Interim Staff Guidance Used by the Spent Fuel Project Staff

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Spent Fuel Project Office
Interim Staff Guidance - 5, Revision 1

Issue: **Confinement Evaluation**

Discussion:

Several changes have occurred since the issuance of NUREG-1536, "Standard Review Plan (SRP) for Dry Cask Storage Systems," that affect the staff's approach to confinement evaluation. The attachment to this ISG integrates the current staff approach into a revision of ISG-5. The highlights of the changes include:

- Reflects October 1998 revisions to 10 CFR 72.104 and 10 CFR 72.106.

- Expands and clarifies acceptance criteria associated with confinement analysis and acceptance of "leak tight" testing instead of detailed confinement analysis.

- Updates staff review guidance for design and requirements for the cask seal monitoring system and adds guidance for accident analysis of "latent" failure concerns.

- Updates source term guidance to (1) include ISG-5 recommendations, (2) include actinide activity that contributes greater than 0.01% of the design basis activity, and (3) allow for a reduction of fines that can escape the cask (with justification by applicant).

- Deletes non-mechanistic (confinement boundary failure) accident analysis and revise staff review guidance for evaluation of normal, off-normal, and accident cases. The significant change is that the evaluated leaks are related to the as-tested leak rate.

- Updates confinement analysis section to reflect ISG-5 and describe what types of analysis should be done. Dose to lens of the eye will be addressed if skin dose and TEDE do not exceed 15 rem.

Regulatory Basis: See attachment

Technical Review Guidance:

To ensure consistency in reviews, consolidate various references, and simplify the reviews, the guidance in the attachment to this ISG should be used instead of SRP Chapter 7.

Recommendation:

SRP, Chapter 7, should be replaced with attached confinement evaluation. In addition, SRP Chapter 11 Section V.2 should be revised regarding classification of the monitoring system to be consistent with SRP Chapter 7. Further, SRP Chapter 2 Section V.2.b.(3)(e) should be updated to remove reference to non-mechanistic failure of confinement boundary event.

Approved ____

E. William Brach **Date**

Attachment: As stated

ATTACHMENT TO ISG-5 REVISION 1
CONFINEMENT EVALUATION

I. Review Objective

In this portion of the dry cask storage system (DCSS) review, the U.S. Nuclear Regulatory Commission (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 0.25 mSv (25mrem) to the whole body, 0.75 mSv (75mrem) to the thyroid, and 0.25 mSv (25mrem) to any other critical 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. Typically, this means that field closures of the confinement boundary must either have two seal welds or two 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. The maximum allowed leakage rate is the "as tested" leak rate measured by the leak test performed on the cask field closure. Generally, as discussed in items a. through d., below, the allowable leakage rate must be evaluated for its radiological consequences and its effect on maintaining an inert atmosphere within the cask. However, for storage casks having closure lids that are designed and tested to be "leak tight", as defined in "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5-1997[2], the analyses discussed in a. through d., below, are unnecessary.[a]

   a. The applicant's leakage analysis should be consistent with the methods described in ANSI N14.5-1997.

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[a] For casks that are demonstrated to be leak tight, the review procedures discussed in sections V.3 and V.4 are not applicable.

b. During normal operations and anticipated occurrences, dose calculations based on the allowable leakage rate must demonstrate that the annual dose equivalent to any real individual who is located at the boundary or outside the controlled area does not exceed the limits given in 10 CFR 72.104(a).

c. After a design-basis accident, dose calculations based on the allowable leakage rate must demonstrate that an individual at the boundary or outside the controlled area does not receive a dose that exceeds the limits given in 10 CFR 72.106(b).

d. The applicant's leakage analysis must demonstrate that an inert atmosphere will be maintained within the cask during the storage lifetime.

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 if closure degradation occurred.

   To show compliance with 10 CFR Part 72.122(h)(4), cask vendors have proposed, and the staff has accepted, routine surveillance programs and active instrumentation to meet the continuous monitoring requirements. 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 fuel 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. The monitoring system is leakage tested to the same leak rate as the confinement boundary. Installed instrumentation is routinely checked per surveillance requirements. 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. For normal operations, radioactive material should not be able to leak to the atmosphere; hence this design allows for detecting a faulty seal 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 staff has accepted monitoring systems as not important to safety and 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 alone does not result in a gross 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).

Dependant on the monitoring system design, there could be a lag time before the monitoring system indicates a postulated degraded seal leakage condition. Degraded seal leakage is leakage greater than the tested rate that is not identified within a few monitoring system surveillance cycles. The occurrence of a degraded seal without detection is considered a "latent" condition and should be presumed to exist concurrently with other off-normal and design-basis events (see SRP section 2, paragraph V.2.b.). Note that once the degraded seal condition is detected, the cask user will initiate corrective actions.

For the off-normal case, the monitoring system boundary remains intact and this condition would be bounded by the off-normal analysis. If the monitoring system would not maintain integrity under design-basis accident conditions, additional safety analysis may be necessary. The staff recognizes that the possibility of a degraded seal condition is small and that the possibility of a degraded seal condition concurrent with a design-basis event that breaches the monitoring system pressure boundary is very remote. However, these probabilities have not been quantified. To address this concern, the staff accepts a demonstration that the probability of occurrence of a latent, degraded seal, condition concurrent with a design basis event that breaches the monitoring system boundary is acceptably low (e.g. less than 1×10^{-6} per year). Alternatively, the staff accepts a demonstration that the dose consequences of this event are within the limits of 10 CFR 72.106(b).

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 fractions of radioactive materials available for release from spent fuel, provided in Table 7-1 for pressurized-water reactor (PWR) fuel and boiling-water reactor (BWR) fuel for normal, anticipated occurrences (off-normal), and accident conditions, should be used in the confinement analysis to demonstrate compliance with 10 CFR Part 72. These fractions account for radionuclides trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that is not releasable to the environment under credible normal, off-normal, and accident conditions. Other release fractions may be used in the analysis provided the applicant properly justifies the basis for their usage. For example, the staff has accepted, with adequate justification, reduction of the mass fraction of fuel fines that can be released from the cask.

The staff has accepted the following rod breakage fractions for the confinement evaluations:

1% for normal conditions
10% for off normal conditions
100% for design basis accident and extreme natural phenomena

For the source term, the NRC staff has accepted, as a minimum for the analysis, the activity from the Co^{60} in the crud, the activity from iodine, fission products that contribute greater than 0.1% of design basis fuel activity, and actinide activity that contributes greater than 0.01% of the design basis activity. In some cases, the applicant may have to consider additional radioactive nuclides depending upon the specific analysis. The total activity of the design basis fuel should be based on the cask design loading that yields the bounding radionuclide inventory (considering initial enrichment, burnup, and cool time).

Table 7.1[*]		
Variable	**Fractions Available for Release[**]**	
	PWR AND BWR FUEL	
	Normal and Off-normal Conditions	Hypothetical Accident Conditions
Fraction of gases released due to a cladding breach, f_G†	0.3	0.3
Fraction of volatiles released due to a cladding breach, f_V†	2×10^{-4}	2×10^{-4}
Mass fraction of fuel released as fines due to cladding breach, f_F	3×10^{-5}	3×10^{-5}
Fraction of crud that spalls off cladding, f_C	0.15[#]	1.0[#]

* Values in this table are taken from NUREG/CR-6487[5].

** 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.

† In accordance with NUREG/CR-6487, gases species include H-3, I-129, Kr-81, Kr-85, and Xe-127; volatile species include Cs-134, Cs-135, Cs-137, Ru-103, Ru-106, Sr-89, and Sr-90.

\# The source of radioactivity in crud is ^{60}Co on fuel rods. At the time of discharge from the reactor, the specific activity, S_c, is estimated to be 140 μCi/cm^2 for PWRs and 1254 μCi/cm^2 for BWRs. Total ^{60}Co activity is this estimate times the total surface area of all rods in the cask[6]. Decay of ^{60}Co to determine activity at the minimum time before loading is acceptable.

The quantities of 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 developed the source term.

It is important to recognize that design basis normal or accident conditions resulting in confinement boundary failure are not acceptable. Preservation of the confinement boundary during design basis conditions is confirmed by the structural analysis. The confinement analyses demonstrate that, at the measured leakage rates, and assumed nominal meteorological conditions, the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b) can be met. Each ISFSI, wether it is a site specific or a general license, is also required to have a site specific confinement analysis and dose assessment to demonstrate compliance with these regulations.

4. Confinement Analysis

Review the applicant's confinement analysis and the resulting doses for the normal, off-normal, and accident conditions at the controlled area boundary.

The analysis typically includes the following common elements:

- calculation of the specific activity (e.g. Ci/cm^3) for each radioactive isotope in the cask cavity based on rod breakage fractions, release fractions, isotopic inventory, and cavity free volume
- using the tested leak rate and conditions during testing as input parameters, calculation of the adjusted maximum seal leakage rates (Cm^3/sec) under normal, off-normal, and hypothetical accident conditions (e.g. temperatures and pressures)
- calculation of isotope specific leak rates (Q_i - Ci/sec) by multiplying the isotope specific activity by the maximum seal leakage rates for normal, off-normal, and accident conditions
- determination of doses to the whole body, thyroid, other critical organs, lens of the eye, and skin from inhalation and immersion exposures at the controlled area boundary (considering atmospheric dispersion factors - χ/Q)

The applicant should specify maximum allowable "as tested" seal leakage rates as a Technical Specification, as discussed in Chapter 12. Guidance on the calculations of the specific activity for each isotope in the cask and the maximum allowable helium seal leakage rates for normal, off-normal, and accident conditions can be found in NUREG/CR-6487 and ANSI N14.5-1997. 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.

For the dose calculations, the staff has accepted the use of either an adult breathing rate (BR) of
2.5×10^{-4} m^3/s, as specified in Regulatory Guide 1.109[7], or a worker breathing rate of 3.3×10^{-4} m^3/s, as specified in EPA Guidance Report No. 11[8]. The dose conversion factors (DCF) in EPA Guidance Report No. 11 for the whole body, critical organs, and thyroid doses from inhalation should be used in the calculation. The bounding DCFs from EPA Report No. 11 should be used for each isotope unless the applicant justifies an alternate value. No weighting or normalization of the dose conversion factors is accepted by the staff. For each isotope, the committed effective dose equivalent ($CEDE_i$ - for the internal whole body dose) or the committed dose equivalent (CDE_i - for the internal organ dose) are calculated as follows:

$CEDE_i$ or CDE_i (in mrem per year for normal/offnormal or mrem per accident)

$$= Q_i * DCF_i * \chi/Q * \text{B-Rate} * \text{Duration} * \text{conversion factor}^{b}$$

[b]The conversion factor, if required, converts the input units into the desired form, e.g. mrem/year.

For the contributions to the whole body, thyroid, critical organs, and skin doses from immersion (external) exposure, the DCFs in EPA Guidance Report No. 12[9] should be used. Again, no weighting or normalization of the dose conversion factors is accepted by the staff.

The deep dose equivalent (DDE_i - for the external whole body) and the shallow dose equivalent (SDE_i - for the skin dose) are calculated as follows:

DDE_i or SDE_i (in mrem per year for normal/offnormal or mrem per accident)
$= Q_i * DCF_i * \chi/Q *$ Duration * conversion factor[b]

The total effective dose equivalent, TEDE = $\sum CEDE_i + \sum DDE_i$

For a given organ, the total organ dose equivalent, TODE = $\sum CDE_i + \sum DDE_i$

The total skin dose equivalent SDE = $\sum SDE_i$

Compliance with the lens dose equivalent (LDE) limit is achieved if the sum of the SDE and the TEDE do not exceed 0.15 Sv (15 rem). This approach is consistent with guidance in ICRP-26[10].

In general, the staff evaluates analyses for normal, off-normal, and accident conditions.

a. Normal Conditions

For normal conditions, a bounding exposure duration assumes that an individual is present at the controlled area boundary for one full year (8760 hours). An alternative exposure duration may be considered by the staff if the applicant provides justification.

Because any potential release, resulting from seal leakage, would typically occur over a substantial period of time, the staff accepts (for applications for certificates) calculation of the atmospheric dispersion factors (χ/Q) according to Regulatory Guide 1.145[11] assuming D-stability diffusion and a wind speed of 5 m/s.

For the likely case of an ISFSI with multiple casks, the doses need to be assessed for a hypothetical array of casks during normal conditions. Therefore, the staff anticipates that the resulting doses from a single cask will be a small fraction of the limits prescribed in 10 CFR 72.104(a) to accommodate the array and the external direct dose.

Note: If the region between redundant, confinement boundary, mechanical seals is maintained at a pressure greater than the cask cavity, the monitoring system boundaries are tested to a leakage rate equal to the confinement boundary, and the pressure is routinely checked and the instrumentation is verified to be operable in accordance with a Technical Specification Surveillance Requirement, the staff has accepted that no discernible leakage is credible. Therefore, calculations of dose to the whole body, thyroid, and critical organs at the controlled area boundary from atmospheric releases during normal conditions would not be required for normal conditions.

b. Off-normal Conditions

For off-normal conditions, the bounding exposure duration and atmospheric dispersion factors (χ/Q) are the same as those discussed above for normal conditions.

To demonstrate compliance with 10 CFR 72.104(a), the staff accepts whole body, thyroid, and critical organ dose calculations for releases from a single cask. However, the dose contribution from cask leakage should also be a fraction of the limits specified in 10 CFR 72.104(a) since the doses from other radiation sources are added to this contribution.

c. Accident Conditions

For hypothetical accident conditions, the duration of the release is assumed to be 30 days (720 hours). A bounding exposure duration assumes that an individual is also present at the controlled area boundary for 30 days. This time period is the same as that used to demonstrate compliance with 10 CFR 100 for reactor facilities licensed per 10 CFR 50 and provides good defense in depth since recovery actions to limit releases are not expected to exceed 30 days.

For hypothetical accidents conditions, the staff has accepted calculation of the atmospheric dispersion factors (χ/Q) of Regulatory Guide 1.145 or Regulatory Guide 1.25[12] on the basis of F-stability diffusion, and a wind speed of 1 m/s.

To demonstrate compliance with 10 CFR 72.106(b), the staff accepts whole body, thyroid, critical organ, and skin dose calculations for releases of radionuclides from a single cask.

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, 1997.

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, "Containment Analysis for Type B Packages Used to Transport Various Contents," NUREG/CR-6487, November 1996.

6. 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.

7. 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.

8. 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.

9. 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.

10. International Commission on Radiation Protection, "Statement from the1980 Meeting of the ICRP," ICRP Publication 26, Pergammon Press, New York, NewYork, 1980.

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