ArevaEPRDCPEm Resource

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Sent:	Thursday, September 10, 2009 9:10 AM
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Subject:	U.S. EPR Design Certification Application RAI No. 266 (3408,3443,3444,3445,3446), FSAR Ch. 6
Attachments:	RAI_266_SPCV_3408_3443_3444_3445_3446.doc

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on August 3, 2009, and discussed with your staff on August 13, 2009. RAI Questions 06.02.01-47, 06.02.01-48, 06.02.01.04-2, 06.02.01.04-5, and 06.02.01.04-7 were revised as a result of that discussion. 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.

Thanks,
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Hearing Identifier: AREVA_EPR_DC_RAIs
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9/10/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 06.02.01 - Containment Functional Design
SRP Section: 06.02.01.02 - Sub-compartment Analysis
SRP Section: 06.02.01.03 - Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)
SRP Section: 06.02.01.04 - Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures
SRP Section: 06.02.02 - Containment Heat Removal Systems

Application Section: FSAR Chapter 6

QUESTIONS for Containment and Ventilation Branch 1 (AP1000/EPR Projects) (SPCV)

06.02.01-47

This question relates to containment functional design in FSAR Section 6.2.1.1.1. The NRC staff understands that AREVA is developing multimode containment analytical capability using GOTHIC 7.2b. Provide input files for representative cases of LOCA and MSLB.

The above question is a follow-up to the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01-48

This question relates to negative containment pressure analysis in FSAR Section 6.2.1.1.1. FSAR Section 6.2.1.1.1 lists 5 potential events which cause negative pressure across the containment wall: 1. a sudden containment temperature reduction, 2. removal of IRWST inventory, 3. HVAC pulldown of containment pressure, 4. post-accident cooldown, and 5. post-severe accident cooldown. The greatest amount of pressure reduction, calculated for the sudden containment temperature reduction, was calculated to be 2.92 psi. The design negative pressure is 3.0 psi. Therefore the maximum temperature reduction was concluded to be within the design. For the sudden containment temperature reduction, drop from 122 ^{O}F to 59 ^{O}F was assumed. The minimum allowed containment temperature was stated to be 59 ^{O}F.

AREVA did not explain how the minimum allowed containment temperature was determined or explain the actions which would be taken to prevent the containment temperature from decreasing below 59 ^{O}F. AREVA did not explain the basis of assuming an initial relative humidity of 70% rather than 100% which would be more conservative for this calculation. Since the reduction in containment pressure might take place over a period of time, the operational staff might have time to mitigate the event. AREVA did not provide any mitigation procedures or evaluate their effectiveness. The staff requires additional information in order to close this issue.

AREVA did not describe the event that can result in the sudden temperature reduction from 122°F to 59°F. Describe the event. Can this event be prevented from occurring by interlocks and/or administrative controls? Provide justification for the assumed initial relative humidity of 70%. Furthermore, provide the mitigation procedures available to the operators to mitigate containment negative pressure events.

In response to RAI #104, 14.03-1a.1 AREVA stated that the screening approach using discipline checklists based on SRP 14.3 guidance did not identify safety-significant design features for a negative pressure inside containment. AREVA further stated that no design basis events described in the U.S. EPR FSAR result in a negative pressure inside containment. Therefore, ITAAC for a negative pressure inside containment are not included in U.S. EPR FSAR Tier 1. Provide justification for these statements in view of the statements made in FSAR Section 6.2.1.1.1 which discusses design basis events which might cause a negative containment pressure.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.02-2

This question relates to conservativeness of subcompartment differential pressure calculations in FSAR Section 6.2.1.2. In RAI #82 06.02.01.02-1a1, the staff requested additional justification that the use of the homogeneous equilibrium model (HEM) is conservative for the prediction of break flow for subcompartment analysis. AREVA supplied additional information in response to this RAI and also in response to RAI #1 06.02.01-8. The response to RAI #1 06.02.01-8 and RAI #82 06.02.01.02-1a1 are insufficient. AREVA has not submitted sufficient information to demonstrate that use of HEM for break flow is appropriate for subcompartment analysis. Use of other more conservative correlations is recommended in SRP 6.1.2-3. The values of break mass flux calculated by AREVA in FSAR table 6.2.1.2-2 using the HEM model are much lower than the break mass flux calculated by the staff using the SRP recommended models. In the response to RAI #82 06.02.01.02-a1, AREVA refers to the response to RAI #1 06.02.01-8 and to EPRI Report NP-2192 (Critical-Flow Data Review and Analysis). In the EPRI Report, predictions with HEM are compared with Marviken full scale test data. The measured flow rate is in many cases significantly higher than predicted by the HEM model. AREVA admits that HEM produces, in some of the subcompartments, significantly lower blow down flow than would be obtained using the models listed as acceptable in SRP 6.1.2-3, but states that the impact on the pressurization of critical rooms is negligible. AREVA did not present analyses to prove that the effect on the pressure loads is small if the recommended models were used. AREVA's answer does not provide justification that the use of HEM is conservative for the prediction of break flow for subcompartment analysis.

In order to resolve this issue, unless there is new convincing evidence or justification for use of the HEM, provide new subcompartment analyses using the models listed as acceptable in SRP 6.1.2-3 for the prediction of break flow for subcompartment analysis. The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.02-3

This question relates to conservativeness of subcompartment differential pressure calculations in FSAR Section 6.2.1.2. In RAI #82 6.01.02.02-1a3 the staff noted that not all subcompartments with high energy lines are considered in the for pressure evaluation. AREVA has limited the subcompartment analysis to those compartments that support the nuclear steam supply system components. Subcompartments that experience a pressure load but do not support NSSS components are omitted. SRP 6.2.1.2 defines subcompartments as any fully or partially enclosed volume within the primary containment that houses high-energy piping and which limit the flow of fluid to the main containment in the event of a pipe rupture within the volume. The NRC staff is required to review the noding scheme, initial thermodynamic condition, vent flow path and distribution of mass and energy released, design pressure, ITAAC and COL action items and certification requirements and restrictions for all subcompartments with high energy lines. AREVA has not provided sufficient information for the staff to perform this review.

In order to resolve this issue for all subcompartments with high energy lines, provide evaluations of the potential pressure loads including the accident pressures and comparisons of the calculated subcompartment pressure with the maximum pressure allowed. The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.02-4

This question relates to conservativeness of subcompartment differential pressure calculations in FSAR Section 6.2.1.2. In RAI #82 06.02.01.02-1b.2 the staff requested that for each subcompartment for which the pressure response to a high energy pipe break was calculated, that AREVA provide a comparison of the calculated subcompartment pressure with the maximum pressure allowed by the subcompartment design and justify that sufficient margin is available. In response, AREVA provided a table of calculated subcompartment differential pressures with the maximum differential pressure allowed for some of the subcompartments in FSAR Table 6.2.1-10 but not all of the subcompartments. Provide this information for all the subcompartments described in FSAR Table 6.2.1-10 as well as the additional subcompartments for which the staff has requested analyses under RAI # 82 6.02.01.02-1 a.3 to close this issue.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.03-2

This question relates to mass and energy release analysis for postulated loss-of-coolant accidents in accordance with SRP Section 6.2.1.3. In RAI #82 6.02.01.03-p the staff requested that AREVA provide justification for decreasing the core decay heat multiplier from 1.2 to 1.1 in the mass and energy release calculations for the long term post reflood phase. In the containment mass and energy release calculations for the EPR, AREVA uses the 1971 ANS standard with a 1.20 multiplier for the first 1000 seconds. After that a multiplier of 1.10 is used.

AREVA presented a comparison of the decay heat from the 1971 ANS standard using the 20 and 10 percent multipliers with the 1979 ANS standard using a 2σ multiplier. The staff has recently accepted containment mass and energy release calculations using the1979 ANS standard with a 2σ multiplier. The 1979 ANS standard has a more extensive data base than the 1971 standard and is considered to be the more accurate. AREVA concluded that the core decay heat used in the mass and energy calculations for the EPR containment analysis is conservative compared with the 1979 ANS standard with a 2σ multiplier and is therefore acceptable. AREVA extended the comparison to 10,000 seconds. The staff obtained a similar result for the first 10,000 seconds. At times greater than 20,000 the staff found the 1979 ANS standard with a 2σ multiplier to be the more conservative. In order to resolve this issue, demonstrate that the decay heat model used in the containment mass and energy calculations is conservative. In application of the 1979 ANS decay heat standard or later standards justify the input assumptions selected. These include assumptions for actinide decay, actinide production factor, multiplier to account for neutron capture activation, fissions per initial fissile atom and power history. The effect of these assumptions is discussed in NRC IN 96-39.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.03-3

This question relates to mass and energy release analysis for postulated loss-of-coolant accidents in accordance with SRP Section 6.2.1.3. RAI #82 6.02.01.03-1c is related to flow oscillations (chugging) in the reactor core calculated by RELAP5-BW. In the response to this RAI (Sup. 3), AREVA provided Figure 6.02.01.03-1-1 giving the inlet core flow from the original FSAR model and from a revised model. The revised models showed the core flow to be much smoother. Describe the changes in the RELAP5-BW model which produced the smoother core flow.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.04-2

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. In response to RAI #82 6.02.01.04-1a Sup.1 AREVA described, in general, the heat transfer models in the RELAP5/MOD2-B&W computer code. AREVA did not designate which heat transfer correlations were used in the main steam line break mass and energy release calculations. The most significant locations of heat transfer are the heat transfer from the primary system to the two phase mixture in the affected steam generator, the reversed heat transfer from the unaffected steam generators to the primary system, and heat transfer from the core. Demonstrate that these processes are conservative for the US-EPR limiting MSLB case. SRP 6.2.1.4 recommends that calculation of heat transfer to the water in the affected steam generator should be based on nucleate boiling heat transfer. Demonstrate that nucleate boiling was assumed for the limiting case or provide justification for other assumptions.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.04-3

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. In RAI #82 6.02.01.04-1c the NRC staff requested that AREVA identify the decay heat model that was used for the main steam line break analyses and provide justification that the model is conservative for containment analysis. SRP 6.2.1.4 does not address decay heat. However the SRP states that among the energy sources which should be considered is the energy transferred from the primary coolant to the water in the affected steam generator during blowdown. Additional decay heat will increase the reactor system temperature which will cause additional heat to flow to the affected steam generator. This energy source should therefore be made conservative in the safety analysis. Demonstrate that decay heat model used for evaluating the containment response to a main steam line break is conservative. The staff has accepted the 1979 ANS decay heat standard with a 2σ multiplier. If the 1979 ANS decay heat standard or a later standard is used for analysis, provide and justify the input assumptions selected. These include assumptions for actinide decay, actinide production factor, multiplier to account for neutron capture activation, fissions per initial fissile atom and power history. The effect of these assumptions is discussed in NRC IN 96-39.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.04-4

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. In RAI #82 6.01.02.04-1d the staff questioned the assumptions used to calculate reactor power following a main steam line break and the effect of these assumptions on containment analysis. In the response AREVA provided a curve of reactor power vs. time for the MSLB case calculated to produce the highest calculated containment temperature and pressure. This was for a postulated double ended break with the reactor at an initial power level of 50%. Offsite power was assumed to remain available so that the reactor coolant pumps would continue to operate. The break flow was assumed to be limited by the area of the flow restrictors in the steam generator nozzles to 1.4 ft2. The RELAP5/Mod2-B&W computer code was used to model the reactor system. No return to power was calculated even though the most reactive control rod was assumed to be stuck and to not enter the core following reactor trip.

AREVA also evaluated the consequences of a main steam line break in FSAR Section 15.1.5 to determine the potential for reactor core damage. In these evaluations the S-RELAP5 computer code was used. For the limiting break a return to power was calculated which reached a maximum of 23.14% over a period of approximately 200 seconds. This energy generation is sufficient, if considered in the containment analysis to have a considerable effect on the calculated containment temperature and pressure. In the response to RAI #34 15.01.05-2, AREVA provided the results from the sensitivity study of postulated steam line breaks for which the potential for core damage was evaluated. Initial power levels of 100%, 60%, 25% and 0% were investigated using S-RELAP5. The core was calculated to return to power generation following reactor trip regardless of the initial power level. The staff understands that part of the reason that return to power was calculated for the Chapter 15 analyses but not for the Chapter 6 analysis, was that a much higher control rod shutdown margin was assumed for the Chapter 6 analyses as compared with the Chapter 15 analyses. Provide analyses of the containment response to postulated main steam line breaks for which the core physics assumptions are consistent or conservative as compared to those which the staff is reviewing to support the main steam line break analyses in FSAR chapter 15.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.04-5

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. The staff understands that following a postulated main steam line break, if the liquid fraction in the steam separator region of the RELAP5-BW model exceeds a threshold value then entrained liquid is assumed to exit the break and to flow into the containment. If the liquid fraction in the steam separator region of the RELAP5-BW model is less than a second threshold value, any calculated entrained liquid is treated as steam. SRP 6.2.1.4 states that if liquid entrainment is assumed, experimental data should be provided to support the predictions. In the past the staff has accepted MSLB calculations with liquid entrainment based on model validation applying full scale steam generator test data. Provide experimental validation of the RELAP5-BW MSLB methodology or provide new analyses for which no liquid entrainment is assumed.

The above question is a follow-up question to the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.04-6

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. In response to RAI #82 6.02.01.04-1k concerning containment response to a main feedwater line break (FWLB) AREVA maintained that the consequences of (FWLBs) in terms of mass and energy release to the containment are less limiting than the consequences of MSLBs. The analysis establishing the conclusion, as well as the conclusion needs to be documented in the FSAR. GDC 50 and GCD 16 require that the containment accommodate releases for any loss of coolant accident. SRP 6.2.1.1.A states that main FWLBs should be included in the evaluations to show that these GDCs are met. Provide a section in the U.S. EPR FSAR evaluating postulated feedline breaks. Include a single failure analysis identifying the worst single failure.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.01.04-7

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. For each main steam and main feedwater line break analyzed, provide the signal producing reactor trip and that producing steam line and main feedwater line isolation. For the maximum break size which will not produce a reactor trip and the maximum break size which will not produce main steam and feedwater isolation, demonstrate that these break sizes will not produce more severe containment consequences than larger break sizes.

The above question is a follow-up question to the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.02-33

This question relates to containment heat removal systems in FSAR Section 6.2.2.
In RAI #82 6.02.02-1a, the staff requested that AREVA justify the statements, that: (1) in the LHSI heat exchangers conservative fouling factors were used; and (2) the GOTHIC heat exchanger model is a conservative representation. The response to the RAI provided the fouling factors, characterizing them as *typical.* It was mentioned that the fouling factors can be monitored with available systems information. The response also indicates , that benchmarking of the GOTHIC heat exchanger model against heat exchanger performance data yielded an overall heat transfer coefficient, UA value, of 2.47 E + 06 BTU/(hr-F°).

The containment analysis seems to use an overall heat transfer coefficient of 3.53 E + 06 BTU/(hr-F°). See Table 6.3-5 of the FSAR. The use of this coefficient together with typical fouling factors does not provide sufficient conservatism in the GOTHIC representation of the LHSI heat exchangers.

It is the staff's understanding that AREVA is performing revised containment analyses with a multi-node model of the GOTHIC code. For the revised calculations demonstrate that the fouling factors in the revised calculations conservatively represent those that will be present over the life of the unit.

Concerning the GOTHIC heat exchanger model, demonstrate the conservativeness of the model relative to the benchmarking information, and quantify the conservativeness. The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

06.02.02-34

This question relates to containment heat removal systems in FSAR Section 6.2.2. Sufficient information has been provided facilitating verification of water drainage into the IRWST pool. However, some of the water will be in transit accumulating on both vertical and horizontal surfaces, on equipment and on structures. If the containment has any rooms or compartments with solid floors, they can also add to the water accumulation. In order to calculate properly the minimum water level available in the IRWST during long term containment heat removal, the water holdup must be quantified. Provide a conservative estimate of water holdup in the containment following a LOCA. Specify the size and orientation of the surfaces considered in the estimate together with the assumed thickness of the water layer.

The above question is a follow-up question to the containment audit held in Lynchburg on July 14 and 15, 2009.