July 30, 2007

Mr. Ronnie L. Gardner
AREVA NP Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: SECOND REQUEST FOR ADDITIONAL INFORMATION REGARDING ANP-10268P, "U.S. EPR SEVERE ACCIDENT EVALUATION TOPICAL REPORT" (TAC NO. MD3803)

Dear Mr. Gardner:

By letter dated October 31, 2006 (ML063100154), AREVA NP (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) ANP-10268P, Revision 0, "U.S. EPR Severe Accident Evaluation Topical Report" [ML063100166 (proprietary) and ML063100157 (non-proprietary)]. By letter dated June 13, 2007 (ML071550071), the staff issued the first set of request for additional information (RAI) to AREVA and AREVA provided its response by letter dated July 13, 2007 (ML071990057). As we informed you in our June 13, 2007 letter, we were unable to include RAIs from running the MAAP 4.0.7 code in our first set of RAIs in time to meet our commitment to provide you RAIs by June 15, 2007 (ML070030125). This letter transmits the staff's second set of RAIs that resulted from running MAAP 4.0.7 code. Our questions are provided in the enclosure.

A draft of the second set of RAIs was provided to you on July 16, 2007 (ML072000440), and discussed with your staff on July 20, 2007. Your staff has agreed that your response would be provided within 30 days of the date of this letter.

If you have any questions regarding this matter, I may be reached at 301-415-3361.

Sincerely,

/RA/

Getachew Tesfaye, Sr. Project Manager
EPR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Project No. 733

Enclosure: Request for Additional Information

cc: DC AREVA - EPR Mailing List

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SECOND REQUEST FOR ADDITIONAL INFORMATION (RAI)

ANP-10268P, "U.S. EPR SEVERE ACCIDENT EVALUATION

TOPICAL REPORT" (TAC NO. MD3573)

PROJECT NUMBER 733

RAI 50) General: The plant model for the U.S. EPR, as implemented in the MAAP 4.0.7 parameter file, is very elaborate, with 27 containment regions and 188 junctions. Most other plant models have been far less elaborate. Because of the complexity, each sequence takes about two orders of magnitude longer to run than for similar sequences in other plants. This will compromise the ability to run many sequence variations when doing the PRA during the design certification phase. Investigation of the plot files in the large-break LOCA sequence reveals that the containment gases are well-mixed, and the hydrogen concentrations are close to each other and behave very similarly vs time. The only differences are the spreading room and chimney, the reactor pit, and the cooling channel. Given this, please explain why the containment is modeled with so many regions and junctions, and what information is being sought by doing this?

It is also noted that the time step sometimes is reduced to the minimum allowable value and remains there for significant numbers of time steps. Given this, please explain what steps are being taken to assure that the minimum allowable time step chosen allows for sufficient accuracy and stability?

RAI 51) Loss-of-offsite power scenario: No RCP pump seal LOCAs were assumed for this scenario. Are such LOCAs expected for the U.S. EPR design? If so, how would the scenario be affected? How would the scenario behave if the severe accident depressurization valve was not opened? Would the hot legs be predicted to fail by creep rupture before the steam generator tubes? Would the vessel fail prior to creep rupture? If so, how close would the peak pressure from direct containment heating effects get to the failure pressure?

RAI 52) Loss-of-balance of plant scenario: The results from running the input file supplied by AREVA were reviewed. During this review, certain thermal hydraulic phenomena not generally observed in high RCS pressure, dry steam generator cases (high/dry/high cases) played a dominant role, leading to prediction of induced creep rupture of steam generator tubes before hot leg creep rupture was predicted (the severe accident depressurization valve was not activated for this case). Specifically, unidirectional flow of steam and hydrogen through the loops is calculated to occur, while no steam enters the bottom of the core. In addition, no upper plenum-to-steam generator, or steam generator inlet plenum to outlet plenum counter current flows are calculated to occur. Please explain whether or not these phenomena are to be expected for such scenarios given the U.S. EPR design.

ENCLOSURE

Please run a variation of this scenario, where the secondary sides of all four steam generators would be depressurized when the steam generator safety valves are first opened shortly after S/G dryout (assume the valves stick open after the first demand). This is called a high/dry/low situation, which is a risk-dominant scenario in existing LWRs. Please explain any key similarities and differences between the high/dry/high and high/dry/low scenarios. Also identify any potential numerical problems (such as prolonged periods of time when the minimum allowable time step is taken).

For each case (high/dry/high and high/dry/low), if the severe accident valves were to be actuated at the appropriate time (when the core outlet temperature was 650 °C), please explain how soon would depressurization occur relative to steam generator tube creep rupture.

cc:

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