From: Getachew Tesfaye
Sent: Friday, May 02, 2008 4:23 PM
To: 'usepr@areva.com'
Cc: John Rycyna; Edward Fuller; Lynn Mrowca; Theresa Clark; Hanh Phan; Joseph Colaccino
Subject: U.S. EPR Design Certification Application RAI No. 3
Attachments: RAI 3 SPLA 150.doc

Attached please find the subject requests for additional information (RAIs). A draft of the RAIs was provided to you on April 14, 2008, and your staff has indicated that the RAIs are understood and no follow-up telecon was necessary. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any 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.

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Request for Additional Information No. 3, Revision 0
05/02/2008
U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19
SPLB Branch

QUESTIONS

19-47

The NRC staff is requesting the applicant to provide the information listed below in order to complete its MELCOR input deck, to carry out the confirmatory assessment of AREVA's Level 2 PRA results, and severe accident evaluation reported in Chapter 19 of the U.S. EPR FSAR:

1) Containment elevation and overhead drawings, if available, would be helpful for understanding the containment layout, especially in the reactor pit region and the containment spreading compartment/area (Note that the information shown in the figures in Section 3B of EPR-FSAR are not considered sufficient). These should be provided in order to supplement, but not substitute for, the detailed requests for data listed below.
2) Please provide the following core and reactor pressure vessel data:
   - Elevation of the bottom of the cylindrical portion of the reactor vessel wall, with respect to the inside bottom of the reactor pressure vessel (RPV)
   - Total length of the control rod guide tubes
   - Outer diameter of the control rod guide assembly in the upper plenum region of the reactor vessel
   - Outer diameter of the normal columns in the upper plenum region of the reactor vessel
   - Outer diameter of the Level Measurement Probe (LMP) columns in the upper plenum region of the reactor vessel
   - Outer diameter of the instrument lance in the upper dome of the reactor vessel
   - Mass of the control poison inside the control rods
   - Loss coefficient or the pressure drop along the fuel rods in the core region, as well as for the lower support plate
   - Loss coefficient or the pressure drop at the inlet of the heavy reflectors
   - Loss coefficient or the pressure drop at the exit of the heavy reflectors
   - Loss coefficient or the pressure drop for the upper support plate
   - Loss coefficient or the pressure drop for the upper core plate
3) Please provide the following reactor coolant system and steam generators data:
   - The elevation of the centerline of the hot leg at the reactor pressure vessel nozzles, with respect to the inside bottom of the RPV lower head
   - Loss coefficient or pressure drop along the cross-over leg (i.e., the leg from the steam generator outlet plenum to the reactor coolant pump)
   - Elevation of the bottom of the pressurizer quench tank with respect to the bottom of the reactor vessel
   - Height of the pressurizer quench tank
   - Opening and closing pressures of the steam generator safety valves
   - Opening and closing pressures of the steam generator relief valve

- Mass of separators per steam generator
- Surface area of steam generator separators per steam generator
- Mass and surface area of dryers per steam generator
- Outer diameter of steam generator dryers
- Total free volume in the shield building
- Thermo-physical properties (i.e., density, thermal conductivity and specific heat as functions of temperature) of the Inconel 690 alloy used to fabricate the steam generator tubes

4) Please provide the following containment data:
- Density, ablation temperature, liquidus temperature of the sacrificial material in the reactor pit
- Density, ablation temperature, liquidus temperature of the sacrificial material covering the spreading area
- Thermo-physical properties of the protective layer in the containment spreading area, including its density, thermal conductivity, and specific heat as functions of temperature
- Dimensions of the reactor pit (with drawings, if available)
- Dimensions of the spreading area (with drawings that show the dimensions, if available), including the following:
  - Surface area of the sidewalls
  - Thickness of the sacrificial floor
  - Total heat transfer area of the bottom and sidewalls (separately) of the sacrificial region facing the coolant in the channel
  - Hydraulic equivalent diameter, length, and the channel gap for the coolant channel underneath and on the side of the spreading floor.
  - Diameter (or hydraulic area) and length of the pipe section connecting the IRWST to the spreading floor).

5) Table 6.2.1-5 of EPR FSAR tabulates the thicknesses and surface areas of the various heat sinks in the containment; however, the data for the height and bottom and top elevations are not provided. Please provide the height and the bottom and the top elevations of each heat sink listed in Table 6.2.1-5 with respect to the ground level. The following data sources have been utilized to prepare the preliminary MELCOR 1.8.6 input deck for the U.S. EPR confirmatory assessment:
- U.S. EPR Final Safety Analysis Report (FSAR) (AREVA NP Inc., 2007).
- The U.S. EPR MAAP4 input deck (AREVA NP Inc., 2007).
- Database – Parameter.doc (AREVA NP Inc., 2007).

In the course of preparing the MELCOR input deck, the missing design data requested above were identified. Depending on the nature and completeness of the data that will be made available to NRC by AREVA, pursuant to these requests and in the progression of the MELCOR deck development, additional requests for data may be made at a later date. Note that some data are requested here due to their unavailability in the EPR FSAR and other initially supplied sources, while other requests for clarification are present because of apparent inconsistencies that need to be resolved among the various data sources.

19-48

Section 19.1.4.2.2.4 discusses initiating event contributions to the large release frequency (LRF). According to Table 19.1-27, the most significant initiator from the LRF standpoint is a main steam line break inside containment, followed by steam generator tube rupture, loss of offsite power, and induced steam generator tube rupture. The large release frequencies of these four initiators sum to a total of 2.0E-09/year, more than 90% of the total LRF for the plant (2.2E-09/year).

1. Considering that a main steam line break inside containment has never occurred, please explain how the initiating event frequency of 1.0E-03/year was determined?
   a. Were any of the methods in the paper by Williams and Thorne, "The Estimation of Failure Rates for Low Probability Events" [1] used?
   b. If so, which equations were used?
   c. If not, what was the basis for selecting the value?
2. There have only been a few tube ruptures in the industry to date. What was the basis for selecting the initiating event frequencies for a steam generator tube rupture and induced steam generator tube rupture?
   a. Was Reference [1] used?If so, how?
   b. Was tube failure from wear at anti-vibration bars considered?
   c. Was tube failure from foreign object wear above the tube sheet considered?
   d. Was stress corrosion cracking considered? (Note that the EPR steam generator tubes will be made from Alloy 690 and will be thermally treated.)
   e. How were the tube failure rates from these mechanisms estimated?
3. Why are the LRF values so much lower than the initiating event frequencies (six orders of magnitude for the MSLB inside containment and SGTR initiators, and three orders of magnitude for the ISTGR initiator)?
4. Was the possibility of induced steam generator tube failure from MSLB (inside or outside containment) considered? If so, how were the frequencies determined?

Reference: 1:M. M. R. Williams and M. C. Thorne, "The Estimation of Failure Rates for Low Probability Events," Progress in Nuclear Energy Vol. 31, No. 4, pp 373-446, 1997.