calculations, in practice, can give only a general assessment of the proposed cask system. In addition to unknown information about the ISFSI itself, the implied assumption that an individual would be at the controlled area boundary for 8760 hours (the entire year) is very conservative.

If the above dose rate criteria are satisfied, NRC accepts that the direct-dose regulatory requirements can also be satisfied, although the exact details needed to comply with these limitations will vary from site to site. Therefore, the SAR needs to address such requirements only in general terms. Detailed calculations need not be presented if SAR Section 12, “Operating Controls and Limits,” assigns ultimate compliance responsibilities to the site licensee.

In addition, the applicant should calculate the dose rate at one meter from the cask surface for accident conditions. The model used for these calculations must be consistent with the expected condition of the cask after an accident or natural phenomenon event.

d. Confirmatory Calculations

Independently evaluate the dose rates in the vicinity of the cask for both normal and accident conditions. In determining the level of effort appropriate for these calculations, consider the following factors:

- the degree of sophistication and margin in the SAR analysis
- a comparison of SAR dose rates with those of similar casks that have previously been reviewed, if applicable
- the typical variation in dose rates expected between different codes and cross-section sets
- the fact that actual dose rates will be monitored and limited by the requirements of 10 CFR Part 20
- the restrictions that can be placed on ISFSI operations affecting measured dose rates, as documented in SER Section 12, the site-specific license, or the Certificate of Compliance
- the applicant’s experience in using the methods and codes in previous ISFSI submittals
- use of new, or previously reviewed, methods or codes
- inclusion in the design of any significant departures from previous cask system designs (e.g., unusual shield geometry, new types of materials, or different source terms)

At a minimum, the review should include examination of the applicant’s input to the computer program used for the shielding analysis. Verify use of proper dimensions, material properties, and an appropriate cross-section set. In addition, independently evaluate the use of gamma and neutron source terms.

If a more detailed review is required, independently evaluate the dose rates to ensure that the SAR results are reasonable and conservative. As previously noted, the use of a simple code for neutron calculations is often not appropriate. An extensive evaluation is necessary if major errors are suspected. To the degree possible, the use of a different shielding code with a different analytical technique and cross-section set from that of the SAR analysis will provide a more independent evaluation.

A good reference regarding the treatment of uncertainty in thick-shielded cask analyses has been published by the Electric Power Research Institute[b].

5. Supplemental Information

Supplemental information 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 has deemed necessary. Likewise, the reviewer should request any additional information needed to complete the review process.

VI. Evaluation Findings

Review the 10 CFR Part 72 acceptance criteria and provide a summary statement for each. These statements should be similar to the following model:

Section(s) _____ of the SAR describe(s) shielding structures, systems, and components (SSCs) important to safety in sufficient detail to allow evaluation of their effectiveness.

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[b] B.L. Broadhead, *et. al.*, Electric Power Research Institute, “Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments,” EPRI TR-104329, Palo Alto, California, May 1995.

Section(s) _____ of the SAR evaluates these shielding SSCs important to safety with the objective of assessing the impact on health and safety resulting from operation of the independent spent fuel storage installation (ISFSI).

The staff concludes that the design of the shielding system of the [cask designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria including 10 CFR Part 20 have been satisfied. The evaluation of the shielding system design provides reasonable assurance that the [cask designation] will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

VII. References

1. *U.S. Code of Federal Regulations*, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste," Part 72, Title 10, "Energy."

2. *U.S. Code of Federal Regulations*, Part 20, "Standards for Protection Against Radiation," Title 10, "Energy."

3. L.M. Petrie, *et al.*, Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Vol. 1–4, Rev. 4, April 1995.

4. Oak Ridge National Laboratory, "ORIGEN2.1: Isotope Generation and Depletion Code—Matrix Exponential Method," 1991.

5. TRW Environmental Safety Systems, Inc., "DOE — OCRWM Characteristics Database," DOE/RW-0184-R1.

6. C.V. Parks, *et. al.*, Oak Ridge National Laboratory, "Assessment of Shielding Analysis Methods, Codes, and Data for Spent Fuel Transport/Storage Applications," ORNL/CSD/TM-246, July 1988.

7. American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 6.1.1-1977.