ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent:	Thursday, January 08, 2009 6:12 PM
To:	Getachew Tesfaye
Cc:	BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 142 (1623, 1630) ,FSAR Ch. 19
Attachments:	RAI 142 Response US EPR DC.pdf

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 142 Response US EPR DC.pdf" provides technically correct and complete responses to 5 of the 7 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 142 Questions 19-266 and 19-267.

The following table indicates the respective pages in the response document, "RAI 142 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 142 — 19-262	2	2
RAI 142 — 19-264	3	5
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RAI 142 — 19-266	7	7
RAI 142 — 19-267	8	10
RAI 142 — 19-268	11	11
RAI 142 — 19-269	12	12

A complete answer is not provided for 2 of the 7 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 142 — 19-262	March 6, 2009
RAI 142 — 19-269	March 6, 2009

Sincerely,

Ronda Pederson
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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Wednesday, November 26, 2008 12:34 PM
To: ZZ-DL-A-USEPR-DL
Cc: Hanh Phan; Theresa Clark; Edward Fuller; Lynn Mrowca; John Rycyna; Joseph Colaccino
Subject: U.S. EPR Design Certification Application RAI No. 142 (1623, 1630),FSAR Ch. 19

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on November 14, 2008, and discussed with your staff on November 24 and 25, 2008. Draft RAI Questions 19-263 and 19-270 were deleted, and Draft RAI Questions 19-262 and 19-269 were modified as a result of those discussions. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs, excluding the time period of **December 20, 2008 thru January 1, 2009, to account for the holiday season** as discussed with AREVA NP Inc. For any RAIs that cannot be answered **within 45 days**, it is expected that a date for receipt of this information will be provided to the staff within the 45-day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 97

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Response to

Request for Additional Information No. 142 (1623, 1630), Revision 0

11/26/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application FSAR Sections: 19.1.5 & 19.1.6.1

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 1 (AP1000/EPR Projects) (SPLA)

Question 19-262:

Follow-up to Question 19-50. Please revise the estimated EPR flooding frequencies in accordance with the statement on Page 19.1-115 of the EPR FSAR by including the potential flooding sources of valves, pumps, tanks/pools, and heat exchangers. Also, describe the process and references used to assign the equipment rupture frequencies. Otherwise, justify why the exclusion of these potential flooding sources in the analysis is conservative.

Response to Question 19-262:

A response to this question will be provided by March 6, 2009.

Question 19-264:

FSAR, Page 19.1-117, last paragraph indicates that flooding scenarios are quantified using the same event tree logic used in the Level 1 internal events evaluation, however it doesn't specifically indicate the event trees. Thus, for each analyzed scenario, please identify the conditional event tree used to quantify the internal flooding CDF and provide the basis for selection.

Response to Question 19-264:

Flooding scenarios are quantified using the event trees from the Level 1 internal events probabilistic risk assessment (PRA) that are presented in U.S. EPR FSAR, Tier 2, Appendix 19.A. The choice of an event tree to represent a specific flooding scenario is based on the similarity in the expected plant response. For example, flooding scenarios that affect a specific Safeguards Building (SB) or essential service water (ESW) pumphouse are assumed to result in a loss of at least one train of component cooling water (CCW), and are, therefore, evaluated and quantified through the loss of CCW (LOCCW) event tree. Flooding-induced system unavailabilities associated with each initiating event and scenario are modeled explicitly in the fault trees. The RiskSpectrum® PRA software allows such an approach, because it makes it possible to disable any fault tree gate by a specific initiator, as discussed in the response to RAI 53, Question 19-209.

Table 19-264-1 shows the event tree (ET) used to quantify each flooding scenario, as well as the mitigating systems assumed to be disabled by these flooding scenarios.

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Table 19-264-1—Event Trees Used to Quantify Flooding Scenarios

FLOOD IE	Description	Event Tree	Additional Impact on Mitigating Systems (Systems set to failure)
IE FLD-SAB14 FB	Flood in Safeguards Building 1 or 4 (Pump Room), including Fuel Building (FB), excluding EFW-caused floods	Loss of Component Cooling Water (LOCCW)	SB4 systems: CCW4 (and CCW Common Header 2), EFW4, MHSI4, LHSI4, and SAHR. FB systems: EBS and CVCS.
IE FLD-SAB23	Flood in Safeguards Building 2 or 3 (Pump Room), excluding EFW-caused floods	LOCCW	SB2 systems: CCW2, EFW2, MHSI2, and LHSI2
IE FLD-EFW	EFW-caused flood in Safeguards Building 1 or 4, including Fuel Building	LOCCW	SB4 systems: CCW4 (and CCW Common Header 2), EFW4, MHSI4, LHSI4, and SAHR. FB systems: EBS and CVCS. If isolation fails, all four EFW trains are assumed to be lost.
IE FLD-TB	Flood in the Turbine Building	Loss of Balance of Plant (LBOP)	None
IE FLD-ESW	Flood in the Essential Service Water Building	LOCCW	UHS4, disabling ESW4/CCW4
IE FLD-ANN ALL	Flood in the Annulus, contained within the Annulus and reaching the electrical penetrations.	Single function event: core damage with 0.5 probability	None
IE FLD-ANN SAB23	Flood in the Annulus with propagation to Safeguards Building 2 and 3 (Pump Room)	LOCCW	SB1 systems (conservatively selected due to the running CCW pump): CCW1 (and CCW Common Header 1), EFW1, MHSI1, and LHSI1 SB3 systems: CCW3 (and conservatively CCW Common Header 2), EFW3, MHSI3, and LHSI3
IE FLD-ANN SAB2	Flood in the Annulus with propagation to Safeguards Building 2 (Pump Room)	LOCCW	SB1 systems (conservatively selected due to the running CCW pump): CCW1 (and CCW Common Header 1), EFW1, MHSI1, and LHSI1

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 19-265:

In accordance with guidance provided in SRP Section 19.1.3.4, please explicitly describe the uses of EPR PRA-based seismic margins assessment and insights/assumptions in the design process to reduce the weaknesses/vulnerabilities, develop design requirements, and improve the EPR design safety profile.

Response to Question 19-265:

The PRA-based seismic margin assessment (SMA) is within the scope of the U.S. EPR probabilistic risk assessment (PRA), as stated in U.S. EPR FSAR, Tier 2, Section 19.2.1. The uses and application of the PRA during the design phase are stated in the U.S. EPR FSAR Tier 2, Section 19.1.1.1 and Section 19.1.3.4, and describe the uses of the PRA in accordance with the Standard Review Plan (SRP) Section 19.1.3.4. These uses apply to the PRA-based seismic margin assessment (SMA). Uses of the SMA are based on key assumptions and insights derived from the qualitative PRA-based SMA results. The analysis is based on the seismic equipment list (SEL) as shown in U.S. EPR FSAR, Tier 2, Table 19.1-106 and Table 19.1-107 (see RAI 8, Supplement 1, FSAR markup and the RAI 97 response to Question 19-215). The SMA did not identify any seismic-related weaknesses or vulnerabilities in the U.S. EPR design.

The seismic risk insights are also considered in the Reliability Assurance Program (RAP), by including the SEL as part of the PRA input to the RAP.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 19-266:

The applicant refers to “transient-induced LOCA [loss-of-coolant accident]” on page 19.1-159 of the Final Safety Analysis Report (FSAR) and in the event tree for loss of the residual heat removal system (RHRS) in Appendix 19B of the FSAR. However, transient-induced LOCA is not defined. Revise the FSAR to describe the phenomenon to which this term refers (e.g., a stuck-open relief valve caused by the energy addition following a loss of RHRS).

Response to Question 19-266:

In the low power and shutdown probabilistic risk assessment (PRA), the transient-induced LOCA top event (TR LOCASD) is used in the loss of residual heat removal (RHR) event trees during plant operating states C (CA and CB). Three ways to induce a transient LOCA, or to fail the TR LOCASD top event, have been considered:

- Pressurizer safety valve fails to re-close after the reactor coolant system heats up.
- Reactor coolant pump (RCP) seal LOCA, for plant operating states with the RCPs operating (CAd and CAu).
- Reactor pressure vessel or pressurizer vents fail to close. This condition was considered and screened because the time to uncover the core is more than a day, allowing time for operators to isolate the path.

The FSAR will be revised to include the definition of the transient-induced LOCA top event.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 19.1.6.1.5 will be revised as described in the response and indicated on the enclosed markup.

Question 19-267:

(Follow-up to Question 19-131) The shutdown probabilistic risk assessment (PRA) assumes that plant operating state (POS) CAd and CBd are entered 8 hours and 44 hours following shutdown, respectively. The response to Question 19-131 acknowledges that, for a specific outage, entrances to different POS could occur earlier and therefore with higher decay heat loads. Before the shutdown schedule was extended by seven days per year to account for the assumed capacity factor (as discussed in the response to Question 19-19), what were the times of entry into POS CAd and CBd? Discuss whether the decay heat at these earlier POS entry times would be high enough to reduce the time available for any shutdown operator actions below the threshold allowed in the shutdown probabilistic risk assessment. Revise the FSAR as needed to ensure that this assumption about POS entry times is clearly identified and can be confirmed in the future.

Response to Question 19-267:

In the shutdown probabilistic risk assessment (SD PRA), the average shutdown duration during the plant life was modeled, and not a specific, most conservative, shutdown schedule. Based on the overall decay heat and POS timing-related assumptions, it is expected that the model conservatism is preserved, and that earlier entrances to the starting POSs are unlikely to change the SD PRA insights and conclusions. This conservatism is defined in the response to RAI 14, Question 19-131.

If, as requested in the question, the seven days that are added to the shutdown duration to match 94% availability are removed, POS durations would be shortened to approximately:

- One day total for POSs CAD, CBD, and DD.
- Five days total for POSs E and F.
- One day total for POSs DU, CBU, and CAU.

Based on the decay heat impact time line presented in RAI 14, Question 19-131, Table 19-131-1, this shortening would only impact entrance to POS CBD and DD. Still, in this sensitivity evaluation, it was also assumed that CAD (Mode 4) could be entered in six hours, instead of eight hours. Entrance to CAD (248 to 212°F) in less than six hours after power operations is unlikely, based on industry experience and cooldown rate limitations.

This sensitivity evaluation is summarized in Table 19-267-1, which presents the “old” and “new” time entries into affected POSs, and the corresponding decay heat increase factor. The increase in decay heat is unlikely to change the time interval for the corresponding operator actions. A few operator actions that are close to the bottom end of the “old” time interval are evaluated in the “new” time interval. Even with these very conservative assumptions, possible impact on the SD core damage frequency is less than 3 percent, as presented in the table.

U.S. EPR FSAR, Tier 2, Table 19.1-109 item 51, will be revised to reflect that the shutdown POS durations and schedule in the low power shutdown PRA are based on the following assumptions:

- 18-month refueling cycle.
- 94 percent plant availability.

- Normal refueling outage of 14 days.
- Forced outage rate of five days per year.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 19.1-109 item 51 will be revised as described in the response and indicated on the enclosed markup.

Table 19-267-1—Shutdown Sensitivity Studies on Times to Entry for POS CAd, CBd, and Dd

POS	SD PRA Assumed Entry Time (hrs)	Sensitivity Case Entry Time (hrs)	Decay Heat Increase Factor	Possibly Affected Operator Actions	Corresponding SD PRA Time Interval (hrs)	Corresponding Human Error Probability	Sensitivity Case Time Interval (hrs)	Sensitivity Case Human Error Probability	Operator (OP) Action Importance Measure (Fussell-Vesely (FV) if not indicated differently)	Corresponding Increase in Core Damage Frequency
CAd	8	6	1.06	OPE-FB-CAD(C2): OP Fails to initiate F&B after a LOCA in CAd, given MHSI failure	(0.42 - 0.58)	2.0E-02	0.42	1.1E-01	0.0014	0.6%
CBd	44	32	1.11	OPF-RHR-CBD & OPE-RHRLO-CBD: OP Fails to Start RHR in CBd (after a loss of RHR or a LOCA)	(0.92 - 1.42)	1.1E-03	(0.58 - 0.92)	2.0E-03	0.0018	0.1%
Dd	92	70	1.10	OPF-RHR-DD: OP Fails to Start RHR in Dd	(0.42 - 0.58)	2.0E-02	< 0.42	1	1.01 (Risk Achievement Worth)	1.0%
Dd	92	70	1.10	OPF-LHSILO-DD & OPF-LHSIULD-DD: OP Fails to Start LHSI Pump in Dd, for LOCA or ULD	(0.58 - 0-92)	2.0E-03	(0.42 - 0.58)	2.0E-02	0.0005	0.5%
									TOTAL:	**2.2%**

Question 19-268:

(Follow-up to Question 19-166) Item 13 in Table 19.1-108, provided in response to Question 19-166, states that NUMARC 91-06 should be considered when developing plant-specific operations procedures. Given that NUMARC 91-06 provides only high-level risk management guidance and that more specific procedural actions are detailed in Generic Letter (GL) 88-17, discuss whether GL 88-17 should be added to item 13 of Table 19.1-108.

Response to Question 19-268:

U.S. EPR FSAR, Tier 2, Section 5.4.7.2.1 and Table 15.0-60 provide the disposition of GL 88-17 for the U.S. EPR. For the U.S. EPR, GL 88-17 related items that are significant to the probabilistic risk assessment are in the insights and assumptions in U.S. EPR FSAR, Tier 2, Table 19.1-108 and Table 19.1-109. For example, see Table 19.1-108 item numbers 16 and 20 and Table 19.1-109 items related to low power shutdown (LPSD).

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 19-269:

One of the acceptance criteria stated in Standard Review Plan (SRP) Section 19.0 is "that the applicant has used the PRA results and insights, including those from uncertainty analyses, importance analyses, and sensitivity studies, in an integrated fashion to identify and establish specifications and performance objectives (e.g., ITAAC [inspections, tests, analyses, and acceptance criteria], technical specifications [TS], RAP [reliability assurance program], RTNSS [regulatory treatment of non-safety systems], and COL [combined license] action items) for the design, construction, testing, inspection, and operation of the plant." Links between the PRA and the design process (FSAR Section 19.1.3.4), RAP (FSAR Section 19.1.7.4), development of PRA-based insights (FSAR Table 19.1-108), and development of COL items (FSAR Table 1.8-2) are specifically outlined. However, it is not clear how PRA results and insights were used to identify or establish any TS requirements. Therefore, the staff needs additional information to ensure the SRP acceptance criterion is met. Specifically:

a. Describe the process used to identify and establish TS requirements based on PRA results and insights (e.g., risk achievement worth (RAW) values), and revise the FSAR as appropriate to include this discussion. Include a discussion of how the process considers TS criterion 4, as presented in Title 10 of the Code of Federal Regulations (10 CFR) 50.36(c)(2)(ii)(D): "[a SSC] which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

b. In light of the response to part (a) above, justify the lack of TS requirements for the equipment for which sensitivity studies were performed in response to Question 19-176. This lack of TS requirements means that the shutdown risk profile of the U.S. EPR is largely dependent on voluntary actions (e.g., NUMARC 91-06 application) that could be withdrawn without NRC approval. If these design features were not available, the resulting risk profile is highly uncertain. Core damage frequency (CDF) and large release frequency (LRF) could increase by two or more orders of magnitude, and the impact on the significant initiating events, sequences, and SSCs is unclear. The justification should address the impact of equipment unavailability on the results and insights of the PRA.

Response to Question 19-269:

A response to this question will be provided by March 6, 2009.

U.S. EPR Final Safety Analysis Report Markups

Table 19.1-109—U.S. EPR PRA General Assumptions
Sheet 9 of 15

No.	Category[1]	PRA General Assumptions[2]
47	I&C	The system SAS contains controls for post-accident safety systems. The SAS model in the PRA is simplified because design details were unavailable.
48	I&C	The normal plant control systems (PAS and RCSL) have features to reduce the frequency and consequence of plant transients that may challenge the safety systems. This is accomplished both by the way that the control functions are distributed within the I&C system divisions and by the limitation I&C functions. In as much as the PRA uses historic operating experience for the initiating event frequencies, the impact of these features is not evaluated in the PRA.
49	I&C	Instrument calibration errors are not evaluated for the design certification PRA. Instrumentation calibration errors will be analyzed in more detail after maintenance procedures and insights from maintenance practices are available.
50	LPSD	RCS level and volume are treated conservatively during the RCS level transitions in outages. For example, whenever the reactor cavity is not flooded and RCS level is not in the pressurizer, mid-loop operation is assumed. The following further summarizes this conservatism: • Whenever the pressurizer is being drained, this time is applied to mid-loop. • Whenever the reactor cavity is being drained after refueling, this time is applied to mid-loop. • When level is near the flange during RPV head removal and installation, this time is applied to mid-loop. • When level is increased from mid-loop to fill the cavity or pressurizer, this time is applied to midloop.
51 19-267	LPSD	The shutdown POS durations and schedule in the LPSD PRA are based on the following assumptions: • 18-month refueling cycle. • 94% plant availability. • Normal refueling outage of 14 days. • Forced outage rate of 5 days/year. The LPSD PRA model assumes that the RCS status as well as decay heat are constant during the time within the POS. The analysis considers an early entry time after shutdown for the start of a POS and then decay heat is not reduced during the POS. This is conservative during a shutdown to cold conditions (e.g., unplanned maintenance) when decay heat levels would be much lower over time than that assumed in POS CA or POS CB.

Event Tree "SD RHR C" models plant responses to loss of RHR while in POS CA or CB. The loss of RHR initiating event model includes operator actions to recover RHR (e.g., start a standby pump train). Event tree top event "TR LOCASD" models the probability of a transient-induced LOCA. LOCA response requires feed-and-bleed cooling success because it is conservatively assumed that the LOCA may not be large enough to provide sufficient bleed. Three ways to fail the TR LOCASD top event have been considered:

19-266

- PSV fails to reclose after RCS heats up.
- RCP seal LOCA
- RPV or PZR vent fails to close. This condition was considered and screened because the time to uncover the core is more than a day, allowing significant time for operators to isolate the path.

Event Tree "SD RHR D" models plant responses to loss of RHR while in POS D. Since the RPV head is off, the model is much simpler than for State C. The initiating event model includes recovery of RHR standby trains.

Event Tree "SD ULD CB" models plant response to an uncontrolled level drop in POS CB. Since RCS inventory is assumed to be diverted via CVCS storage outside containment, the long-term failure to isolate is assumed to result in a loss of the IRWST outside containment and containment bypass.

Event Tree "SD ULD D" models plant response to an uncontrolled level drop in POS D. Since the RPV head is off, the model is much simpler than for State C. The RCS inventory is assumed to be diverted via CVCS storage outside containment, and the long-term failure to isolate is assumed to result in a loss of the IRWST outside containment and containment bypass.

Event Tree "SD LOCA C" models plant response to a LOCA inside containment while in POS CA or CB. The LOCA initiating event model includes pipe break, as well as RHR flow diversion.

Event Tree "SD LOCA D E" models plant response to a LOCA inside containment while in POS D or E. The LOCA initiating event model includes pipe break, as well as flow diversion from the RHR system.

There are several Event Trees "SD RHR ISLOCA" that model RHR pipe break LOCA events outside containment. The probability of failure to isolate this type of event is already included in the initiating event frequency. Thus, these initiating events result in a loss of the IRWST outside containment, core damage, and containment bypass. The shutdown event trees are shown in Appendix 19B.