Jeff Ciocco

From: Jeff Ciocco
Sent: Thursday, July 24, 2008 11:01 AM
To: us-apwr-rai@mhi.co.jp
Cc: Michelle Hart; Charles Cox; Peter Hearn; Michael Takacs; Larry Burkhart; Harrison Botwin
Subject: US-APWR Design Certification Application RAI No. 38
Attachments: US-APWR DC RAI 38 RSAC 412.pdf

MHI,

Attached please find the subject request for additional information (RAI). This RAI was sent to you in draft form. The schedule we are establishing for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. As requested, responses to questions 20 and 21 may take 90 days. For any of the other RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule. Please submit your RAI response to the NRC Document Control Desk.

Thanks,

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REQUEST FOR ADDITIONAL INFORMATION NO. 38 REVISION 1

7/24/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors

Application Section: DCD 15

QUESTIONS

15.00.03-1

Please identify the dose conversion factors (DCFs) from FGR-11 Table 2.1 that were used to calculate dose equivalent iodine-131 (DEI-131) nuclide-specific reactor coolant concentrations; committed effective dose equivalent (CEDE) or thyroid. Justify which DCFs were used.

15.00.03-2

Provide the calculation of the iodine appearance rates listed in DCD Table 15.0-11, including the basis for the inputs and assumptions. Explain why the iodine appearance rate varies between the three accidents identified in the table.

15.00.03-3

Identify the dose conversion factors from FGR-12 Table III.1(effective dose equivalent or organ dose equivalent) which were used to calculate dose equivalent Xe-133 (DE Xe-133) nuclide-specific reactor coolant concentrations. Justify which DCFs were used.

15.00.03-4

Provide the assumed initial reactor coolant system fluid volume for the design basis accidents (DBAs) that model leakage to the secondary system.

15.00.03-5

Justify the moisture carryover particulate partition coefficient of 1000 used in the reactor coolant pump (RCP) rotor seizure accident and steam generator tube rupture (SGTR) dose analyses. Provide the amount of moisture carryover assumed in these analyses and the basis for the assumption. Please ensure to describe the calculation used.

15.00.03-6

Provide the basis for assuming 0.25% of the core fuel melts as a result of the rod ejection accident.

15.00.03-7

On page 15.4-74, it states that SPR 4.2, Appendix B, was considered in determining the transient fission gas release for the rod ejection accident. Provide the calculation of the transient fission gas release from the fuel rods for the rod ejection accident.

15.00.03-8

For the rod ejection accident, 10 percent of the fuel in the core experiences departure from nucleate boiling and is assumed to have cladding failure with a gap fraction of 10% for iodines and noble gases. Assuming cladding failure with a gap fraction of 10% for iodines and noble gases, and given the calculated transient release of 11% of the iodines and noble gases, compare the total release to a total release from the damaged non-melted fuel of 21% of the iodines and noble gases.

15.00.03-9

Iodine and aerosol removal in containment by spray and natural deposition is discussed in some detail in Section 6.5.2 of the DCD. For clarity, refer to this information in DCD 15.4.8.5 as it relates to the rod ejection accident.

15.00.03-10

Provide the basis for the assumed break flow rate of 97 gallons per minute (gpm) for the sample line break dose analysis.

15.00.03-11

Iodine and aerosol removal in containment by spray and natural deposition are discussed in additional detail in Section 6.5.2 of the DCD. For clarity, refer to this information in DCD Sections 15.6.5.5.1.1, "LOCA Consequence Model" and 15A.1.2, "Airborne Radioactivity Removal Coefficients."

15.00.03-12

Provide the values of the elemental iodine and particulate spray removal coefficients used in the LOCA dose analysis. Provide the time at which the containment spray is no longer operating. Confirm that the spray removal is only credited in the assumed sprayed region of containment.

15.00.03-13

Inputs and assumptions for the sprayed volume of containment, mixing rate between the sprayed and unsprayed regions of containment, elemental iodine and particulate spray removal coefficients, elemental iodine deposition coefficient, time when elemental removal ends, time when containment spray operation ends, and whether elemental iodine removal and/or aerosol removal are credited in the sprayed and unsprayed regions of containment should be added to DCD Table 15.6.5-4 for completeness.

15.00.03-14

In regard to the LOCA dose analysis engineered safety feature (ESF) recirculation system leakage, you state that the ESF systems leakage is taken as two times the sum of the simultaneous leakage from all ESF recirculation systems components above which the Technical Specifications would require such systems inoperable. The Technical Specifications do not appear to include such ESF systems leakage requirements. Please provide the basis for the assumed ESF leakage rate of 17.6 pounds per hour (lb/hr).

15.00.03-15

In regard to the LOCA dose analysis for ESF recirculation system leakage, you state that all fission products released from the fuel to the containment are assumed to mix in the refueling water storage pit (RWSP) water at the time of release from the core. Confirm that recirculation water mass in DCD Table 15.6.5-4 includes water from the reactor coolant system (RCS) released through the break.

15.00.03-16

The LOCA dose analysis states that control room unfiltered inleakage is assumed to be 120 cubic feet per minute (cfm) due to unexpected ingress/egress. Identify the amount of unfiltered inleakage expected due to leakage through the control room envelope, and the associated TS testing requirements.

If the testing requirement for control room envelope inleakage is 120 cfm, then an additional amount should be assumed for ingress/egress through the doors, as discussed in SRP 6.4. The staff considers 10 cfm to be a reasonable estimate for ingress/egress through doors without a vestibule. Further information on control room habitability and control room envelope inleakage may be found in RG 1.196, "Control Room Habitability at Light-water Nuclear Power Reactors," and RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."

REQUEST FOR ADDITIONAL INFORMATION NO. 38 REVISION 1

15.00.03-17

DCD 15.6.5.5.1.3 states that the technical support center (TSC) dose consequences are represented by the control room dose consequences and there are some qualitative statements about the relationship of the TSC model to the control room model. Please provide the details of the TSC dose consequence model, sufficient for the NRC staff to determine independently if the control room dose consequence model is bounding for the TSC. Include information on the TSC intake and unfiltered inleakage receptor locations sufficient for the NRC staff to determine independently that the TSC atmospheric dispersion factors are bounded by the control room atmospheric dispersion factors for all DBAs.

15.00.03-18

Provide the control room direct dose calculations and justify the assumptions and inputs. Indicate the receptor location for the operator used in the control room direct dose model.

15.00.03-19

The non-LOCA fission product gap fractions given in RG 1.183 Table 3 were used as input to the fuel handling accident (FHA) analysis. Confirm that the US-APWR fuel meets the burnup and peak rod average linear heat generation rate conditions for applicability given in footnote 11 to RG 1.183 Table 3.

15.00.03-20

DCD 15A.1.1.3 states that the ORIGEN2.2 generation and depletion code was used to calculate the core fission product inventory. Oak Ridge National Laboratory (ORNL) does not support the ORIGEN2 code any longer, but instead recommends use of the ORIGEN-ARP or ORIGEN-S code included in the SCALE code package, which is kept up-to-date. SCALE 5.1 is the latest release and includes libraries for high burnup fuel, up to 72 gigawatt days per metric ton uranium (GWD/MTU). Please justify the use of an older unsupported version of the ORIGEN code.

15.00.03-21

DCD 15A.1.1.3 stated that the fuel burnup is 55 GWD/MTU in two cycles. Confirm that the cross-section libraries used in the calculation of the core fission product inventory are applicable to the maximum fuel burnup assumed.

15.00.03-22

DCD 15A.1.2.1 stated that the decontamination factor (DF) for the removal of elemental iodine in containment is time dependent, but did not provide the time when removal of elemental iodine is no longer effective. Please provide the time when the DF equals 200 in the associated analysis for the LOCA and the rod ejection accident (REA). For clarity, this information should be added to DCD Table 15.6.5-4 (LOCA) and Table 15.4.8-3 (REA).

15.00.03-23

Particulate iodine removal in containment was modeled by using the 10th percentile Powers natural deposition correlation in RADTRAD. Considering that the Powers natural deposition correlation was developed using operating pressurized water reactor (PWR) and boiling water reactor (BWR) information on containment geometry and power, explain why the Powers natural deposition correlation is applicable to the US-APWR containment.