Enclosure 3 to E-56259

Updated SAR Change Pages for RAIs NP-7-2 and NP-A-2

(Public)

7.7 References

7-1 "Post Transport Package Evaluation," QP-10.02, Revision 1.

7-2 US Nuclear Regulatory Commission NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979.

7-3 US Nuclear Regulatory Commission NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

7-4 ASME NOG-1-2010, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," The American Society of Mechanical Engineers, 2010.

7-5 US Nuclear Regulatory Commission Regulatory Issue Summary 2005-25, Supplement 1, "Clarification of NRC Guidelines for Control of Heavy Loads," May 2007.

7-6 US Nuclear Regulatory Commission Regulatory Guide 1.29, "Seismic Design Classification," Revision 4, March 2007.

7-7 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section IX, Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators, latest Edition at time of PO issuance.

7-8 American Welding Society D1.1, Structural Welding Code – Steel, 1996.

7-9 ANSI N14.6-1993 American National Standard for Radioactive Materials – "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More", 1993.

7-10 Crane Manufacturers Association of America (CMAA) Specification #70, 2004.

7-11 American Society of Mechanical Engineers NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," 1994 Edition.

7-12 NAC Equipment Specification Sheet, ESS-032, "Kuosheng ISFSI Gantry Cranev Lift Booms and Trolley Beam Side Shift Assemblies."

7-13 American Society of Mechanical Engineers B30.16-*2007*, "Overhead Hoists (Underhung)".

7-14 American Society of Mechanical Engineers NUM-1, "Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type)."

7-15 Deleted.

7-16 American Society of Mechanical Engineers B30.10-2009, "Hooks."

7-17 Calculation 630075-2004, rev 13, Gantry Crane Structural Evaluation, NAC International.

7-18 Calculation 630075-2006, rev 4, Adapter Plate and Clevis Plate Structural Evaluation, NAC International.

7-19 Calculation 630075-2015, rev 7, Structural Evaluation of Miscellaneous Gantry Crane Components, NAC International.

All Indicated Changes are in response to RAI P-7-2

A.11.1 Confinement Boundary

The confinement boundary for the FO-, FC- and FF-DSCs is documented in Section 3.3.2.1 of [A.11-1]. Reference [A.11-1] does not include a figure showing the confinement boundary for the FO-, FC- and FF-DSCs. However, Figure 7.1-1 of reference [A.11-12] provides a figures that shows the component and welds that make up the confinement boundary for the 24PT1-DSC which is also applicable to the FO-, FC-, and FF-DSCs with one exception, the FO-, FC-, and FF-DSCs do not have a "helium Leak Test Plug" in the Outer Top Cover Plate. Drawings for the canisters, including the confinement boundary are referenced in Section A.4.6.

The canisters will not release radioactive contents under all normal, off-normal, and accident conditions; see Section 3.3.2 and Section 8.2.2 of [A.11-1]. However, during fabrication and closure operations the confinement boundary was leak tested to 10^{-5} *ref*-cm^3/sec in accordance with ANSI N14.5 [A.11-2]. Therefore, for these canister designs, a non-mechanistic release is postulated based on a leakage rate of 10^{-5} *ref*-cm^3/sec. In addition, bounding evaluations in Section A.7.7 are performed to demonstrate that the confinement boundaries for the FO-, FC-, FF-DSCs do not exceed ASME B&PV Subsection NB Article NB-3200 (Level A allowables) during normal conditions of transport to provide reasonable assurance that the confinement boundary is not adversely impacted by transport to the WCS CISF.

Section 4.3, Codes and Standards, of the Technical Specifications for the Rancho Seco ISFSI [A.11-11] cites the applicable ASME Code for the MP187 FO-, FC-, and FF-DSCs.

Section 3.1, "DSC Integrity," of the Technical Specifications for the Rancho Seco ISFSI; [A.11-11] includes limiting condition for operation (LCO) 3.1.1 for DSC vacuum pressure, LCO 3.1.2 for DSC helium leakage rate, and LCO 3.1.3 for DSC helium backfill pressure. These LCOs create dry, inert, leak tight atmosphere, which contributes to preventing the leakage of radioactive material.

A.11.2 Potential Release Source Term

As noted in Section A.11.1 the FO-, FC-, FF- DSCs, a non-mechanistic leakage rate of 10^{-5} *ref*-cm^3/sec is postulated. The actinides and fission products for a B&W 15x15 fuel assembly are computed using SCALE6/ORIGEN-ARP. Two isotopic sets are considered, based on the design basis neutron and gamma sources. The design basis neutron source has a burnup of 38,268 MWd/MTU, enrichment of 3.18% U-235, and was discharged in 1983. The design basis gamma source has a burnup of 34,143 MWd/MTHM, enrichment of 3.21% U-235, and was discharged in 1989. The two source terms considered are decayed until June 2020, which corresponds to the placement of the first canisters at the WCS Consolidated Interim Storage Facility (WCS CISF). The reported source term in Table A.11-1 is the maximum value of the two isotopic sets considered. The design basis radioactive inventory for the confinement evaluation included in reference [A.11-1] was determined using these same bounding fuel assemblies as documented in Section 7.2.1 of Volume I of [A.11-1] (See also calculation 2069-0507, Revision 0 included in Volume IV of [A.11-1]).

The crud source is determined based on 140 μCi/cm^2 Co-60 on the surfaces of the SNF rods at the time of discharge [A.11-3]. The design basis gamma assembly was discharged in 1989, or 31 years decay until loading. Therefore, the crud source term in Table A.11-1 is decayed 31 years.

All Indicated Changes are in response to RAI NP-A-2

A.11.3 Confinement Analysis

Per Section A.11.1 the FO-, FC-, FF- DSCs, a non-mechanistic leakage rate of 10^{-5} *ref*-cm^3/sec is postulated. A confinement analysis is performed for normal, off-normal, and accident conditions to determine the dose to an individual due to inhalation and ingestion. There is no credible mechanism that would produce a leak of this magnitude through the confinement boundary of the canister. All welds in the canister shell are volumetrically examined, as is the weld between the inner bottom cover plate and the shell. Because it is not feasible to volumetrically examine the inner top cover plate weld, this weld is leak tested in accordance with the stated criteria. However, no credit is taken for the presence of the outer top cover plate, which is welded to the canister shell with a 0.5 inch weld that receives no fewer than three levels of dye-penetrant testing. The releases postulated in this analysis, therefore, are several orders of magnitude greater than any expected release.

A.11.3.1 Methodology

1. Calculate the specific activity (Ci/cm^3) in the canister cavity for each radioactive isotope based on the rod breakage fractions, release fractions, isotopic inventory, and cavity free volume. It is conservatively assumed that every SNF assembly in every canister has the same radiological source as the design basis SNF assembly. This assumption is conservative because many SNF assemblies will have less activity than the design basis source. Two sets of release fractions are considered: fuel-to-canister release fractions and Canister-to-Environment release fractions. The fuel-to-canister release fractions are the fraction of isotopes released from the interior of the SNF rod to the internal void region of the canister upon failure of the SNF rods. The fuel-to-canister release fractions used in this analysis are those specified in NUREG-1536 [A.11-4, Table 5-2] or NUREG-1567 [A.11-5, Table 9.2] and are summarized in Table A.11-2. The Canister-to-Environment release fractions are the fraction of isotopes released from the canister to the environment. As the radioactive materials from the SNF assembly will not be released directly to the environment, there will be some release retention in the canister. The fraction of radioactive materials released from the canister to the environment is justified and provided in [A.11-6, Table 3-5] and reproduced in Table A.11-3. These additional factors account for material that may condense, plate out or be filtered out before escaping the canister due to leakage hole size. This accounting of canister retention is also documented in other NRC documents [A.11-7, Section 7.3.8]. The two sets of release fractions are combined to create the fuel-to-environment release fractions in Table A.11-4. No credit is taken for retention of material released from the canister and potentially retained in the Horizontal Storage Module (HSM).

2. Using the as-tested leak rate and adjusting for normal, off-normal, and accident conditions in the canister cavity, determine the adjusted maximum canister leak rate for each set of conditions. The guidance of ANSI N14.5 [A.11-2] is used to calculate the adjusted leak rates.

All Indicated Changes are in response to RAI NP-A-2

3. Calculate the isotope specific leak rates by multiplying the specific activities by the seal leak rate for each condition.

4. Determine the dose to the whole body, thyroid, lens of the eye, skin, and other critical organs from inhalation and immersion exposures at the controlled area boundary. Atmospheric dispersion factors are determined using Regulatory Guide 1.145 [A.11-8] and dose conversion factors are taken from EPA Guidance Reports No. 11 [A.11-9] and No. 12 [A.11-10].

A.11.3.2 Specific Activities for Release

Specific activities for release are computed for the canister based on SNF assembly activities in Table A.11-1 and normal, off-normal, and accident release fractions in Table A.11-4. The specific activities are based on 24 SNF design basis assemblies per canister and a cavity free volume of 5,592,315 cm^3. The specific activities for release are provided in Table A.11-5. The maximum number of fuel assemblies in any canister is 24 SNF assemblies; therefore, this assumption bounds all of the loaded FO-, FC- and FF-DSCs.

A.11.3.3 Leakage Rates

A leak rate in the units *ref*-cm^3/sec corresponds to a leak of dry air at a temperature of 25°C from a pressure of 1 atm (absolute) to a pressure of 0.01 atm (absolute). Because the canister contains an atmosphere that is primarily helium at various temperatures and pressures, the specified standard leak rate must be adjusted for the change in gas, temperature, and pressure. The design basis conditions for the canisters are provided in Table 8-2a of [A.11-1]. Using the method from ANSI N14.5 [A.11-2] and a leakage hole length assumed to be the size of the weld length (3/16 inches), the hole diameter is computed to be 4.7611×10^{-4} cm for a leakage rate of 10^{-5} *ref*-cm^3/sec.

Based on ANSI N14.5, the computed leakage rates for the three operating conditions are:

- Normal condition leakage rate = 4.4914×10^{-6} cm^3/sec
- Off-normal condition leakage rate = 7.5892×10^{-6} cm^3/sec
- Accident condition leakage rate = 2.5413×10^{-5} cm^3/sec

The isotope specific leak rates (Q_i - Ci/sec) used in the exposure calculations are equal to the number of canisters, multiplied by the specific activity, multiplied by the leakage rate, or:

$$Q_i = N \cdot S_i \cdot L$$

where: N is the number of canisters

S_i is the specific activity of nuclide i (Ci/cm^3)

All Indicated Changes are in response to RAI NP-A-2