ArevaEPRDCPEm Resource

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Sent:	Friday, January 23, 2009 10:22 AM
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Cc:	Fred Forsaty; Jaclyn Dorn; John Budzynski; Shanlai Lu; Joseph Donoghue; Jason Carneal; Prosanta Chowdhury; Joseph Colaccino; Meena Khanna; ArevaEPRDCPEm Resource
Subject:	U.S. EPR Design Certification Application RAI No. 167 (1711, 1838, 1832), FSAR Ch. 15
Attachments:	RAI_167_SRSB_1711_1838_1832.doc

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on January 6, 2009, and discussed with your staff on January 16, 2009. Draft RAI Question 15.03.01-15.03.02-1 was deleted and Draft RAI Questions 15.06.05-39 and 15.06.05-42 were modified 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
Email Number: 145

Mail Envelope Properties (C56E360E9D804F4B95BC673F886381E71D23655ECC)

Subject: U.S. EPR Design Certification Application RAI No. 167 (1711, 1838, 1832), FSAR Ch. 15
Sent Date: 1/23/2009 10:21:42 AM
Received Date: 1/23/2009 10:21:43 AM
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Files	**Size**	**Date & Time**
MESSAGE	843	1/23/2009 10:21:43 AM
RAI_167_SRSB_1711_1838_1832.doc	41978	

Options
Priority: Standard
Return Notification: No
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Expiration Date:
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Request for Additional Information No. 167 (1711, 1838, 1832), Revision 0

1/23/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 15.03.01-15.03.02 - Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
SRP Section: 15.03.03-15.03.04 - Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
SRP Section: 15.06.05 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

Application Section: FSAR Ch 15

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

15.03.01-15.03.02-1

[Intentionally deleted].

15.03.03-15.03.04-9

Explain the analytic and numerical basis of the predicted extent of fuel damage (8%) in the reactor coolant pump shaft seizure postulated accident.

Regulatory basis; SRP 15.3.3-15.3.4 includes the following acceptance criteria:

> The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

15.06.05-27

Provide the administrative controls such as Technical Specifications for the accumulator fluid temperature, containment atmospheric temperature, and subcompartment atmospheric temperature that are used to ensure that the accumulator temperature used in the ECCS analysis bounds plant conditions. This question is asked to satisfy requirement of GDC 35, “Emergency Core Cooling.”

15.06.05-28

Explain the reactor coolant pump (RCP) trip logic and required operator actions should a small break loss of coolant accident occur. Show that the RCP trip will occur in a timely manner such that heat transfer is maximized while coolant mass loss is minimized. Explain how RCP seals are protected. Show that the RCPs will not be inadvertently tripped during a steam generator tube rupture event. Provide emergency procedure guidelines, intended criteria and proposed ITAAC related to operator actions for RCP trip. This question is asked to satisfy requirement of GDC 35, “Emergency Core Cooling.”

15.06.05-29

The report did not present results that describe possible accumulation of condensate in the steam generator (SG) exit chamber and the attached cold leg nozzle/piping nor did it consider the liquid conditions in the pump discharge pipe. Provide an assessment of the total amount of condensate generated in each loop.

Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-30

Explain the availability, quantity and distribution of emergency core cooling system (ECCS) flow for each individual loop consistent with the analysis assumptions made. Address the possibility for conditions that deprive a loop from any ECCS injection. If such a possibility exists, address associated effects on the restart of the natural circulation and mixing in the corresponding cold loop.

Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-31

Describe the SG secondary response in terms of pressure, temperature and liquid inventory conditions for each individual SG consistent with the analysis assumptions made, including EFW availability and depressurization considerations.
Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-32

Discuss the sufficiency of the cases analyzed in terms of number of break sizes considered within the break area range and spectrum.

Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-33

Discuss the adequacy of the S-RELAP model that is used to analyze the boron dilution and boron precipitation effects in the EPR SBLOCA analysis report referenced below.

Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-34

Explain the criticality consequences if a coherent slug of un-borated water of the quantity corresponding to the volume of a single crossover pipe were to be transported to the reactor vessel inlet by resumption of natural circulation.

Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-35

Show that the boron addition from all fluid sources such as the accumulators, and IRWST is sufficient to ensure that following a large break loss of coolant accident (LOCA) the reactor will be shutdown at cold conditions. This question is asked to satisfy requirement of GDC 27, "Combined reactivity control systems."

15.06.05-36

Explain why the qualitative behavior observed in the PKL tests is applicable to the US EPR.

Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-37

Explain the extent of mixing of the un-borated slug and the ECCS injection in the cold leg. Explain the relevancy of the CFD calculations which were performed for Olkiluoto-3 to the US EPR.

Reference: SBLOCA Analysis Report 32-9057270-000 in Support of Boron Dilution and Boron Precipitation Effects, dated December 14, 2006.

15.06.05-38

Provide hot rod center line and volume average fuel temperature predictions as a function of burnup (both core average and hot rod burnup) for EPR fuel as predicted by RODEX3A/S-RELAP5 for RLBLOCA.

15.06.05-39

Explain the conservatism in accounting for burnup effects on fuel characteristics and initial stored energy and associated uncertainties in applying RODEX2/S-RELAP5 for SBLOCA analyses. Assess and demonstrate that the SBLOCA methodology accounts conservatively for the initial core stored energy particularly if the hot rod, hot assembly, inner and outer core initial fuel temperatures are under-predicted (see 15.06.05-42).

15.06.05-40

Provide the S-RELAP5 input files for both the small and RLBLOCA limiting cases, as discussed in the FSAR. In addition, provide the hot rod and core average burnup initialization cases.

15.06.05-41

Explain how the initial core stored energy is accounted for in performing transient analyses for the U.S. EPR. Is RODEX2 or RODEX3 used for other transient analyses other than for LOCA? If a separate code, such as COPERNIC, is used to determine the initial core fuel conditions, specify the parameters that are transferred to the system code used for transient analyses. In particular, explain how burnup effects on fuel characteristics and initial fuel-stored energy are taken into consideration. Discuss the fuel burnup related uncertainties applied to the transient analyses and the justification for these uncertainties in demonstrating the acceptability of the safety margins obtained.

15.06.05-42

Although it is possible that the initial core stored energy may not significantly contribute to the PCT, it may have an impact on the SBLOCA core heat transfer characteristics when the break size approaches the upper break size limit of the applicable range of the SBLOCA method. Explain the impact of initial core stored energy on the break size upper limit if the hot rod and assembly are under-predicted by 400 and 350 degrees F respectively, and the inner core region and outer core region by 200 and 100 degrees F respectively. In addition, it appears that boron dilution analysis result during the SBLOCA transient is sensitive to scenario progression through the accident. Discuss the impact of any initial core stored energy under prediction on the result of the boron dilution analysis.