AREVA

August 29, 2007
NRC:07:043

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Response to Second RAI on ANP-10268P "U.S. EPR Severe Accident Evaluation Topical Report" (TAC No. MD3803)

Ref.: 1. Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "Request for Review and Approval of ANP-10268P Revision 0, 'U.S. EPR Severe Accident Evaluation Topical Report'," NRC:06:049, October 31, 2006.

Ref.: 2. Letter, Getachew Tesfaye (NRC) to Ronnie L. Gardner (AREVA NP Inc.), "Second Request for Additional Information Regarding ANP-10268P, 'U.S. EPR Severe Accident Evaluation Topical Report' (TAC No. MD3803)," July 30, 2007.

Ref.: 3. Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "Response to a Request for Additional Information Regarding ANP-10268P, 'U.S. EPR Severe Accident Evaluation Topical Report' (TAC No. MD3803)," NRC:07:027, July 13, 2007.

Ref.: 4. Letter, Getachew Tesfaye (NRC) to Ronnie L. Gardner (AREVA NP Inc.), "Acceptance for Review of U.S. EPR Severe Accident Evaluation Topical Report (TAC No. MD3803)," January 17, 2007.

AREVA NP Inc. (AREVA NP) requested the NRC's review and approval of the topical report ANP-10268P Revision 0 in Reference 1. The NRC provided a second Request for Additional Information (RAI) regarding this topical report in Reference 2. The response to this RAI is enclosed with the letter as Attachment A, ANP-10268Q2, "Response to Second Request for Additional Information ANP-10268P, 'U.S. EPR Severe Accident Evaluation Topical Report'."

In addition to the RAI response, AREVA NP is providing proprietary replacement pages for ANP-10268P (Attachment B) and non-proprietary replacement pages for ANP-10268NP (Attachment C).

As detailed on the last page of Attachment A, the topical report replacement pages are provided to meet commitments included in AREVA NP's responses to the NRC's first RAI provided in Reference 3. Thus, the information reflected in the replacement pages has been reviewed by the NRC as part of its assessment of those responses. AREVA NP does not anticipate that these replacement pages will have a substantive impact on the review of the topical report or on the NRC's schedule for issuance of its safety evaluation report.

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AREVA NP plans to reference the topical report ANP-10268P in its Design Control Document (DCD) for the U S. EPR. Reference 4 states that the NRC plans to complete its review of the topical report and issue the draft safety evaluation by September 30, 2007. AREVA NP understands that this timely response to the RAI supports the scheduled deliverable of the draft safety evaluation.

AREVA NP considers some of the material contained in Attachment B to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the replacement pages are provided on the enclosed CDs.

If you have any questions related to this submittal, please contact Ms. Sandra M. Sloan, Regulatory Affairs Manager for New Plants Deployment. She may be reached by telephone at 434-832-2369 or by e-mail at sandra.sloan@areva.com.

Sincerely,

Jerold S Holm for

Ronnie L. Gardner, Manager
Site Operations and Corporate Regulatory Affairs
AREVA NP Inc

Enclosures

cc: L. J. Burkhart
G. Tesfaye
Project 733

AFFIDAVIT

STATE OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Russell D. Wells. I am Advisory Engineer, New Plants Deployment Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in Attachment B to AREVA NP letter NRC:07:043, "Response to Second RAI on ANP-10268P 'U.S. EPR Severe Accident Evaluation Topical Report' (TAC No. MD3803)," comprising replacement pages for the U.S. ANP-10268P, "U.S. EPR Severe Accident Evaluation Topical Report," and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 28th

day of August, 2007.

Sherry L. McFaden
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/2010
Registration #7079129

SHERRY L. McFADEN COMMONWEALTH VIRGINIA NOTARY PUBLIC

Response to Second Request for Additional Information – ANP-10268P "U.S. EPR Severe Accident Evaluation Topical Report" (TAC No. MD3803)

***RAI 50**: <u>General</u>*

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?

Response 50:

The U.S. EPR containment model, as represented in the MAAP 4.0.7 input file, was inspired from early EPR development analyses using the lumped parameter code COCOSYS. The U.S. EPR containment is designed with over 100 separate rooms or compartments, which served as a basis for the original nodalization. With regard to atmospheric mixing, it is correct that this model clearly shows that the atmosphere becomes well mixed following the opening of the severe accident depressurization valves (SADVs). As such, a model with less detail could suffice for many analyses.

One area in which a coarser model might not be sufficient is tracking fission products. The presence of structure and compartments provides surfaces onto which fission products may be deposited. In calculations being prepared for PRA Level 3 analysis, the modeled structure receives a broad range of fission product mass depending on its proximity to the RCS breach. Considering the SECY-93-087 expectation that designs "include ... cavity design features to contain ejected core debris in order to reduce the potential for containment failure as a result of direct containment heating," the demonstration of fission product attenuation through the different compartments supports the claim that the U.S. EPR satisfies this objective.

The demonstration of atmospheric mixing and fission product attenuation can be considered specialty analyses.

With regard to time step sensitivity studies, AREVA NP began by using the Fauske and Associates original recommendation (i.e., 10 s). Following an informal study of several calculations among the set of relevant scenarios, the current strategy of using a maximum time step of 1 s was adopted. This has worked satisfactorily in most situations; however, in a few instances related to calculations being prepared for the DCD, the maximum time step has been reduced below 1 s to obtain the necessary code stability. AREVA also remains cognizant of ongoing efforts related to code convergence, accuracy and stability via the MAAP4 Users Group community.

***RAI 51**: <u>Loss-of-offsite power scenario</u>*

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?

Response 51:

The objective of the calculations presented in the topical report was to demonstrate, through sample problems, the performance of the U.S. EPR's severe accident response features. The particular loss-of-offsite power (LOOP) scenario was chosen to illustrate how the U.S. EPR will respond to an event that is commonly expected to result in core damage at high pressure in current generation PWRs. Such conditions should increase the likelihood of creep rupture failure of the steam generator tubes, the pressurizer surge line, or the hot leg. Rather than allowing the event to proceed to that end state, the SADVs are opened and both the high pressure condition and any subsequent creep rupture is avoided. In this situation the reactor pressure vessel lower head will fail as a result of ablation driven by relocated core melt.

For analyses being prepared for the U.S. EPR DCD, several relevant scenarios (see RAI responses 41 and 44) are identified. A relevant scenario is defined as having a core damage frequency (CDF) greater than 10^{-8} occurrences per year. Among these is a LOOP with a reactor coolant pump (RCP) seal LOCA. As a consequence of the slow progression of the LOOP with seal LOCA into a severe accident, this event is expected to result in the highest reactor coolant system (RCS) pressure prior to core damage and RPV failure. While the uncertainty analysis examines process and phenomenological uncertainties associated with the relevant scenarios and considers timing uncertainty associated with SADV actuation, complete failure of the SADVs is considered to be unlikely for relevant scenarios, since there are two SADV trains (see Figure 2-8 of the topical report). Analysis of such events and their potential for induced creep rupture and direct containment heating is included in PRA development activities.

***RAI 52:** <u>Loss-of-balance of plant scenario</u>*

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.

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.

Response 52:

In the simulation identified in the RAI, the RCPs remain on until the void fraction at the pump inlet exceeds 1.0. Coupled with the cycling of the pressurizer relief valves, this leads to a near-total depletion of RCS coolant inventory (i.e., the loops are clear of any liquid) facilitating the onset of natural circulation. The assumption that the pumps remain on until the RCS is completely voided is an extreme variation of the treatment of RCP operability and is not credible. The RCPs in the U.S. EPR are automatically tripped when the differential pressure drops below 80% of nominal.

MAAP4 does include the models necessary to capture possible upper plenum to steam generator and steam generator inlet plenum to outlet plenum countercurrent flows. In fact, with the normal RCP trip, the most realistic scenario is for the loop seals to remain filled with water such that countercurrent natural circulation develops between reactor vessel and steam generators. MAAP4 is capable of capturing this countercurrent flow phenomenon. The design of the U.S. EPR RCS is similar enough to conventional PWRs that the likelihood of these phenomena appearing coincident with a severe accident is expected to be similar in these plant types.

One issue being investigated for Level 2 PRA is the probability of the various induced creep ruptures. The calculation variation involving the low steam generator pressure condition proposed in this RAI is addressed in this Level 2 PRA activity and will be included in a summary appearing in Chapter 19 of the U.S. EPR DCD. The reduced steam generator pressure is expected to increase the potential for induced steam generator tube rupture.

In calculations of relevant scenarios being prepared for the U.S. EPR DCD, AREVA NP expects to show that on-time actuation of the SADVs eliminates the possibility of these induced RCS ruptures. The induced rupture observed in the sample calculation discussed in this RAI is also a consequence of a significant delay applied to the actuation of the SADVs.

Replacement Pages

The following pages are replacement for pages given in the topical. These are offered based on the commitments offered in the original set of RAIs. Specifically, the following commitments were made with the release of those RAI responses:

RAI #	Commitment	Pages Changed
6	A complete suite of Figures and Tables from the latest sample problems will be provided with a revision of the topical report.	7-2 – 7-5, 7-9 – 7-80
31	Given the long lead time prior to SAHRS actuations, AREVA considers the reliability of the SAHRS sufficient with a single train. As such, the topical report will be updated to reflect this design change.	2-10, 2-33, 2-34, 2-38, 2-42, 5-81, and 5-94
32	Figure 6-1 will be revised to remove the link leading from the WALTER code to MAAP4.0.7.	pg. 6-2
39	In preparing this RAI, it was discovered that the CCI-2 and CCI-3 benchmarks were performed using an early developmental version of MAAP4.0.7. Relevant updates to that discussion in the topical will be provided in a topical report revision.	6-44 – 6-47
40	This benchmark will be updated with a topical revision.	6-44 – 6-47
47	The topical report will be revised to include the updated figures, including separate H_2 and CO mole fraction plots.	7-20, 7-44, and 7-68
48	Considering this decision, the inclusion of Appendix C is superfluous to the objectives of the topical report. The topical will be revision to eliminate this Appendix.	6-29, App. C deleted

Tables of Contents, Tables, and Figures updated as necessary to reflect these changes. In addition, three citations in the references contained errors (#9, 10, and 103). Corrections to these references are given in pages 9-1 and 9-9. Page 3-8 had an incorrect Section reference associated with "Sample Problem Analysis".