RAI Volume 2, Chapter 2.1.1.7, Seventh Set, Number 1:

> Provide (a) input/output files used for determining the minimum required enriched boron concentrations to maintain sub-criticality under normal and off-normal handling operations in the Wet Handling Facility (WHF) pool (SAR Section 1.14.2.3.3.4), and (b) information used for the WHF pool criticality analysis including geometry and proposed material specifications, type of fuel, soluble poison composition (for different cases), isotopic content, and all associated errors and uncertainties.
>
> In the SAR Section 1.14.2.3.3.4, the applicant states that, subcriticality is maintained during normal and off-normal handling operations, while crediting no more than 30% of the minimum required boron concentration. However, the analyses information and input files have not been provided. This information is needed to demonstrate compliance with 10 CFR 63.112(e)(6).

1. RESPONSE

In response to Part (a), the input and output files used for determining the minimum required enriched boron concentration to maintain subcriticality under normal and off-normal handling operations in the Wet Handling Facility pool are provided in Attachment 2 of *Nuclear Criticality Calculations for the Wet Handling Facility* (BSC 2007).

In response to Part (b), the information used for the Wet Handling Facility pool criticality analysis, including geometry, material specification, fuel characteristics, soluble boron composition, and isotopic content, along with associated uncertainties or treatment of ranges in the criticality calculations, is provided in Section 6 of *Nuclear Criticality Calculations for the Wet Handling Facility* (BSC 2007) as follows:

- Section 6.1 provides the geometry descriptions for the assemblies, canisters, staging racks, and other potential configurations.
- Section 6.2 provides the material specifications used in the calculation, including the fuel, the neutron absorber, and other structural materials.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2007. *Nuclear Criticality Calculations for the Wet Handling Facility.* 050-00C-WH00-00100-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071212.0001.

RAI Volume 2, Chapter 2.1.1.7, Seventh Set, Number 2:

> Provide a description of the inspection program that the applicant plans to use to enforce Procedural Safety Control (PSC) PSC-9, as per ANSI/ANS 8.14-2004 "*Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors*," which the applicant claimed to follow in SAR Section 1.14.3.1. Include a description of the type of equipment that may be used.
>
> In SAR Table 1.9-10, PSC-9, the applicant states the minimum concentration and enrichment in ^{10}B to be maintained in the WHF pool. However, the applicant has not provided any information on how it plans to enforce the PSC-9, and, as a standard practice, reactor pools use natural boron rather than enriched boron. This information is needed to demonstrate compliance with 10 CFR 63.112(e)(6).

1. RESPONSE

The details of the inspection program to support procedural safety control PSC-9 will be developed during the detailed design phase and prior to receipt and possession of high-level radioactive waste and spent nuclear fuel. This is consistent with the commitment in SAR Section 5.6 to develop, prior to receipt of high-level radioactive waste and spent nuclear fuel, plans and procedures for the conduct of normal operations, maintenance, surveillance, and periodic testing of structures, systems, and components, and processes, based on the preclosure safety analysis and the total system performance assessment. As described in SAR Section 5.6.1, these plans and procedures will include procedural safety controls.

Once the detailed inspection program is developed for PSC-9, it will include a full description of the analytical process and equipment that will be used to measure the dissolved boron concentration and confirm the ^{10}B enrichment. Such processes are available and are currently in use. For example, boron concentration can be measured using mannitol potentiometric titration, and ^{10}B enrichment can be measured using inductively coupled mass spectrometry or thermal ionization mass spectrometry.

SAR Section 5.10.2 describes probable subjects of license specifications which impose conditions or limitations on operations that are consistent with the preclosure safety analysis, including procedural safety controls. Proposed license specifications include limiting conditions for operations, which establish limits on process parameters that are required to be confirmed during various modes of operations. SAR Table 5.10-1 specifically lists a limiting condition for operations for soluble boron concentration control (corresponding to PSC-9 of SAR Table 1.9-10) that must be established and maintained within a specific range of values.

As discussed in SAR Section 1.2.5.1.4, the pool water treatment and cooling system operating procedure will require periodic sampling of the pool water boron concentration. The frequency of sampling will be established to ensure that a minimum boron concentration of 2,500 mg/L is maintained between samplings. The procedure will also require sampling following events that could significantly impact the concentration of boron in the pool. As discussed in SAR Section 1.2.5.3.2, the concentration of boron in the Wet Handling Facility pool is maintained at or above

2,500 mg/L through manual operation of the boric acid makeup subsystem. The boric acid solution is added to the treated water from the pool water treatment subsystem to replenish any losses. The losses and additions are trended, and the frequency of addition is modified to appropriately maintain the required minimum boron concentration.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

RAI Volume 2, Chapter 2.1.1.7, Seventh Set, Number 3:

> Explain how the double contingency principle is used to maintain sub-criticality during the GROA preclosure operations (SAR Section 1.14). In addition, provide justification that the design of the surface facilities is based on the double contingency principle. Explain why the design of the subsurface facilities was not based on this principle (BSC, 2008, *Preclosure Criticality Analysis Process Report,* p. 20).
>
> SAR Section 1.14.2.2 refers to the preclosure criticality safety analysis documented in (SNL 2008, *Preclosure Criticality Analysis Process Report*) for the detailed preclosure safety analysis process. In Section 2.1.2 of that report, DOE states that the design of the GROA surface facilities is based on the double contingency principle, as defined by ANSI/ANS-8.1-1998. However, DOE has not provided information on how, and the extent to which, this principle is used in the design of the GROA facilities. This information is needed to demonstrate compliance with 10 CFR 63.112(e)(6).

1. RESPONSE

The geologic repository operations area (GROA) waste handling facilities as well as spent nuclear fuel (SNF) and high-level radioactive waste (HLW) canisters are designed to meet the criticality safety requirements described in SAR Section 1.14.2.1. The process evaluation, as part of the criticality safety analysis, demonstrates that the design of GROA facilities and canisters in conjunction with the operational requirements are sufficiently robust and independent such that criticality safety is maintained for normal operations and for Category 1 and Category 2 event sequences (i.e., event sequences with a mean probability of occurrence greater than or equal to one chance in 10,000 prior to permanent closure).

The design of GROA waste handling facilities, the design and loading criteria of SNF and HLW canisters, and the operational requirements for the GROA utilize the double contingency principle, which is a design principle as described in Section 4.2.2 of ANSI/ANS-8.1-1998 that states:

> Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent and concurrent changes in process conditions before a criticality accident is possible.

The double contingency principle is discussed with respect to the design of surface facilities in Section 2.1.2 of the *Preclosure Criticality Analysis Process Report* (BSC 2008a), which states:

> …the goal is accident prevention and the Double Contingency Principle provides important guidance in achieving this goal. The design of the surface facilities is based on the principles of double contingency; whereas, the quantitative event sequence-based analysis demonstrates compliance with 10 CFR Part 63 (i.e., all operations shall be determined to be subcritical for normal operations and for

individual event sequences with a mean probability of occurrence greater than or equal to one chance in 10,000 prior to permanent closure).

Although this excerpt discusses the double contingency principle with respect to the surface facility design, exclusion of other facilities is not intended by the statement. In fact, the double contingency principle is utilized for intrasite operations, including the aging facility and subsurface facility operations, and is also utilized in the design of SNF and HLW canisters.

Criticality safety is maintained for GROA operations through controlling multiple parameters including moderator, interaction, and soluble neutron absorber. These parameters are controlled using multiple, redundant, independent, and robust engineered systems such that the process evaluation not only demonstrates that the double contingency principle is met, but also that event sequences that impact a criticality control parameter are not credible (i.e., frequency of occurrence is less than 1×10^{-6} per year). In addition, defense-in-depth structures, systems, and components (SSCs), such as fixed neutron absorbers, are required in SNF canisters and the Wet Handling Facility (WHF) staging racks, even though they are not relied upon to demonstrate subcriticality.

As requested during the clarification call on May 7, 2009, Section 1.1 provides an example of how the double contingency principle is utilized in the criticality safety design criteria and process evaluations for GROA operations. Section 1.2 provides an overview of the defense-in-depth in the criticality safety design criteria, as well as the design of SSCs and their reliability requirements to ensure prevention of an inadvertent criticality.

1.1 EXAMPLE OF DOUBLE CONTINGENCY PRINCIPLE CONSIDERATION IN ESTABLISHING CRITICALITY SAFETY DESIGN CRITERIA AND PROCESS EVALUATION FOR GROA OPERATIONS

The example presented in this section is based on the operation described in SAR Section 1.14.2.4, which provides the criticality safety analysis for the transfer of a pressurized water reactor (PWR) transportation, aging, and disposal (TAD) canister from a transportation cask to an aging overpack, from a transportation cask to a waste package, or from an aging overpack to a waste package, using a canister transfer machine in a Canister Receipt and Closure Facility (CRCF). The event sequence analysis associated with this example is presented in SAR Section 1.7.5 and illustrated in SAR Figure 1.7-2.

The criticality sensitivity calculations presented in SAR Section 1.14.2.4.1 demonstrate that there is no criticality potential in a PWR TAD canister without moderator intrusion inside the TAD canister, irrespective of fuel and basket geometry, fixed neutron absorber efficacy, or reflection conditions. Because the criticality calculations considered bounding fuel characteristics as described in SAR Section 1.14.2.3.2.1, all potential loading configurations are bounded by the calculations; therefore, there is no potential for a PWR assembly misload in a TAD canister that could lead to a criticality. Reliance on geometry control and fixed neutron absorbers, similar to a

10 CFR Part 71 analysis with which the TAD canister must comply prior to receipt at the GROA, is not sufficient to demonstrate subcriticality for GROA operations without moderator control because of the conservatism in the SNF characteristics[1] assumed in the criticality safety analysis. Therefore, the only criticality parameter that must be controlled for the PWR TAD canister transfer operation is moderator.

SAR Figure 1.14-4 demonstrates that the maximum safe (i.e., k_{eff} less than the upper subcritical limit) allowable moderator volume inside a PWR TAD canister is a function of geometry and fixed neutron absorber efficacy, and it ranges between approximately 200 and 400 liters. To ensure that subcriticality is maintained with a significant margin, the criticality safety design criterion is that moderator must be prevented from entering a PWR TAD canister. The guidance provided in ANSI/ANS-8.22, *Nuclear Criticality Safety Based on Limiting and Controlling Moderators*, is followed in establishing the specific design requirements to ensure that moderator is controlled for this operation. Specifically, Section 5, "Engineered Practices for Moderator Control Areas," of ANSI/ANS-8.22 states:

> **5.1 Moderator Control Area Barriers.** Moderator control areas shall be provided with engineered barriers as required by the process evaluation. Consideration should be given to potential hazards external to the moderator control areas that could compromise the integrity of the engineered barriers.
>
> **5.2 Equipment and Containers.** Fissile material processing equipment and containers used in moderator controlled areas shall be designed, constructed, and maintained to limit and control moderators in accordance with the process evaluation.

Consistent with the guidance provided in the excerpt above, moderator control is required to be provided by the PWR TAD canister design and associated containment (i.e., transportation cask, aging overpack, or waste package), the handling equipment, and the CRCF design such that at least two independent systems would need to fail before moderator control is compromised. Based on the CRCF design and handling equipment associated with PWR TAD canister transfer operations, moderator could be available from handling equipment lubricating fluids, fire suppression system actuation, or water system piping failure.

The process evaluation documented in *Canister Receipt and Closure Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008b) considers the process changes associated with mechanical and thermal challenges that could impact moderator control. Process changes

[1] All fuel assemblies are conservatively represented as fresh fuel enriched to 5 wt% ^{235}U, and fuel rods are modeled as simple cylinders without correction for dished ends of pellets and without credit for burnable poison or assembly hardware (e.g., spacer grids). In addition, all PWR assemblies are modeled as Westinghouse 17 × 17 or Babcock & Wilcox 15 × 15 fuel assemblies, which are shown to be the most reactive designs based on a survey of commercial SNF assemblies in various potential preclosure configurations (SAR Section 1.14.2.3.2.1.1).

associated with seismic event sequences are evaluated in *Seismic Event Sequence Quantification and Categorization Analysis* (BSC 2008c).

Human error and equipment failure induced mechanical challenges to which a PWR TAD canister can be exposed during this operation are illustrated in SAR Figure 1.7-2, which include:

- Collision or impact to canister
- Impact associated with transportation cask or aging overpack lid removal
- Drop of object (e.g., grapple) onto canister
- Canister impact due to movement of canister transfer machine, canister transfer trolley, waste package transfer trolley, or site transporter during lift
- Canister dropped at operational height
- Canister dropped above operational height
- Canister dropped inside canister transfer machine.

The mean probability of a TAD canister breach, considering the mechanical challenges listed above, is 1 × 10−4 in the three CRCFs during the preclosure period (BSC 2008b, Table G-2, Event Sequence ID ESD09-TAD-SEQ3-RRF, End State Filtered radionuclide release). Taking into account the concurrent or sequential failure that could result in moderator availability, the mean probability of failure to control moderator for this operation decreases to 1 × 10−8 during the preclosure period (BSC 2008b, Table G-2, Event Sequence ID ESD09-TADS-EQ4-RRC, End State RR-FILTERED-ITC). Note that this event sequence conservatively considers only the availability of moderator, irrespective of whether moderator can enter a breached TAD canister in sufficient quantities (i.e., hundreds of liters).

The fire analysis for the CRCF is provided in Appendix F of *Canister Receipt and Closure Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008b). The mean probability of a TAD canister breach due to a fire in the three CRCFs is 8×10^{-6} during the preclosure period (BSC 2008b, Table G-2, Event Sequence Group ID ESD20-TAD-SEQ5-RRU, End State Unfiltered radionuclide release). Even though moderator entry into breached canisters during fire event sequences is not physically realizable because of a combination of breach mechanisms, building and equipment configuration, and overpack material properties, the concurrent or sequential failure that could result in moderator availability and for that moderator to enter a breached canister has a conservatively assessed probability of 1×10^{-3} (BSC 2008b, Section 6.2.2.9.1). Furthermore, if moderator is available from the fire suppression system, then the thermal challenge to the TAD canister due to the fire would not be sufficient to cause a breach. Therefore, the mean probability of failure to control moderator for this operation due to a fire is less than 8×10^{-9} during the preclosure period (BSC 2008b, Table G-2, Event Sequence Group ID ESD20-TAD-SEQ6-RRC, End State RR-UNFILTERED-ITC).

Seismic event sequences involving a TAD canister in the CRCFs are summarized in Table 6.6-8 of *Seismic Event Sequence Quantification and Categorization Analysis* (BSC 2008c). The mean probability of the most limiting seismic event sequence that results in a TAD canister breach in the CRCF is 6×10^{-5} during the preclosure period (BSC 2008c, Table 6.6-8, Event Sequence ID CRCF-S-IE-TAD-AO 11-05, End State RR-UNFILTERED). The potential availability of moderator during a seismic event affecting the CRCF and for that moderator to enter a breached TAD canister has an assessed mean probability of 1×10^{-2} (BSC 2008c, Section 6.2.2.23). Therefore, taking into account the concurrent or sequential failure that could result in moderator availability and for that moderator to enter a breached canister, the mean probability of failure to control moderator for this operation due to a seismic event is less than 1×10^{-6} during the preclosure period.

Based on the process evaluation described above, the SSCs that are relied upon to provide the necessary moderator controls for this operation are identified as important to safety (ITS) with a specific reliability requirement. These SSCs and their reliability requirements are provided in SAR Table 1.9-3, which includes but is not limited to:

- ITS SSCs relied upon to reduce the probability of TAD canister breach:
  - The mean conditional probability of breach of a canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.
  - The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.
  - The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.
  - The mean conditional probability of breach of a canister contained within a waste package resulting from the spectrum of fires shall be less than or equal to 3×10^{-4} per fire event.
  - The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.
  - The mean conditional probability of breach of a canister located within the aging overpack resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.
  - The mean conditional probability of breach of a canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.

- The mean probability of a spurious movement of the canister transfer machine while a canister is being lifted or lowered shall be less than or equal to 7×10^{-9} per transfer for each canister transfer machine.

- Closure of the canister transfer machine slide gate shall be incapable of breaching a canister.

- The mean frequency of collapse of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 1×10^{-5} per year.

- The mean frequency of a hoist system failure of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.

- The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2×10^{-6} per transfer.

- The canister transfer machine shall preclude non-flat-bottom drops of canisters.

- The mean frequency of collapse of CRCF structure due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.

- ITS SSCs relied upon to limit moderator availability:

  - The mean probability of inadvertent introduction of fire suppression water into a canister shall be less than or equal to 1×10^{-6} over a 720-hour period following a breach.

  - The mean probability of inadvertent introduction of an oil moderator into a canister shall be less than or equal to 9×10^{-5} over a 720-hour period following a breach.

All other operations in the GROA are treated in the same manner as provided in this example in establishing criticality safety controls and design criteria, performing a thorough process evaluation, and ensuring the availability of safety systems through the identification of ITS SSCs.

1.2 OVERVIEW OF THE DEFENSE-IN-DEPTH IN THE CRITICALITY SAFETY DESIGN CRITERIA

Criticality safety is maintained for GROA operations through controlling multiple parameters including moderation, interaction, and soluble neutron absorber. As described below, these parameters are controlled using multiple, redundant, independent, and robust engineered systems, such that the process evaluation not only demonstrates that the double contingency principle is met, but also that event sequences that impact a criticality control parameter are not credible (i.e., frequency of occurrence is less than 1×10^{-6} per year). Defense-in-depth is evident through the low calculated joint probability for such event sequences and criticality safety design features with significant safety margins.

For operations in the CRCF, the Initial Handling Facility (IHF), and the Receipt Facility (RF), as well as dry operations in the Wet Handling Facility (WHF), moderator is controlled based on ITS sealed robust canisters, reliable ITS cranes with lubricating oil retention features, physical layouts that limit water piping in waste handling areas, and a double-interlocked preaction fire suppression system. The nuclear safety design bases for these SSCs and their ITS functions are found in SAR Tables 1.9-3 through 1.9-8. Taking into account the availability of moderator and for that moderator to enter a breached canister, the mean probability of the limiting event sequence that results in failure to control moderator is 1.18×10^{-8} for the three CRCFs (BSC 2008b, Table G-4), 4×10^{-6} for the IHF (BSC 2008d, Table G-4), 2.09×10^{-7} for the WHF (BSC 2008e, Table G-4), and 6×10^{-7} for the RF (BSC 2008f, Table G-4) for the preclosure period. These probability values exclude event sequences for HLW glass canisters because, as stated in SAR Section 1.14.2.3.2.4, there is no criticality potential with HLW even in the presence of moderator.

For intrasite operations, including the aging facility, moderator is controlled based on ITS sealed canisters inside ITS sealed transportation casks or inside ITS aging overpacks. Taking into account the availability of moderator and for that moderator to enter a breached canister, the mean probability of the limiting event sequence that results in failure to control moderator for intrasite operations is 1×10^{-6} for the preclosure period (BSC 2008g, Table G-4).

For the subsurface facility, moderator is controlled based on sealed ITS canisters inside sealed ITS waste packages inside an ITS transport and emplacement vehicle while being transported to the emplacement drifts or on a pallet while in the drifts. Taking into account the availability of moderator and for that moderator to enter a breached canister, the mean probability of the limiting event sequence that results in failure to control moderator for subsurface facility operations is 1×10^{-9} for the preclosure period (BSC 2008h, Table G-4).

For seismic event sequences for the GROA facilities and taking into account common cause failures that could result in a canister breach and availability of moderator to enter a breached canister, the mean probability of the limiting event sequence that results in failure to control moderator for the GROA facilities is less than 1×10^{-6} for the preclosure period (BSC 2008c, Section 6).

For controlling interaction between DOE SNF canisters, ITS staging racks are designed and an ITS canister transfer machine interlock is implemented so that no more than one DOE SNF canister can be placed outside of the staging racks or a codisposal waste package, even though the criticality safety criterion is that no more than four DOE SNF canisters can be placed outside of the staging racks or a codisposal waste package (BSC 2008b, Table 6.0-2).

To control interaction between three or more naval SNF canisters, the IHF is designed such that no more than two naval canisters can be present in the IHF. Therefore, violating this control is precluded through physical barriers (BSC 2008d, Table 6.0-2).

For wet operations, the minimum required soluble boron concentration in the WHF pool is sufficient to maintain subcriticality for normal operations and end-states of event sequences, without credit for fixed neutron absorbers and under hypothetical dilution conditions (see response to RAI 2.2.1.1.3-3-020 for additional detail). Nonetheless, as an additional defense-in-depth measure, neutron absorbers and fuel baskets that provide separation are required for the staging racks, as well as SNF canisters.

In summary, the mean probability of the limiting event sequence that impacts a criticality control parameter for GROA operations is 4×10^{-6} for the preclosure period (BSC 2008d, Table G-4), which is at least an order of magnitude less than the Category 2 screening criterion. Such an event sequence does not necessarily result in a configuration with a k_{eff} that exceeds the upper subcritical limit, even with the bounding SNF characteristics assumed in the criticality safety analysis. In addition, to comply with transportation and postclosure criticality requirements, additional controls are imposed on SNF canisters, including fixed neutron absorbers (e.g., borated stainless steel plates) and loading criteria (e.g., burnup credit loading curves) such that these canisters would remain subcritical even with moderator intrusion. These additional controls are not credited in the preclosure criticality safety analysis and are considered defense-in-depth controls.

The double contingency principle is embodied in the criticality safety design criteria for GROA waste handling facilities and design of SNF canisters. As stated in Section 2.1.2 of the *Preclosure Criticality Analysis Process Report* (BSC 2008a), the quantitative event sequence-based analysis demonstrates compliance with 10 CFR Part 63. GROA operations have been shown to be subcritical for normal operations and for individual event sequences with a mean probability of occurrence greater than or equal to one chance in 10,000 prior to permanent closure. The preclosure criticality safety analysis presented in SAR Section 1.14, in conjunction with the process evaluation presented in SAR Sections 1.6 and 1.7 and the identification of ITS SSCs and procedural safety controls in SAR Section 1.9, give quantitative and probabilistic credence to the inherent use of the double contingency principle utilized in the design.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

ANSI/ANS-8.1-1998. *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. La Grange Park, Illinois: American Nuclear Society.

ANSI/ANS-8.22-1997. *American National Standard for Nuclear Criticality Safety Based on Limiting and Controlling Moderators*. La Grange Park, Illinois: American Nuclear Society.

BSC (Bechtel SAIC Company) 2008a. *Preclosure Criticality Analysis Process Report*. TDR-DS0-NU-000001 REV 03. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080220.0001.

BSC 2008b. *Canister Receipt and Closure Facility Reliability and Event Sequence Categorization Analysis*. 060-PSA-CR00-00200-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080311.0031.

BSC 2008c. *Seismic Event Sequence Quantification and Categorization Analysis*. 000-PSA-MGR0-01100-000-00A CACN 001. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080311.0032.

BSC 2008d. *Initial Handling Facility Reliability and Event Sequence Categorization Analysis*. 51A-PSA-IH00-00200-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080312.0031.

BSC 2008e. *Wet Handling Facility Reliability and Event Sequence Categorization Analysis*. 050-PSA-WH00-00200-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080312.0033.

BSC 2008f. *Receipt Facility Reliability and Event Sequence Categorization Analysis*. 200-PSA-RF00-00200-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080312.0030

BSC 2008g. *Intra-Site Operations and BOP Reliability and Event Sequence Categorization Analysis*. 000-PSA-MGR0-00900-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080312.0032.

BSC 2008h. *Subsurface Operations Reliability and Event Sequence Categorization Analysis*. 000-PSA-MGR0-00500-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080312.0034.

RAI Volume 2, Chapter 2.1.1.7, Seventh Set, Number 4:

> Provide documentation detailing the applicant's MCNP (Monte Carlo N-Particle) code validation, including the critical experiments benchmarked against, for CSNF and DOE SNF, as per Reg. Guide 3.71. Ensure that the effects of using high concentrations of enriched boron are considered in the range of applicability of the benchmark experiments used in validation (SAR Section 1.14.2.3.4.1). This information is needed to demonstrate compliance with 10 CFR 63.112(e)(6).

1. RESPONSE

The documentation for Monte Carlo N-Particle (MCNP) code validation, consistent with the guidance of Regulatory Guide 3.71, including applicable critical benchmark experiments for commercial spent nuclear fuel (SNF) preclosure configurations, is provided in *Bias and Range of Applicability Determinations for Commercial Nuclear Fuels* (BSC 2008a). The effects of using enriched boron on the range of applicability of the benchmark experiments used in the validation are discussed in Section 6.3 of *Bias and Range of Applicability Determinations for Commercial Nuclear Fuels* (BSC 2008a).

The documentation for MCNP code validation, consistent with the guidance of Regulatory Guide 3.71, including applicable critical benchmark experiments for DOE SNF preclosure configurations, is provided in *Bias Determination for DOE Nuclear Fuels* (BSC 2008b). This document references *Analysis of Critical Benchmark Experiments and Critical Limit Calculation for DOE SNF* (BSC 2003) for the calculations of the bias and bias uncertainty values for the groups of benchmark experiments applicable to each specific DOE SNF type. The benchmark experiments, along with their ranges of applicability, are provided in *Benchmark and Critical Limit Calculation for DOE SNF* (BSC 2002).

Benchmark and Critical Limit Calculation for DOE SNF (BSC 2002) and *Analysis of Critical Benchmark Experiments and Critical Limit Calculation for DOE SNF* (BSC 2003) were provided with the response to RAI 3.2.2.1.2.1-4-023. *Bias and Range of Applicability Determinations for Commercial Nuclear Fuels* (BSC 2008a) and *Bias Determination for DOE Nuclear Fuels* (BSC 2008b), including the associated MCNP input and output files, are provided with this response.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2002. *Benchmark and Critical Limit Calculation for DOE SN*F. CAL-EDC-NU-000008 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20020416.0053.

BSC 2003. *Analysis of Critical Benchmark Experiments and Critical Limit Calculation for DOE SNF*. CAL-DSD-NU-000003 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20030724.0002.

BSC 2008a. *Bias and Range of Applicability Determinations for Commercial Nuclear Fuels.* 000-00C-MGR0-04700-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080222.0014.

BSC 2008b. *Bias Determination for DOE Nuclear Fuels.* 000-00C-MGR0-04800-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080225.0028.

RAI Volume 2, Chapter 2.1.1.7, Seventh Set, Number 5:

> Justify that the values for fissile material concentrations reported in SAR Table 1.14-1 for Savannah River site HLW are those that are actually used by the vitrification plant staff for criticality control. This information is needed to demonstrate compliance with 10 CFR 63.112(e)(6).

1. RESPONSE

As discussed during the clarification call on May 7, 2009, the intent of the RAI is to verify that the license application contains waste acceptance provisions to ensure that preclosure criticality safety will be maintained with potential changes to the composition of vitrified high-level radioactive waste (HLW) glass. SAR Table 1.14-1 provides the fissile material concentration values for four HLW canister types based on data provided by the waste custodians. The fissile material concentration values provided in SAR Table 1.14-1 are also consistent with the HLW composition provided in SAR Table 1.5.1-21.

The discussion of criticality control parameters for HLW presented in SAR Section 1.14.2.3.2.4 demonstrates that the fissile material concentration in HLW glass (for four HLW glass types) is approximately an order of magnitude lower than a single parameter limit based on moderated systems. If a waste custodian increases the fissile material concentration in the vitrified glass beyond what has been analyzed and licensed, then the criticality safety analysis will be updated to determine acceptability of the HLW glass from a preclosure criticality safety perspective based on the criticality safety requirements in SAR Section 1.14.2.1. This commitment is provided in SAR Table 5.10-3, which describes the waste form and waste package qualification program. Waste acceptance criteria include such criticality safety requirements as waste form, physical, chemical, and nuclear characteristics (e.g., geometries, fissile material content, burnup). SAR Sections 1.8, 1.14, 2.3.7, and 2.2.1.4.1 describe the methodology and analyses required to confirm that waste forms are enveloped by the preclosure safety analysis and postclosure performance assessment. This administrative control will require that similar analyses be completed prior to receiving individual waste forms or waste package designs that are not explicitly analyzed in the license application.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.