10.0 RADIATION PROTECTION

I. Review Objective

In this portion of the dry cask storage system (DCSS) review, the NRC evaluates the radiation protection capabilities of the proposed cask system. In particular, the NRC staff considers the following aspects:

- Do the proposed DCSS radiation protection features meet the NRC's design criteria for direct radiation?
- Has the applicant proposed engineering features and operating procedures for the DCSS that will ensure the worker's exposures remain as low as is reasonably achievable (ALARA)?
- Will the radiation doses to the general public meet regulatory standards during both normal operation and accident situations?

In ISFSI operation, the major mode of radiation exposure associated with spent fuel storage cask handling results from direct radiation. Because of the cask design requirements, radionuclides are not expected to be released from the cask during either normal operations or design-basis accidents (DBAs).

II. Areas of Review

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

1. radiation protection design criteria and features
2. occupational exposures
3. public exposures
   a. normal conditions
   b. accident conditions and natural phenomenon events
4. ALARA

III. Regulatory Requirements

1. Criteria for radioactive material released due to effluents and direct radiation from an ISFSI or MRS are contained 10 CFR 72.104[1].

2. Criteria for Occupational Exposures are contained in 10 CFR 20.1201, 10 CFR 20.1207, 10 CFR 20.1208, and 10 CFR 20.1301

3. Criteria for public exposures under normal and accident conditions are contained within. [10 CFR 72.104 and 10 CFR 72.106]

4. Criteria for ALARA are contained within 10 CFR 20.1101, 10 CFR 72.24(e), 10 CFR 72.104(b), and 10 CFR 72.126(a)]

IV. Acceptance Criteria

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

1. Design Criteria

Limitations on dose rates associated with direct radiation from the cask are established on the basis of the shielding and confinement evaluations in order to satisfy the regulatory requirements for public dose limits. As stated in 10 CFR Part 72.104, during normal operations and anticipated occurrences, the annual dose equivalent to a real individual located beyond the controlled area, must not exceed the limits discussed below.

2. Occupational Exposures

a. dose limits for adults: 5 rem/yr (total effective dose equivalent)

b. dose limits for minors: 0.5 rem/yr

c. dose to an embryo or fetus (declared pregnant woman): 0.5 rem during entire pregnancy

3. Public Exposures

a. Normal Conditions:

whole body: 25 mrem/yr
thyroid: 75 mrem/yr
other organ: 25 mrem/yr

These doses include the cumulative effects of other nuclear fuel cycle facilities that may be at the same location as the storage system (i.e., the nuclear power plant) and apply to the limiting real individual of the general public residing at a permanent location nearest the facility.

b. Accident Conditions and Natural Phenomenon Events

5 rem to the whole body or any organ of any individual located at or beyond the nearest boundary of the controlled area.

4. ALARA

As a minimum, the proposed ALARA policy must fulfill the following criteria:

a. To the extent practicable, the applicant should employ procedures and engineering controls that are founded upon sound radiation protection principles.

b. Any design change should account for radiation protection, technological, and economical considerations.

c. The applicant should have a written policy statement reflecting management commitment to maintain occupational and public exposures to radiation and radioactive material ALARA.

V. Review Procedures

1. Radiation Protection Design Criteria and Features

a. Design Criteria

Review the principal design criteria presented in Chapter 1 of the applicant's safety analysis report (SAR), as well as any additional detail regarding radiation protection provided in the Shielding and Confinement Evaluation sections of the SAR. Additional criteria that should be presented in SAR Section 10 (if not previously discussed) include (but are not limited to) the following:

(1) The cask system design must satisfy the ALARA and other occupational exposure requirements of 10 CFR Part 20[2].

(2) The sum of the doses from direct radiation and from release of nuclides to the atmosphere must satisfy the requirements of 10 CFR 72.104(a) and 72.106(b). Because of the stringent design requirements for spent fuel cask systems, the release of nuclides into the atmosphere is expected to be insignificant under both normal and accident conditions. Direct radiation is the major mode of exposure.

b. Design Features

Review the general description and functional features of the cask presented in the General Description, as well as any additional information provided in the Shielding and Confinement Evaluation sections of the SAR. In general, the applicant's approach to the relevant design criteria are discussed in these earlier sections of the SAR. They may also be summarily noted in the Radiation Protection section of the safety evaluation report (SER) prepared by the NRC staff.

2. Occupational Exposures

Review the operating procedures in SER Section 8 and direct radiation dose calculations in SER Section 5. The applicant should use these data in SER Section 10 to estimate the doses received by occupational personnel during cask loading and transportation to the ISFSI. The applicant should also identify any significant differences from these doses that may occur during cask retrieval and unloading. In addition, the applicant should present similar dose estimates for periodic or routine maintenance, as well as surveillance activities. These estimates may require additional assumptions concerning adjacent casks for a typical storage configuration.

Also in SAR Section 10, the applicant should present the rationale used to justify the bases for the various exposure times, personnel locations relative to the casks (including hot spots), number of personnel required, and appropriate gamma and neutron dose rates. Verify that the calculated doses are consistent with these estimates. Keep in mind that the actual operations will be performed under an active dose monitoring program that further ensures compliance with the requirements of 10 CFR Part 20. NRC Regulatory Guide (RG) 8.34[3], which was developed to implement revisions to 10 CFR Part 20, can be used to determine the acceptability of the applicant's occupational exposure evaluation and monitoring recommendations.

3. Public Exposures

An SAR for an application seeking approval of a DCSS under 10 CFR Part 72, Subpart L, should include an analysis of potential public exposures that will facilitate a future site-specific suitability analysis required by a licensee prior to DCSS use. One approach is for the applicant to include a dose rate versus distance curve for an assumed array of casks. This curve would assist the reviewer in the determination of the cumulative exposure effects. As an alternative, the analyses documented in the SAR may presume that the public exposure occurs at a distance of 100 meters from the closest stored fuel, with the most severe concentration of casks, and a distance of at least 100 meters between the transfer path and the closest point of public access. 10 CFR 72.106(b) specifies 100 meters as the minimum distance to the closest boundary of the controlled area. These assumptions should be conservative relative to most actual site conditions.

a. Normal Conditions

Review the information in SAR Section 5 regarding the direct dose rate at the controlled area boundary. For applications requesting approval of a cask system under 10 CFR Part 72, Subpart L, the dose for the public should be determined at a distance of 100 meters from the closest boundary of the controlled area, as specified in 10 CFR 72.106(b). However, the applicant may use a longer distance, provided that the longer distance is made a condition of use.

The sum of doses, including an additional margin to account for doses received from other fuel cycle (reactor) operations, must satisfy the requirements of 10 CFR 72.104(a). As discussed in Chapter 5 of this SRP, the direct dose at the controlled area boundary depends on many site-specific conditions, which the SAR may treat in a general manner. Verify that the SAR includes a requirement for site-specific dose analysis and monitoring by the ISFSI licensee, or that the applicant has presented sufficient bounding analyses. (The latter approach will generally require extensive calculations.)

b. Accident Conditions and Natural Phenomenon Events

Review the direct dose rate associated with accident conditions at the boundary of the controlled area, as discussed in Chapter 5. Also review the dose rate resulting from accidental release of radionuclides, as presented in Chapter 7. The accident-related radionuclide release dose should account for both air and liquid pathways as appropriate. In addition, verify that the applicant has evaluated the source terms for both spent fuel fission product and cask surface contamination. The sum of these must satisfy the requirements of 10 CFR 72.106(b). For purposes of demonstrating compliance with 10 CFR 72.106(b), and evaluation against the Environmental Protection Agency Protective Action Guides[4], the skin, extremities, and the lens of the eye may be considered separately from other organs.

As noted in Chapter 5 of this SRP, the time-integrated direct dose at the boundary of the controlled area may be small (compared with that of a hypothetical instantaneous release of all available fission product gases). Consequently, the applicant should estimate the doses at a distance of 100 meters from the storage location to the nearest boundary of the controlled area, unless the SAR specifies a greater distance that is also made a condition of use for the proposed DCSS. Alternatively, applicants may depict dose estimation using a curve showing dose versus distance from an assumed array of casks

4. ALARA

Review the applicant's stated commitment to ALARA policy, and determine whether this commitment influenced the proposed cask design features and operating procedures.

To determine if the applicant's ALARA policy is acceptable, review the evidence that the design methods, approaches, and interactions are in accordance with the ALARA provision in Regulatory Guides 8.8[5] and 8.10[6].

VI. Evaluation Findings

Review the acceptance criteria in Chapter 8.IV of this SRP and provide a summary statement for each. These statements should be similar to the following model:

- The [cask designation] provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106.

- Occupational radiation exposures satisfy the limits of 10 CFR Part 20 and meet the objective of maintaining exposures ALARA.

- The staff concludes that the design of the radiation protection system of the [cask designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection 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. U.S. Nuclear Regulatory Commission, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," Regulatory Guide 8.34, July 1992.

4. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents", EPA 410-R-92-001, May 1992.

5. U.S. Nuclear Regulatory Commission, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable," Regulatory Guide 8.8, Rev. 3, June 1978.

6. U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable," Regulatory Guide 8.10, Revision 1-R, May 1977.