From: Peter Tam
To: Amy C. Hazelhoff; Jesus Arias; Laurie A. Lahti
Date: 02/26/2007 11:54:18 AM
Subject: Palisades - Draft RAI on Proposed AST Amendment (TAC MD3087)

Laurie:

This is regarding your application for amendment dated 9/25/05 (alternative source term). Our dose analysis reviewer has the following draft RAI questions that he would like to discuss with you in a conference call:

1. The AST dose consequence analyses are based on a power level of 2703 megawatt thermal (MWt) representing the original full power design rating of 2650 MWt, with a 2% margin for uncertainty. It appears that certain parameters used in the AST evaluations may be based on information from tables in the final safety analysis report (FSAR) with listed power levels other than 2703 MWt.

RAI 1.1 Please provide additional information to clarify whether the 2703 MWt power level used in the AST dose consequence analyses for the evaluation of the radiological source term, applies to all other power related aspects of the evaluations.
RAI 1.2 Please provide additional information to clarify the limiting power level for each of the AST dose consequence analyses considering all power related aspects of the evaluations.

2. The staff will address the acceptability of the retention of the TID-14844 source term for both equipment qualification (EQ) purposes and for NUREG-0737 analyses other than for the control room envelope (CRE), based on the resolution of Generic Issue 187 and the logic provided in Enclosure 1 of the license amendment request (LAR). However, the staff is concerned that such an action may be interpreted as an acceptability for these analyses at the AST power level, especially if the AST power level exceeds the power level used in the EQ and NUREG-0737 analyses. Therefore:

RAI 2.1 Please verify the core thermal power level and the associated uncertainty used in the licensing basis EQ dose evaluations and the relationship, if any, to the power level used in the AST LAR.

RAI 2.2 Please verify the core thermal power level and the associated uncertainty used in the licensing basis NUREG-0737 dose evaluations and the relationship, if any, to the power level used in the AST LAR.

3. The loss-of-coolant accident (LOCA) analysis credited an elemental iodine wall deposition removal coefficient of 2.3 hr -1 for the duration of the accident. EA-PAH-91-06, Revision 2, Fission Product Removal Coefficients for Design Basis Radiological Consequence Analyses, September 2006, was cited as the reference for the elemental iodine wall deposition removal coefficient of 2.3 hr -1.

RAI 3.1 The cited reference is not readily available to the staff. Please provide additional information describing the technical basis for the elemental iodine wall deposition removal coefficient of 2.3 hr -1.

RAI 3.2 Please provide additional information describing the technical justification for the use of the elemental iodine wall deposition removal coefficient of 2.3 hr -1 for the duration of the accident. In particular, please provide additional information describing the technical justification for the application of this coefficient during the period in which credit for elemental iodine removal from the operation of the containment sprays is taken currently.

RAI 3.3 The staff notes that the aerosol natural deposition coefficient of 0.1 hr -1 is specified only for the time period after containment spray credit ends. Please provide additional information describing the differences in the application of the natural removal mechanisms and their relationship to the containment spray removal assumptions. Also, please note that the table on page 17 of NAI-1149-014 Revision 3, describing the timing of the credited aerosol natural deposition coefficient of 0.1 hr -1, appears to contain a typographical error in the title "Particulate Spray Removal Coefficients."

4. Containment purge is not considered as a means of combustible gas or pressure control in the AST LOCA analysis. In addition, routine containment purging is not active for the AST LOCA analysis.

RAI 4. Please provide additional information describing the controls that are in place to preclude the use of on-line containment purging and containment purging for post LOCA hydrogen control.

5. Regulatory Guide (RG) 1.183, Appendix A, Regulatory Position 3.3 states that, "The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown." The AST LAR references Amendment No. 31 and states that, "Per the current licensing basis, there is at least 90% spray coverage of the containment (Reference 5.30); therefore, the containment is treated as a single well mixed volume." The safety evaluation (SE) for Amendment No. 31 used a containment free air volume of 1.64 million cubic feet and sprayed volume of 1.48 million cubic feet, thereby establishing the 90% spray coverage. However, the SE for Amendment No. 31 also assumed an air exchange between the unsprayed and sprayed regions of two unsprayed region volumes per hour.

RAI 5. Please provide additional information addressing the mechanisms credited for providing adequate mixing of the unsprayed compartments to substantiate the assumption of a single, well-mixed containment building atmosphere.

6. The PNP LOCA analysis credits a 50% reduction of the emergency core cooling system (ECCS) leakage into the auxiliary building as per current design basis. The staff notes that Table 9.0-1 of the SE supporting license Amendment No. 31, Section A, LOCA, Item 11, states, "Iodine plateout factor due to high-radiation trip of engineered safety feature (ESF) cubicles ventilation system if significant leakage occurred: 2." If the basis for the 50% reduction for plateout is due to the lack of ventilation in the cubicle area, the staff would assume that the ESF leakage would be evaluated using ground level meteorology. Based on an examination of Table 1.8.1-2 of the technical report and Section 6.2.6 of NAI-1149-014 Rev. 3, it appears that the ESF leakage into the auxiliary building is being evaluated as a stack release.

RAI 6. Please provide additional information on the basis for the 50% reduction of the ECCS leakage into the auxiliary building and the relationship of this credited reduction to the assumed stack release pathway.

7. The AST fuel handling accident (FHA) evaluation assumes a minimum water cover depth of 22.5 feet over the damaged fuel and adjusts the allowed iodine decontamination factor accordingly. Numerical Applications, Inc. Calculation Number: NAI-1149-016 Rev. 1, Palisades Design Basis Fuel Handling Accident AST Radiological Analysis, states that, "Per the inputs listed in Reference 5, 22.5 feet of water will be maintained above the damaged fuel; therefore, the decontamination factor for elemental iodine must be adjusted."

RAI 7.1 The cited reference, [Email Jeffery Voskuil (Nuclear Management Company) to Jim Harrell (Numerical Applications, Inc.)], dated June 28, 2004, Subject: FHA Inputs], is not readily available to the staff. Please provide additional information describing the basis for the 22.5 feet of water cover used in the FHA analysis.

RAI 7.2 Pleases provide additional information describing the differences in the water cover assumptions for the FHA vs the cask drop accident.

8. RG 1.183, Appendix F, Regulatory Position 5.3, states that, "The primary-to-secondary

leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." The AST steam generator tube rupture (SGTR) analysis assumes that the release of radioactivity from both the ruptured steam generator (SG) and the unaffected SG continues for 8 hours, until shutdown cooling is in operation, and steam releases from the steam generators have been terminated. The AST main streamline break (MSLB) analysis assumes that both primary-to-secondary leakage and releases from the faulted SG continue for 12 hours at which time the temperature of the leakage is projected to be less than 100°C (212°F) and the faulted SG is completely isolated. Both analyses assume that the release of radioactivity from the unaffected SG continues for 8 hours until shutdown cooling is in operation and releases from the unaffected steam generator have been terminated.

RAI 8. Please provide additional information describing the difference in the transport assumptions used in the MSLB and the SGTR accidents regarding the time duration for releases from the affected SG, i.e., the ruptured SG in the SGTR and the faulted SG in the MSLB.

9. In the MSLB, SGTR and control rod ejection (CRE) analyses, the time specified to establish shutdown cooling is 8 hours. In the MSLB, the time specified for the cessation of both primary-to-secondary leakage and releases from the faulted SG is 12 hours, at which time the temperature of the leakage is projected to be less than 100°C (212°F) and the faulted SG is completely isolated.

RAI 9.1 Please provide additional information to verify that the 8-hour time period for alignment to residual heat removal (RHR) is based on the time required to reduce the system heat load to the point where the RHR system can remove all the decay heat using only safety grade equipment.

RAI 9.2 Please provide additional information to verify that the 12-hour time period for the cessation of both primary-to-secondary leakage and releases from the faulted SG in the MSLB analysis, is based on the time required to reduce the temperature of the primary-to-secondary leakage to less than 100°C (212°F), using only safety grade equipment.

10. In Section 1.6.3 of the AST Technical Report, "Control Room Heating, Ventilation, and Air-Conditioning System Description," the net volume of the CRE is given as 76,451 ft3. The control room (CR) volume used in the inhalation dose consequence analyses, as shown in Table 1.6.3-1 of the AST Technical Report, is 35,923 ft3.

RAI 10.1 Please provide additional information describing the basis for the use of the CR volume of 35,923 ft3, as opposed to the CRE volume of 76,451 ft3 in the CR inhalation dose consequence analyses.

RAI 10.2 Please provide additional information defining the boundaries of the CR proper, the CRE, and the CR/technical support center (TSC) envelope if the latter is a separately designated area.

11. For the LOCA CR habitability analysis, the LOCA analysis assumes an unfiltered in-leakage of 10 cfm after CR isolation. Page 9.8-12 of the FSAR, Section E, Control Room/TSC Envelope, states, "Four vestibules are used to provide egress and ingress to the control room/TSC during post-accident operations. These vestibules are adjacent to Doors 108, 115, 175 and 52. Their function is to prevent air in-leakage."

RAI 11.1 Please provide additional information describing the area for which the 10 cfm unfiltered in-leakage restriction is to apply.

RAI 11.2 Please provide a plan view of the boundary of the area to which the 10 cfm unfiltered in-leakage restriction is to apply. Please indicate all doorways into the area showing that they are equipped with double door vestibules to preclude unfiltered in-leakage from ingress/egress.

12. The term reactor building is used the technical report NAI-1149-027 in the description of the small line break outside containment (SLBOC). FSAR Section 14.23, describing the SLBOC, uses the term auxiliary building.

RAI 12. Please provide clarification of the use of the terms reactor building versus auxiliary building at PNP.

This e-mail aims solely to prepare you and others for the proposed conference call. It does not convey an NRC staff position, and it does not formally request for additional information. We will discuss disposition of the above draft RAI questions in the conference call.

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