7.0 CONFINEMENT EVALUATION

I. Review Objective

In this portion of the dry cask storage system (DCSS) review, the NRC evaluates the confinement features and capabilities of the proposed cask system. In conducting this evaluation, the NRC staff seeks to ensure that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

II. Areas of Review

This chapter of the DCSS Standard Review Plan (SRP) provides guidance for use in evaluating the design and analysis of the proposed cask confinement system for normal, off-normal, and accident conditions. This evaluation includes a more detailed assessment of the confinement-related design features and criteria initially presented in Sections 1 and 2 of the applicant's safety analysis report (SAR), as well as the proposed confinement monitoring capability, if applicable. In addition, the NRC staff assesses the anticipated releases of radionuclides associated with spent fuel, by independently estimating their leakage to the environment and the subsequent impact on a hypothetical individual located beyond the controlled area boundary.

As prescribed in 10 CFR Part 72, 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 leaktightness of the confinement). Thus, an overall assessment of the compliance of the proposed DCSS with these regulatory limits is deferred until Chapter 10, "Radiation Protection," of this SRP. In addition, the performance of the cask confinement system under accident conditions, as evaluated in this section, may also be addressed in the overall accident analyses, as discussed in Chapter 11 of this SRP.

As described in Section V, "Review Procedures," a comprehensive confinement evaluation *may* encompass the following areas of review:

1. confinement design characteristics
   a. design criteria
   b. design features
2. confinement monitoring capability
3. nuclides with potential for release
4. confinement analyses
   a. normal conditions
   b. leakage of one seal
   c. accident conditions and natural phenomenon events
5. supplemental information

III. Regulatory Requirements

1. Description of Structures, Systems, and Components Important to Safety

The SAR must describe the confinement structures, systems, and components (SSCs) important to safety in sufficient detail to facilitate evaluation of their effectiveness. [10 CFR 72.24(c)(3) and 10 CFR 72.24(l)]

2. Protection of Spent Fuel Cladding

The design must adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage, or the fuel must be confined through other means such that fuel degradation during storage will not pose operational safety problems with respect to removal of the fuel from storage. [10 CFR 72.122(h)(1)]

3. Redundant Sealing

The cask design must provide redundant sealing of the confinement boundary. [10 CFR 72.236(e)]

4. Monitoring of Confinement System

Storage confinement systems must allow continuous monitoring, such that the licensee will be able to determine when to take corrective action to maintain safe storage conditions. [10 CFR 72.122(h)(4) and 10 CFR 72.128(a)(1)]

5. Instrumentation

The design must provide instrumentation and controls to monitor systems that are important to safety over anticipated ranges for normal and off-normal operation. In addition, the applicant must identify those control systems that must remain operational under accident conditions. [10 CFR 72.122(i)]

6. Release of Nuclides to the Environment

The applicant must estimate the quantity of radionuclides expected to be released annually to the environment. [10 CFR 72.24(l)(1)]

7. Evaluation of Confinement System

The applicant must evaluate the cask and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l) and 10 CFR 72.24(d)]

In addition, SSCs important to safety must be designed to withstand the effects of credible accidents and severe natural phenomena without impairing their capability to perform safety functions. [10 CFR 72.122(b)]

8. Annual Dose Limit in Effluents and Direct Radiation from an Independent Spent Fuel Storage Installation (ISFSI)

During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ. [10 CFR 72.104(a)]

IV. Acceptance Criteria

In general, DCSS confinement evaluation seeks to ensure that the proposed design fulfills the following acceptance criteria, which the NRC staff considers to be minimally acceptable to meet the confinement requirements of 10 CFR Part 72:

1. The cask design must provide redundant sealing of the confinement boundary sealing surface. Typically, this means that field closures of the confinement boundary must either have double seal welds or double metallic O-ring seals.

2. The confinement design must be consistent with the regulatory requirements, as well as the applicant's "General Design Criteria" reviewed in Chapter 2 of this SRP. The NRC staff has accepted construction of the primary confinement barrier in conformance with Section III, Subsections NB or NC, of the Boiler and Pressure Vessel (B&PV) Code[1] promulgated by the American Society of Mechanical Engineers (ASME). (This code defines the standards for all aspects of construction, including materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components.) In such instances, the staff has relied upon Section III to define the minimum acceptable margin of safety; therefore, the applicant must fully document and completely justify any deviations from the specifications of Section III. In some cases after careful and deliberate consideration, the staff has made exceptions to this requirement.

3. The applicant must specify the maximum allowed leakage rates for the total primary confinement boundary and redundant seals. (Applicants frequently display this information in tabular form, including the leakage rate of each seal.) In addition, the applicant's leakage analysis should be consistent with the principles specified in the "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials" (ANSI N14.5)[2]. Generally, the allowable leakage

rate must be evaluated for its radiological consequences and its effect on maintaining the necessary inert atmosphere within the cask.

4. The applicant should describe the proposed monitoring capability and/or surveillance plans for mechanical closure seals. In instances involving welded closures, the staff has previously accepted that no closure monitoring system is required. This practice is consistent with the fact that other welded joints in the confinement system are not monitored. However, the lack of a closure monitoring system has typically been coupled with a periodic surveillance program that would enable the licensee to take timely and appropriate corrective actions to maintain safe storage conditions after closure degradation. The discussion in (a) below taken from chapter 2 of this SRP expands on the requirement for continuous monitoring.

   (a) Continuous Monitoring

   The Office of the General Counsel (OGC) has developed an opinion as to what constitutes "continuous monitoring" as required in 10 CFR Part 72.122(h)(4). The staff, in accordance with that opinion has concluded that both routine surveillance programs and active instrumentation meets the intent of "continuous monitoring." Cask vendors may propose, as part of the SAR, either active instrumentation and/or surveillance to show compliance with 10 CFR Part 72.122(h)(4).

   The reviewer should note that some DCSS designs may contain a component or feature whose continued performance over the licensing period has not been demonstrated to staff with a sufficient level of confidence. Therefore the staff may determine that active monitoring instrumentation is required to provide for the detection of component degradation or failure. This particularly applies to components whose failure immediately affects or threatens public health and safety. In some cases the vendor or staff in order to demonstrate compliance with 10 CFR Part 72.122(h)(4), may propose a technical specification requiring such instrumentation as part of the initial use of a cask system. After initial use, and if warranted and approved by staff, such instrumentation may be discontinued or modified.

5. The cask must provide a non-reactive environment to protect fuel assemblies against fuel cladding degradation, which might otherwise lead to gross rupture.[3] Measures for providing a non-reactive environment within the confinement cask typically include drying, evacuating air and water vapor, and backfilling with a non-reactive cover gas (such as helium). For dry storage conditions, experimental data have not demonstrated an acceptably low oxidation rate for UO_2 spent fuel, over the 20-year licensing period, to permit safe storage in an air atmosphere. Therefore, to reduce the potential for fuel oxidation and subsequent cladding failure, an inert atmosphere (e.g., helium cover gas) has been used for storing UO_2 spent fuel in a dry environment. (See Chapter 8 of this SRP for more detailed information on the cover gas filling process.) Note that other fuel types, such as graphite fuels for the high-temperature gas-cooled reactors (HTGRs), may not exhibit the same oxidation reactions as UO_2 fuels and, therefore, may not require an inert atmosphere. Applicants proposing to use atmospheres other than inert gas should discuss how the fuel and cladding will be protected from oxidation.

V. Review Procedures

1. Confinement Design Characteristics

a. Design Criteria

Review the principal design criteria presented in SAR Section 2, as well as any additional detail provided in SAR Chapter 7.

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 7. All drawings, figures, and tables describing confinement features must be sufficiently detailed to stand alone.

Verify that the applicant has clearly identified the confinement boundaries. This identification should include the confinement vessel; its penetrations, valves, seals, welds, and closure devices; and corresponding information concerning the redundant sealing.

Verify that the design and procedures provide for drying and evacuation of the cask interior as part of the loading operations, and that the design is acceptable for the pressures that may be experienced during these operations.

Verify that, on completion of cask loading, the gas fill of the cask interior is at a pressure level that is expected to maintain a non-reactive environment for at least the 20-year storage life of the cask interior under both normal and off-normal conditions and events. This verification can include pressure testing, seal monitoring, and maintenance for casks with seals that are not welded if these are included in chapter 12 as conditions of use. The NRC has previously accepted specification of an overpressure of approximately 14 kiloPascals (~2 psig) and cask leak testing as conditions of use for satisfying this requirement. In addition, if conditions of use require routine inspection of seals by the pressure testing of the cask interior, the cask fill pressure may be linked to that activity.

Coordinate with the structural reviewer (Chapter 3 of this SRP) to ensure that the applicant has provided proper specifications for all welds and, if applicable, that the bolt torque for closure devices is adequate and properly specified.

If applicable, assess the seals used to provide closure. Because of the performance requirements over the 20-year license period, evaluate the potential for deterioration. The NRC staff has previously accepted only metallic seals for the primary confinement. Coordinate with the thermal reviewers (Chapter 4 of this SRP) to ensure that the operational temperature range for the seals, specified by the manufacturer, will not be exceeded.

2. Confinement Monitoring Capability

The NRC staff has found that casks closed entirely by welding do not require seal monitoring. However, for casks with bolted closures, the staff has found that a seal monitoring system has been needed in order to adequately demonstrate that seals can function and maintain a helium atmosphere in the cask for the 20-year license period. A seal monitoring system combined with periodic surveillance enables the licensee to determine when to take corrective action to maintain safe storage conditions. (Note that some designs may not require an inert atmosphere in the cask. In such designs, a periodic surveillance program to check seal leak tightness may be appropriate.)

Although the details of the monitoring system may vary, the general design approach has been to pressurize the region between the redundant seals, with a non-reactive gas, to a pressure greater than that of the cask cavity and the atmosphere. A decrease in pressure between these seals indicates that the non-reactive gas is leaking either into the cask cavity or out to the atmosphere. (Radioactive gas should not be able to leak to the atmosphere in either case; hence, a faulty seal can be detected without radiological consequence.) Note that the volume between the redundant seals should be pressurized using a *non-reactive* gas, thereby preventing contamination of the interior cover gas.

The monitoring system is generally not important to safety and, as such, is classified as Category B under the guidelines of NUREG/CR-6407[4]. Although its function is to monitor confinement seal integrity, failure of the monitoring system does not result in a release of radioactive material. Consequently, the monitoring system for bolted closures need not be designed to the same requirements as the confinement boundary (i.e., ASME Section III, Subsections NB or NC).

In order to meet confinement boundary design standards, either the entire pressurized portion of the monitoring system or the portion extending from the confinement boundary to a second isolation valve would have to meet design-basis requirements for accident conditions (i.e., seismic, tipover, and drop loadings). From a practical perspective, external components of the monitoring system (such as tubing, tanks, and pressure gauges) could not readily be designed to prevent confinement rupture during such accident loadings, since they would have to be able to withstand the dynamic crush loading of the cask. However, these accident loadings would not impair the capability of the inner O-ring to maintain the confinement barrier, since it is designed for these loadings and its operation is confirmed through surveillance using the monitoring system.

Therefore, having a monitoring system that is not designed to confinement barrier standards does not pose any significant risk of radioactive gas release from storage. This practice is justified, since the possibility of a design-basis event occurring at a time between surveilances when the inner seal has randomly failed is extremely remote. Other quality assurance (QA) practices associated with fabrication, examination, testing, and inspection of the monitoring system should be commensurate with components of the confinement system.

The monitoring system should be designed so that its failure can readily be identified during routine surveillance. The NRC staff reviews the monitoring system to assess its ability to fulfill its intended function and to determine whether failure of the monitoring system would degrade the safety systems. Although the monitoring system need not remain functional during a particular accident, monitoring capability must be restored following the accident. Consequently, SAR Section 11 should address the corrective action necessary to resume monitoring.

Examine the specified pressure of the gas in the monitored region to verify that it is higher than both the cask cavity and the atmosphere. Coordinate with the structural and thermal reviewers (Chapters 3 and 4 of this SRP) to verify the pressure in the cask cavity.

Review the applicant's analysis to verify that the total volume of gas in the seal monitoring system is such that normal seal leakage will not cause all of the gas to escape over the lifetime of the cask.
In determining the proposed maximum leakage rate, the applicant should consider the volume between the redundant seals of the confinement cask, the minimum pressure to be maintained, and the length of the proposed routine recharge cycle. The applicant should then specify the leakage rate as an acceptance test criterion in SAR Section 9, even though the actual leakage rate of the seals is expected to be significantly lower.

For redundant seal welded closures, ensure that the applicant has provided adequate justification that the seal welds have been sufficiently tested and inspected to ensure that the weld will behave similarly to the adjacent parent material of the cask. Any inert gas should not leak or diffuse through the weld and cask material in excess of the design leak rate.

Verify that any leakage test, monitoring, or surveillance conditions are appropriately specified in SAR Sections 9 and 11, the license, and/or the Certificate of Compliance.

3. Nuclides with Potential for Release

The NRC staff has determined that, as a minimum, the nuclides shown below in Table 7.1 must be analyzed for potential accident release. The indicated fractions account for the fact that some of these nuclides will be trapped in the fuel matrix or exist in a chemical or physical form that is not capable of release to the environment under credible accident conditions. The NRC accepts the following fractions available for release from spent fuel from boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) for the purpose of analysis regarding compliance with 10 CFR Part 72. Other accident scenarios may be considered provided the applicant properly justifies the associated release fractions. In some cases the applicant may have to consider other radioactive nuclides depending upon the specific source term analysis of its spent fuel.

The quantities of these radioactive nuclides are often presented in SAR Section 5, since they are generally determined during the evaluation of gamma and neutron source terms in the shielding analysis. Coordinate with the shielding review (Chapter 5 of this SRP) to verify that the applicant has adequately determined these nuclides.

Table 7.1	
Nuclide	Fractions Available for Release*, [5, 6]
^{3}H	0.30
^{85}Kr	0.30
^{129}I	0.10
^{137}Cs	2.3×10^{-5}
^{134}Cs	2.3×10^{-5}
^{90}Sr	2.3×10^{-5}
^{106}Ru	1.5×10^{-5}
^{60}Co**	0.15

* Except for ^{60}Co, only failed fuel rods contribute significantly to the release. Total fraction of radionuclides available for release must be multiplied by the fraction of fuel rods assumed to have failed.

** Source of ^{60}Co is crud on fuel rods, estimated to be 140 μCi/cm^2 for PWRs and 600 μCi/cm^2 for BWRs. Total ^{60}Co activity is this estimate times the total surface area of all rods in the cask [7].

4. Confinement Analyses

Review the applicant's confinement analysis and the resulting annual dose at the controlled area boundary. In general, the staff evaluates analyses for three specific scenarios, as follows:

a. Normal Conditions

If the confinement boundary is welded, or if the region between the two mechanical seals is monitored, the staff accepts that no discernible undetected leakage is credible. Hence, the dose at the controlled area boundary from atmospheric release is negligible.

b. Leakage of One Seal

Depending on its extent, failure of one redundant mechanical seal should not result in release of radioactive material.If the between-seal volume remains at higher pressure than the interior of the cask, no release of radioactive material should ensue. If the pressure differential between the between-seal volume and the atmosphere or the interior of the cask is equalized, however, radioactive material may escape at a rate associated with the acceptable leakage rate across one seal (see V.2, above), the actual pressure differential, and the gas viscosity. Failure of both redundant seals would result in a greater release. Note that components of the between-seal volume pressurization and pressure monitoring system form part of the outer seal.

The NRC staff has accepted this scenario with the assumption that 3 to 10 percent of the fuel rods have failed. Current practice is to assume 10 percent unless the applicant provides sufficient justification for considering a lesser figure. Coordinate with the structural and thermal reviewers (Chapters 3 and 4 of this SRP) to verify that the applicant has adequately determined the cask cavity pressure applicable for this condition.

In addition to the quantity of nuclides available for release and the pressure of the cask cavity, the dose at the controlled area boundary depends on the following factors:

- seal leakage rate
- distance from the cask to the controlled area boundary

- atmospheric dispersion factor
- an individual's breathing rate (except for Kr, for which the dose should be determined using EPA Guide No. 12[8])
- dose conversion factors

The applicant should specify maximum allowable seal leakage rates as design criteria, as discussed in Chapter 12. The minimum distance between the casks and the controlled area boundary is generally also a design criterion; however, 10 CFR Part 72 requires this distance to be at least 100 meters from the ISFSI.

Because a release resulting from seal failure will occur over a substantial period of time, the staff has accepted, as a bounding condition, the atmospheric dispersion factors of Regulatory Guide 1.145[9] on the basis of F-stability diffusion, a wind speed of 1 m/s, and plume meandering. Also, the staff has accepted either an adult breathing rate of 2.5×10^{-4} m^3/s, as specified in Regulatory Guide 1.109[10], or a worker breathing rate of 3.3×10^{-4} m^3/s, as specified in EPA Guidance Report No. 11[11]. Dose conversion factors for inhalation, whole body dose, and thyroid dose should be equivalent to those indicated in EPA Guidance Report No. 11.

Review the applicant's controlled area boundary dose calculation. Verify that the applicant has determined both the whole body dose and the thyroid dose. A conservative bound is established by assuming that an individual is present at the controlled area boundary for the full year (8760 hours). The estimates of the dose that would be received by this individual have typically been low relative to the regulatory limits. An alternative to this conservative assumption may be acceptable if the applicant provides a convincing justification. The dose that an individual would receive in case of a seal leak is usually very small, and this conservatism has not historically posed any difficulties in meeting the regulatory limits; however, this criterion may be reconsidered if the applicant provides sufficient justification.

c. Accident Conditions and Natural Phenomenon Events

Coordinate with the structural reviewers (Chapter 3 of this SRP) to determine the effect of specific accident conditions and natural phenomenon events on the cask confinement system. A full confinement barrier must remain intact under all design-basis accident and natural phenomenon events. Failure of one of the redundant seals may be acceptable as long as the failure of one seal does not result in loss of the confinement function. Nevertheless, to demonstrate the overall safety of dry spent fuel storage, the staff conservatively assumes a failure of the confinement boundary with 100 percent of the fuel rods failed for calculation of an accident dose to a hypothetical individual located at or beyond the boundary of the controlled area.

The analysis for the above scenario is similar to that for failure of one seal. In this situation, the applicant need not consider the cask cavity pressure. Because the leak is assumed to be instantaneous, the plume meandering factor of Regulatory Guide 1.145 is not typically applied. This is equivalent to using an atmospheric dispersion factor based on Regulatory Guide 1.25. Hence, this dispersion factor is generally found to be 4 times higher than that for the case of a single seal failure.

Review the applicant's calculation for the dose at the controlled area boundary, in relation to the regulatory limits listed in 10 CFR 72.106(b). Verify that the applicant has determined both the whole body dose and the thyroid dose. Note that for an instantaneous release (and instantaneous exposure), the time that an individual remains at the controlled area boundary is not a factor in the dose calculation.

5. Supplemental Information

Ensure that all supportive information or documentation has been provided or is readily available.This includes, but is not limited to, justification of assumptions or analytical procedures, test results, photographs, computer program descriptions, input and output, and applicable pages from referenced documents. Reviewers should request any additional information needed to complete the review.

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) confinement structures, systems, and components (SSCs) important to safety in sufficient detail in to permit evaluation of their effectiveness.

- The design of the [cask designation] adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the safety evaluation report (SER) discusses the relevant temperature considerations.

- The design of the [cask designation] provides redundant sealing of the confinement system closure joints by ______.

- The confinement system is monitored with a _______ monitoring system as discussed above (if applicable). No instrumentation is required to remain operational under accident conditions.

- The quantity of radioactive nuclides postulated to be released to the environment has been assessed as discussed above. In Section 10 of the SER, the dose from these releases will be added to the direct dose to show that the [cask designation] satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

- The cask confinement system has been evaluated [by appropriate tests or by other means acceptable to the Commission] to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

- The staff concludes that the design of the confinement 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 confinement 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, the applicant's analysis and the staff's confirmatory analysis, and accepted engineering practices.

VII. References

1. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code," Section III, Subsections NB and NC.

2. American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5, 1987.

3. Pacific Northwest Laboratory, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, November 1987.

4. Idaho National Engineering Laboratory, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," NUREG/CR-6407, INEL-95/0551, February 1996.

5. U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25 (Safety Guide 25), March 1972.

6. E.L. Wilmot, Sandia National Laboratory, "Transportation Accident Scenarios for Commercial Spent Fuel," SAND80-2124, Albuquerque, NM, February 1981.

7. R.P. Sandoval, *et al.*, Sandia National Laboratories, "Estimate of CRUD Contribution to Shipping Cask Containment Requirements," SAND88-1358, TTC-0811, UC-71, January 1991.

8. U.S. Environmental Protection Agency, "Federal Guidance Report No. 12: External Exposure to Radiouclides in Air, Water, and Soil," EPA 402-R-93-081, September 1993.

9. U.S. Nuclear Regulatory Commission, "Atmospheric Dispersement Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1989.

10. U.S. Nuclear Regulatory Commission, "Calculations of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, October 1977.

11. U.S. Environmental Protection Agency, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," DE89-011065, 1988.