ArevaEPRDCPEm Resource

From:	WELLS Russell D (AREVA NP INC) [Russell.Wells@areva.com]
Sent:	Monday, July 06, 2009 6:08 PM
To:	Tesfaye, Getachew
Cc:	Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 227, FSAR Ch 19
Attachments:	RAI 227 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 227 Response US EPR DC.pdf" provides technically correct and complete responses to 12 of the 20 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 227 Questions 19-299 and 19-300.

The following table indicates the respective pages in the response document, "RAI 227 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

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RAI 227 — 19-284	2	3
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A complete answer is not provided for 8 of the 20 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 227 — 19-284	September 18, 2009
RAI 227 — 19-285	July 20, 2009
RAI 227 — 19-287	September 18, 2009
RAI 227 — 19-292	September 18, 2009

RAI 227 — 19-293	September 18, 2009
RAI 227 — 19-294	September 18, 2009
RAI 227 — 19-295	September 18, 2009
RAI 227 — 19-298	August 28, 2009

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Cc: Theresa Clark; Hanh Phan; Edward Fuller; Lynn Mrowca; Prosanta Chowdhury; Joseph Colaccino; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 227 (2564, 2598),FSAR Ch. 19

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on May 15, 2009, and discussed with your staff on May 29, 2009. Draft RAI Questions 19-296, 19-300, and 19-302 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: 622

Mail Envelope Properties (1F1CC1BBDC66B842A46CAC03D6B1CD4101AD1191)

Subject: Response to U.S. EPR Design Certification Application RAI No. 227, FSAR Ch 19
Sent Date: 7/6/2009 6:07:58 PM
Received Date: 7/6/2009 6:08:06 PM
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Files	Size	Date & Time
MESSAGE	3315	7/6/2009 6:08:06 PM
RAI 227 Response US EPR DC.pdf	1871057	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Response to

Request for Additional Information No. 227

6/05/2009

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

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

Question 19-284:

(Follow-up to Question 19-68) The staff needs additional information on the software common-cause failures (CCF) modeled in the U.S. EPR probabilistic risk assessment (PRA) to conclude that the low postulated failure rates do not result in an over-optimistic estimation of risk. Specifically:

a. The assumed software CCF probabilities can be found in Final Safety Analysis Report (FSAR) Table 19.1-13, but not in the text of Section 19.1.4.1.1.3 where digital instrumentation and control (I&C) modeling is discussed. Revise the FSAR to state the software CCF probabilities assumed.

b. Page 19.1-34 of the FSAR states that the TELEPERM XS (TXS) “operating history…is used to generate a bounding value” for the operating system CCF probability. Interim Staff Guidance (ISG) DI&C-ISG-03 cautions that “extrapolation of statistical data of the same system used in a different operating environment or profile is not necessarily meaningful.” Discuss how this bounding value was developed, including a justification of the applicability of operating history to a new environment in which a different set of input parameters could reveal a fault not exposed in the previous operating history. Revise the FSAR to include a summary of this information, and confirm that all important assumptions implicit in the operating system CCF value are included in FSAR Table 19.1-109.

c. Page 19.1-34 states that application software CCF probabilities are “based on comparison of the software development…process and the TXS platform design characteristics with applicable international standards.” Revise the FSAR to describe the specific process and design characteristics that contribute to the low application software CCF probability. Discuss how the software development process supports the assumption that the application software in diversity groups A and B can be considered independent in the PRA, given the use of “qualified software functional blocks from a controlled library.” Confirm that all important assumptions implicit in the application software CCF value are included in FSAR Table 19.1-109.

d. Confirm whether the error factor of five used for digital I&C equipment (stated in FSAR Section 19.1.4.1.2.7) was also applied to the operating system and application software CCFs. Justify the uncertainty parameters applied to software CCFs given the limited state of knowledge about these failures. Revise the FSAR to include a summary of this information.

e. The sensitivity studies performed in response to Question 19-68 provide useful information on the effect of modeling uncertainty on the at-power core damage frequency (CDF). However, the effect of these sensitivity cases on both CDF and large release frequency (LRF) resulting from all modes of operation is unclear. Provide CDF and LRF results from the fourth sensitivity case for both at-power and shutdown modes.

f. In the sensitivity studies performed in response to Question 19-68, the operating system and application software CCF probabilities are increased by one order of magnitude (to 1E-6 and 1E-4, respectively). So that the staff can understand the importance of low software CCF probabilities to the overall risk profile, provide the CDF and LRF results from a sensitivity study in which these probabilities are increased to demonstrably conservative values (e.g., 1E-4 and 1E-3).

Response to Question 19-284:

A response to this question will be provided by September 18, 2009.

Question 19-285:

(Follow-up to Question 19-125) The significant sequences (i.e., those with a sequence frequency greater than 1 percent of internal events or shutdown core damage frequency (CDF) or those that have an aggregate contribution of 95 percent of CDF when ranked by frequency) provided in response to Question 19-125 yield different insights than the cutset groups listed in FSAR Table 19.1-7. For example:

- The response to Question 19-125 states that sequence 14 in the loss-of-offsite-power (LOOP) event tree has a sequence frequency of 8.58E-8 per year (/yr), about 30 percent of the internal events point estimate for CDF. In comparison, the first cutset group in FSAR Table 19.1-7 includes cutsets from this sequence that are included in the top 100 cutsets, and represents only about 19 percent of the internal events CDF.
- The top five cutset groups based on the percent contributions listed in FSAR Table 19.1-7 are 1 (LOOP-14), 9 (SLOCA-17), 17 (ATWS-12), 8 (SLOCA-34), and 18 (GT-15). In contrast, the top five sequences based on the response to Question 19-125 are LOOP-14, GT-15, SLOCA-17, SLOCA-34, and LOOP-45.

The staff uses the significant cutsets and sequences to communicate important scenarios both to other reviewers and to the public in the Safety Evaluation Report (SER). In addition, the staff uses individual cutsets to understand the modeling of systems and operator actions. Therefore, the staff needs to see both cutsets and sequences and understand the reasons for any discrepancies in the rankings.

a. Revise the FSAR to include a ranking of significant sequences (those provided in response to Question 19-125), with a description of each. The representative cutsets and sequence descriptions currently provided in FSAR Table 19.1-7 are one way of describing the sequences.

b. Provide (in the RAI response only) the top 10 cutsets for each significant sequence, or the cutsets contributing 95 percent to the sequence frequency, whichever is less.

c. Provide (in the RAI response only) the top 200 core damage cutsets for internal events, internal fire, internal flooding, shutdown, and the total at-power and shutdown model.

Response to Question 19-285:

A response to this question will be provided by July 20, 2009.

Question 19-286:

(Follow-up to Question 19-143) The response to Question 19-143 states that “JNG10AA192 injection safety valve is screened [for spurious operation and flow diversion during shutdown] because this is a 1-inch line downstream of the low head safety injection (LHSI) minflow orifice.” However, cutset group 13 in Table 19.1-92 includes event “JNG10AA192SPO,” premature opening of safety valve JNG10AA192. Clarify why this failure event appears in the shutdown PRA. Discuss how the screening approach described in the response to Question 19-143 was communicated to the PRA developers, and discuss whether any similar discrepancies exist.

Response to Question 19-286:

The response to Question 19-143 stated that JNGx0AA192 injection safety valves (four valves in different trains) are screened from consideration for flow diversions in the FSAR shutdown (SD) probabilistic risk assessment (PRA) model. The response was based on design information not yet incorporated into the PRA model, which reduces the size of the lines containing JNGx0AA192 valves to one-inch. As stated in the question, this failure event appears in the SD PRA because in the PRA model used to generate the FSAR inputs, including Table 19.1-92, it was assumed that these were two-inch lines, and therefore they were included as flow diversion paths.

A sensitivity case is performed to assess the impact of this design change on the SD PRA results and insights. In this sensitivity case, “JNGx0AA192” valves are deleted from the SD LOCA initiating events fault tree to reflect screening of this valve from consideration for flow diversions. The results of this sensitivity case are shown in Table 19-286-1. Total core damage frequency (CDF) decreased by approximately one percent, due to elimination of flow diversions during shutdown that were a result of spurious opening of JNG10AA192.

No similar discrepancies have been identified.

Table 19-286-1

Results of the Sensitivity Case with Flow Diversions through JNG10AA192 Removed

Hazard Group	**Base Case CDF (1/yr)**	**Sensitivity Case CDF (1/yr)**	**Delta CDF**
Total (all modes and initiating events)	**5.84E-07**	**5.78E-07**	**1%**

FSAR Impact:

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

Question 19-287:

(Follow-up to Question 19-259) The response to Question 19-259 discusses the undeveloped basic events used for the process automation system (PAS) and safety automation system (SAS). It appears that, other than the sensors that may provide input to both the protection system (PS) and these systems, no dependency is assessed between the PS and PAS or SAS. The descriptions of all three systems, as well as that of severe accident (SA) I&C, in FSAR Section 7.1 state that they include "subracks, I/O modules, function processors, and communication modules, and optical link modules." The staff needs additional information to understand how these systems are modeled in the PRA.

a. List the systems or functions (e.g., EDG actuation, partial cooldown) that the PRA assumes are actuated by each I&C system. For systems or functions actuated by the PS, state which diversity group is assumed to support the actuation.

b. Describe all scenarios that include independent failures of both PS and another digital I&C system (e.g., PAS, SAS, SA I&C).

Discuss whether CCFs of I&C components, which are modeled in detail for the PS, could be expected to affect PAS or SAS as well. If so, how is this dependence modeled in the PRA?

Response to Question 19-287:

A response to this question will be provided by September 18, 2009.

Question 19-288:

Page 19.1-45 of the FSAR states that “[t]he model is solved by using a 1E-20 truncation limit, and a 1E-6 relative truncation limit. The CDF quantification, for Level 1 at power, all events, resulted in over 73,000 cutsets.” The staff needs additional information on these truncation limits to determine that the reported CDF adequately represents the U.S. EPR design. Specifically, address the following topics and revise the FSAR as appropriate.

a. Describe the evaluation performed to determine that these truncation limits were appropriate (e.g., results had converged).

b. Define the “relative truncation limit.”

c. Justify the 1E-6 relative limit given the high level of redundancy in the U.S. EPR design, which would yield many low-frequency cutsets with different combinations of independent failures.

Response to Question 19-288:

Response to Question 19-288a:

A simplified evaluation was performed with different cutoff points, to evaluate cutoff (truncation limit) values where the quantification results would have stabilized, while still allowing a reasonable time for quantification (the current probabilistic risk assessment (PRA) Level 2 quantification requires over 5 days). An example of this evaluation for a loss of offsite power (LOOP) event tree is shown in Table 19-288-1, and, for relative cutoff values, illustrated in Figure 19-288-1. In this example, core damage frequency (CDF) cutsets are not post-processed or minimized.

As shown in the table and the figure, running time is sensitive to both an absolute and a relative cutoff, while the results are more sensitive to relative cutoff values. Figure 19-288-1 illustrates that the results would stabilize around a 1E-6 cutoff value, and that further lowering of this value would not significantly change the results, though it would increase the time for calculation.

As shown in Table 19-288-1, a selected relative cutoff of 1E-6 in the U.S. EPR PRA results in a corresponding absolute cutoff value close to 1E-13, and lowering the absolute cutoff value below 1E-13 does not change the results.

Also, note that in the U.S. EPR PRA peer-review, the related PRA ASME Standard SR QU-B3: “ESTABLISH truncation limits by an iterative process of demonstrating that the overall model results converge and that no significant accident sequences are inadvertently eliminated. For example, convergence can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF or LERF, and the final change is less than 5%” was evaluated as “SR Met”.

Response to Question 19-288b:

The RiskSpectrum® manual (Reference 1) provides a definition of the relative cut-off value:

"The relative cut-off value RCUTM is used in the following way: When the cut sets are generated, the program calculates the unavailability/frequency for each minimal cut set, which is found. As the MCSs are found and stored, the sum of their unavailabilities/frequencies is calculated. This sum, QSUM, is an approximate upper bound for the top event unavailability/frequency. The relative cut-off value RCUTM is used to calculate a new (absolute) cut-off value which is relative to QSUM: Cut-off value = RCUTM x QSUM."

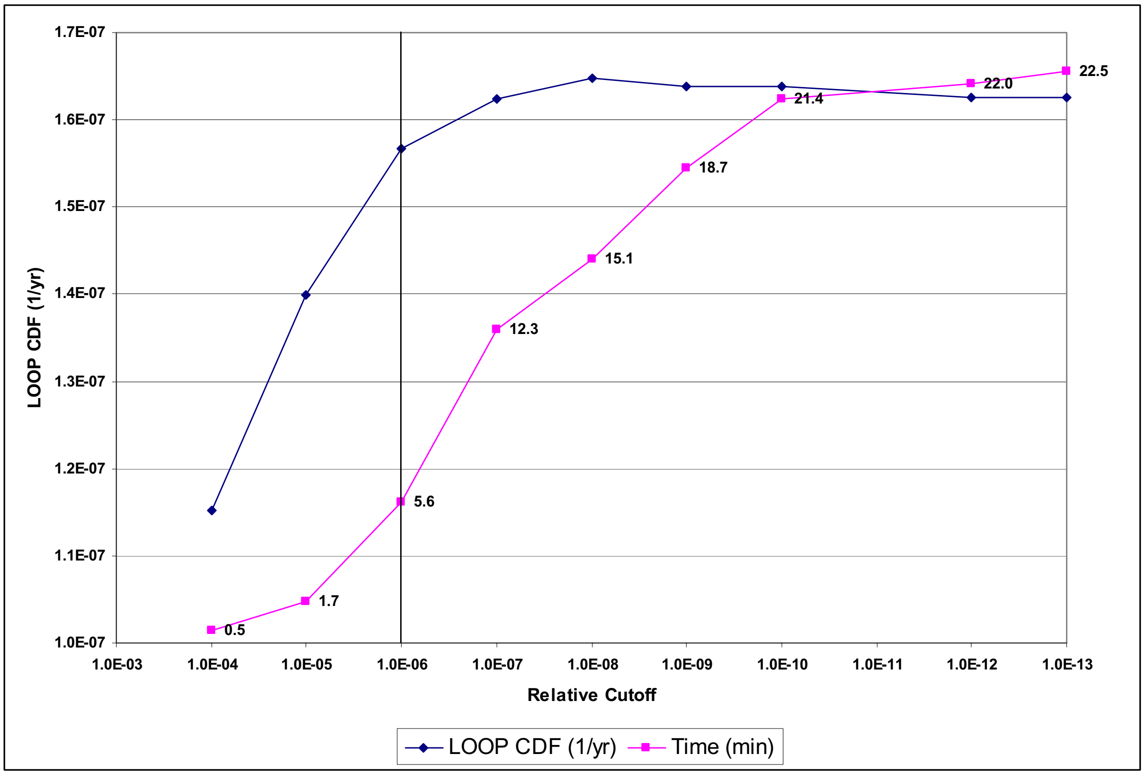
Response to Question 19-288c:

As discussed in the Response to Question 19-288a, Figure 19-288-1 illustrates that the results would stabilize around a 1E-6 cutoff value, and that further lowering of this value would not significantly change the results, though it would significantly increase processing time, making more complex quantifications unpractical. Based on this, the relative cutoff of 1E-6 was selected as an acceptable compromise between results stabilization and running time. Also as stated in the Response to Question 19-288a, this relative cutoff is close to an absolute cutoff of 1E-13, and should not have a significant impact on the results. While lower cutoff values could somewhat alter the results (still within the uncertainty limits), they are not likely to change the conclusions or insights from the probabilistic risk assessment (PRA) model.

Table 19-288-1: Sensitivity Study on the PRA Model Quantification Cutoff Values

Initiating Event	Cutoff		Consequence Analysis	
	Absolute	Relative	Core Damage Frequency	Time (min)
Sensitivity to Absolute Cutoff				
LOOP	1.00E-10	1.00E-06	9.3433E-08	0.2
LOOP	1.00E-11	1.00E-06	1.2542E-07	0.3
LOOP	1.00E-12	1.00E-06	1.4631E-07	1.1
LOOP	1.00E-13	1.00E-06	1.5667E-07	7.2
LOOP	1.00E-15	1.00E-06	1.5676E-07	10.3
LOOP	1.00E-20	1.00E-06	1.5676E-07	10.8
LOOP	1.00E-25	1.00E-06	1.5676E-07	11.9
Sensitivity to Relative Cutoff				
LOOP	0.00E+00	1.00E-04	1.1518E-07	0.5
LOOP	0.00E+00	1.00E-05	1.3996E-07	1.7
LOOP	0.00E+00	1.00E-06	1.5676E-07	5.6
LOOP	0.00E+00	1.00E-07	1.6241E-07	12.3
LOOP	0.00E+00	1.00E-08	1.6477E-07	15.1
LOOP	0.00E+00	1.00E-09	1.6377E-07	18.7
LOOP	0.00E+00	1.00E-10	1.6375E-07	21.4
LOOP	0.00E+00	1.00E-12	1.6250E-07	22.0
LOOP	0.00E+00	1.00E-13	1.6250E-07	22.5

Figure 19-288-1: Impact of Relative Cutoff Values on the CDF Results and Quantification Time



References:

1. RiskSpectrum® Analysis Tools: Theory Manual, Relcon Scandpower AB, Version 3.0.0

FSAR Impact:

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

Question 19-289:

(Follow-up to Questions 19-4 and 19-258) FSAR Section 19.1.4.1.4 and the response to Question 19-4 provide the data sources and failure data used in the U.S. EPR PRA. However, the rationale for selecting the specific data source used for each failure in the PRA is not provided. For example, the response to Question 19-258 states that failures from European sources are “sometimes based on standby failure rates” without discussing whether standby or demand failure rates are appropriate for the failures being modeled. Provide a detailed discussion of the U.S. EPR failure database development process, including (a) a rationale for the data sources used for the failures identified in Table 19-04-1, and (b) a discussion of how failure models (e.g., standby versus demand) were selected for the basic events.Response to Question 19-289:

Response to Question 19-289a:

The rationale behind the U.S. EPR data sources selection is consistency with the European EPR probabilistic risk assessment (PRA) model. The reasons for this decision are summarized below:

1. The data compared well with the data from the available data sources in the U.S. (NUREG 1715), including the advanced light-water reactors (ALWR) Electric Power Research Institute (EPRI) data, as illustrated in Table 19-04-1.

2. There are no data available for the ALWR, other than ALWR data used in the EPRI comparison.

3. Vendor and equipment specifics depend upon procurement of the components.

Response to Question 19-289b:

As discussed in the response to RAI 138, Question 19-258, data selected in the U.S. EPR PRA are consistent with the European data that in some cases includes assumptions on test intervals or stand-by versus demand failure rates. However, as stated in the response to RAI 138, Question 19-258, in the U.S. EPR PRA application, no assumptions are made about fraction of stand-by versus demand failure rates. Demand failures are always used as probabilities, and as such they are compared to other data in Table 19-04-1.

No speculations were made about what fraction of the demand failures is due to stand-by hourly failure rate, and which one is due to the demand or shock, itself. No assumptions are made about test intervals since these require test procedures. However, it was verified that the risk-important components that would be subject to infrequent testing, such as the common safety injection check valves in the containment (JNG13AA005, JNG23AA005, JNG33AA005, JNG43AA005), have a demand failure rate that adequately accounts for infrequent testing. For example, the (JNGX3AA005) safety injection check valves have a failure rate to open on demand of 9.63E-4 per demand, compared to CCW check valves that have a failure rate of 5E-5/demand.

FSAR Impact:

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

Question 19-290:

(Follow-up to Questions 19-126 and 19-260) The response to Question 19-260 identifies a beta factor of 2.81E-2 for failure of the two station blackout diesel generators (SBODG) to run, using CCF data for emergency diesel generators (EDG). The basic event data provided in response to Question 19-126 indicates that the SBODG failure probability over the mission time is 5.44E-2, about twice the EDG failure probability. As a result, the independent failure of both SBODGs appears to have a higher probability than a CCF, resulting in the cutset presented in group 4 of Table 19.1-7. Typically, CCFs are more likely than independent failures. Justify this treatment or revise the PRA as appropriate. Discuss whether similar discrepancies exist for other components modeled in the PRA.

Response to Question 19-290:

As presented in the Response to RAI 138, Question 19-260, the common cause failure (CCF) parameters applied to the SBODG are based upon NUREG/CR-5497, "2003 CCF Parameter Estimations". The U.S. EPR PRA basic event failure probabilities and the U.S. EPR multiple Greek letter (MGL) factors are not directly correlated in accordance with standard CCF methodologies (as explained in NUREG/CR-5485 and NUREG/CR-5801, for example). A low basic event failure probability does not (in itself) suggest a low susceptibility to common-mode failure, nor does a large basic event failure probability (in itself) suggest a high susceptibility to common mode failure.

For situations where the random independent failure probability is greater than Beta (and the common cause group size is two), it is appropriate for the random independent failure contribution to exceed the common cause failure contribution. This occurs occasionally with high failure probability equipment (e.g. diesel generators fail-to-run).

FSAR Impact:

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

Question 19-291:

(Follow-up to Question 19-249) The sensitivity study performed in response to Question 19-249 demonstrates a significant difference in both CDF (nearly 40 percent reduction) and the overall risk profile when component cooling water system (CCWS) trains 2 and 3 are operating rather than trains 1 and 4. This difference demonstrates the importance of the assumption that the CCWS common header switchover fails following a ventilation failure, leading to failure of CCWS-cooled ventilation in a second safeguard building (SB) and thereby failure of that building's equipment.

FSAR page 19.1-37 indicates that the heat-up of electrical rooms in the affected SB is "relatively slow" and that equipment is lost after about two hours. The operators' ability to start a maintenance train of ventilation or to recover room cooling is credited in the PRA. However, no automatic or operator actions related to the CCWS system (e.g., swapping to a standby train before the running train fails) are credited.

FSAR page 19.1-61 states that "[s]ensitivity studies did not identify any events where a design change would lead to a significant reduction in the CDF." The staff views this ventilation dependence, which page 19.1-37 indicates is a conservative assumption in a PRA model designed to be realistic, as a candidate for a significant reduction in CDF. How were the effects of the current ventilation and CCWS model on the PRA results communicated to plant designers? Discuss any design changes, such as automatic actions or procedural steps (operator actions), that were considered to eliminate or reduce the likelihood of this scenario in the PRA. Justify why these changes were not implemented.

Response to Question 19-291:

Generally speaking, PRA models should be designed to be realistic versus conservative. The U.S. EPR PRA is also designed to be realistic; however, a number of conservative assumptions were made when complete design details are to be developed later in the design process. This practice has been minimized to the extent possible.

AREVA concurs with the statement in the question that the possible HVAC dependency is important to address, as it is clearly identified in the U.S. EPR FSAR. However, no specific design change has been identified which would reduce this dependency, without introducing the other risk concerns such as: eliminating interlocks between CCW header closing and opening valves, or introducing an automatic CCW switchover on room temperature alarms. This issue can be addressed through the plant procedures that, on a loss of HVAC for a specific division, instruct operators to make sure that a running CCW pump is not supplied from this division (perform a CCW switchover if necessary).

FSAR Impact:

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

Question 19-292:

(Follow-up to Question 19-203) The response to Question 19-203 states that the general transient (GT) initiating event includes spurious actuation of the PS. How are I&C failures (e.g., software CCF) that could both cause an initiating event and affect mitigation considered in the PRA?

Response to Question 19-292:

A response to this question will be provided by September 18, 2009.

Question 19-293:

(Follow-up to Question 19-260) The table of CCF parameters provided in response to Question 19-260 shows separate CCF groups for processors in seven different actuation logic units (ALU) and acquisition and processing units (APU). Even if these processors perform different functions, they could be of the same type with common manufacturing, maintenance, or installation errors. Therefore, justify the use of separate CCF groups. Revise the FSAR to document any important assumptions related to CCF of these processors.

Response to Question 19-293:

A response to this question will be provided by September 18, 2009.

Question 19-294:

(Follow-up to Question 19-260) The table of CCF parameters provided in response to Question 19-260 shows separate CCF groups for many similar sensors (e.g., different groups for each steam generator's level and pressure sensors, different groups for pressurizer and hot leg level sensors). Justify each group of sensors with reference to the function, operating environment, and potential for manufacturing, maintenance, and installation errors. Revise the FSAR to document any important assumptions related to CCF of these sensors.

Response to Question 19-294:

A response to this question will be provided by September 18, 2009.

Question 19-295:

(Follow-up to Question 19-260) The table of CCF parameters provided in response to Question 19-260 does not include CCFs of analog or digital input and output modules, for which failure data was provided in response to Question 19-4. Clarify whether CCFs of these modules were postulated; if not, justify their exclusion.

Response to Question 19-295:

A response to this question will be provided by September 18, 2009.

Question 19-296:

(Follow-up to Question 19-274) The response to Question 19-274 describes the use of the demineralized water distribution system (DWDS) or fire water distribution system (FWDS) to refill an emergency feedwater system (EFWS) tank. An operator action with a human error probability (HEP) of 8E-4 is postulated. The staff needs additional information to understand this operator action.

a. Describe how the HEP of 8E-4 was estimated, and document any important assumptions (especially those related to procedural steps and training) in the FSAR.

b. If failures of the DWDS and FWDS that prevent refill of the tank(s) are not modeled in the PRA, provide a quantitative justification for their exclusion. Describe how components from these systems were considered for input to the reliability assurance program (RAP) and other programs that receive input from the PRA.

Response to Question 19-296:

Response to Question 19-296a

The action to refill an EFWS tank (OPF-EFW RF-6H) is required following a pressure boundary failure in one EFWS train (a pipe break or a tank leak). For this scenario, the operators would be required to initiate a refill of the tank in one of the intact trains. There are several means to perform this action, including the DWDS, which is the preferred means, and the FWDS.

The HEP associated with this action is calculated using the SPAR-H method. All performance shaping factors (PSF), including procedures and training are assumed to be 1 for “nominal/not enough information”, even if it is likely that the operators will be well trained in this action. PSFs for timing, stress and complexity are further evaluated, as discussed below.

The total time available is six hours, which is a representative timing for the exhaustion of one tank. A four hour delay for the cue is estimated, which represents the time to reach a low-level alarm in one of the intact tanks. The action to refill a tank can be performed from the control room (DWDS) or locally (FWDS). Thus, a range between 5 and 15 minutes is assumed for the action time. The resulting timing PSF is “expansive time” (x 0.01) for cognition, and “extra time” (x 0.1) for the action.

Stress is high (x 2) and complexity is moderate (x 2) for both cognition and the action. This reflects a degraded situation, with the simultaneous occurrence of a flooding event and a loss of feedwater, with multiple cues and alarms.

Thus the resulting HEP is

- Cognitive HEP: 0.01 x 0.01 x 2 x 2 = 4E-4
- Action HEP: 0.001 x 0.1 x 2 x 2 = 4E-4
- <u>Total HEP: 4E-4 + 4E-4 = 8E-4</u>

No assumptions are made in the evaluation of this action regarding procedures, training or any PSF other than stress and complexity. General assumptions made regarding human reliability analysis and documented in FSAR Tier 2, Table 19.1-109 (more specifically items 22 to 24) are applicable to this evaluation.

Response to Question 19-296b

Failures of the DWDS or the FWDS are not explicitly considered in the failure of the tank refill. These systems are not modeled in the probabilistic risk assessment (PRA). (As discussed in the response to RAI 7, Question 19-76, the PRA model only includes the emergency part of the DWDS). The impact of this exclusion is low, because:

- The tank refill function has a high level of diversity and redundancy; therefore it is expected that its failure is dominated by the human error probability. Each of the two systems could supply enough water to adequately refill one tank and support operation of the EFWS for 24 hours. One of the two DWDS 100 percent pumps could be used to achieve mission success, or one of the three FWDS 100 percent pumps. The three FWDS pumps are diverse (two diesel-driven, one electrical). Therefore, the available pumps would not be vulnerable to common-cause failure or failure due to a common support system (e.g., electrical system or heating, ventilation, air conditioning).
- The ability to refill the EFWS tanks is found to have a low importance. It is only credited in the case of an EFW tank failure or a pipe break. The HEP “OPF-EFW RF-6H” has a Fussell-Vesely of 0.0001and a risk achievement worth of 1.15.

Therefore it is concluded that DWDS/FWDS failures do not affect the results or insights of the PRA with respect to the EFW tank refill model.

The DWDS is considered in the RAP program because the emergency portion of the system to refill the feedwater storage tank to support the startup and shutdown system operation is explicitly modeled in the PRA. Components whose importance measures are below the threshold for inclusion or which are not explicitly credited in the PRA, could still be considered in the RAP program and evaluated based on the expert panel judgment.

FSAR Impact:

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

Question 19-297:

(Follow-up to Question 19-274) For the sensitivity study performed in response to Question 19-274, it is unclear how the logic representing “[i]f one (or more) EFW[S] trains are unavailable” and “[i]f one EFW[S] train is unavailable due to a pressure boundary failure” was developed. Provide revised EFWS fault trees that show the changes made, as well as any other relevant PRA changes, so that the staff can understand the effect on the model. If the eventual PRA update will be implemented differently from the sensitivity study, discuss how it will be implemented.

Response to Question 19-297:

The fault trees that model the failure of the EFWS due to insufficient inventory for the sensitivity study performed in response to RAI 197, Question 19-274 are shown in Figures 19-297-1 and 19-297-2.

The fault tree shown in Figure 19-297-1 is the input to the function event “EFW INV”. That function event replaces the existing header “EFW PBF”, which currently represents EFW pressure boundary failure only. It represents the failure of the EFWS to provide sufficient inventory to support secondary cooling for 24 hours.

The top gate is an OR gate between the two cases (with or without pressure boundary failure). The branch that represents the case where a pressure boundary fails is shown in Figure 19-297-1. The case where train unavailability requires tank interconnection is shown in Figure 19-297-1 as a transfer to the “EFW LT” fault tree, which is illustrated in Figure 19-297-2.

These two fault trees include the two operator actions described in the response to Question 19-274:

- OPF-EFW RF-6H (Figure 19-297-1) is used in the case of a pressure boundary failure to represent the failure to refill one of the intact storage pools.

- OPF-EFW-6H (Figure 19-297-2) is used in the case where one EFWS train is not available for a reason other than pressure boundary failure. It represents the failure to manually open the suction cross-connect valves.

The next PRA update is expected to use a model similar to the one described in this response. However, consideration will be given to the possibility of refilling the EFWS storage pools as a backup alternative to cross-tying the storage pools in the “EFW LT” fault tree. An EFW suction source could be maintained if either EFW storage pool makeup is provided, or if the inventory from the disabled EFW train(s) is provided to one of the operating trains. A human error dependency would be applied between the two operator actions.

FSAR Impact:

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

Figure 19-297-1: Fault tree "EFW INV": Failure of EFW Inventory Including Pressure Boundary Failure

--EFW INV **RP192742**

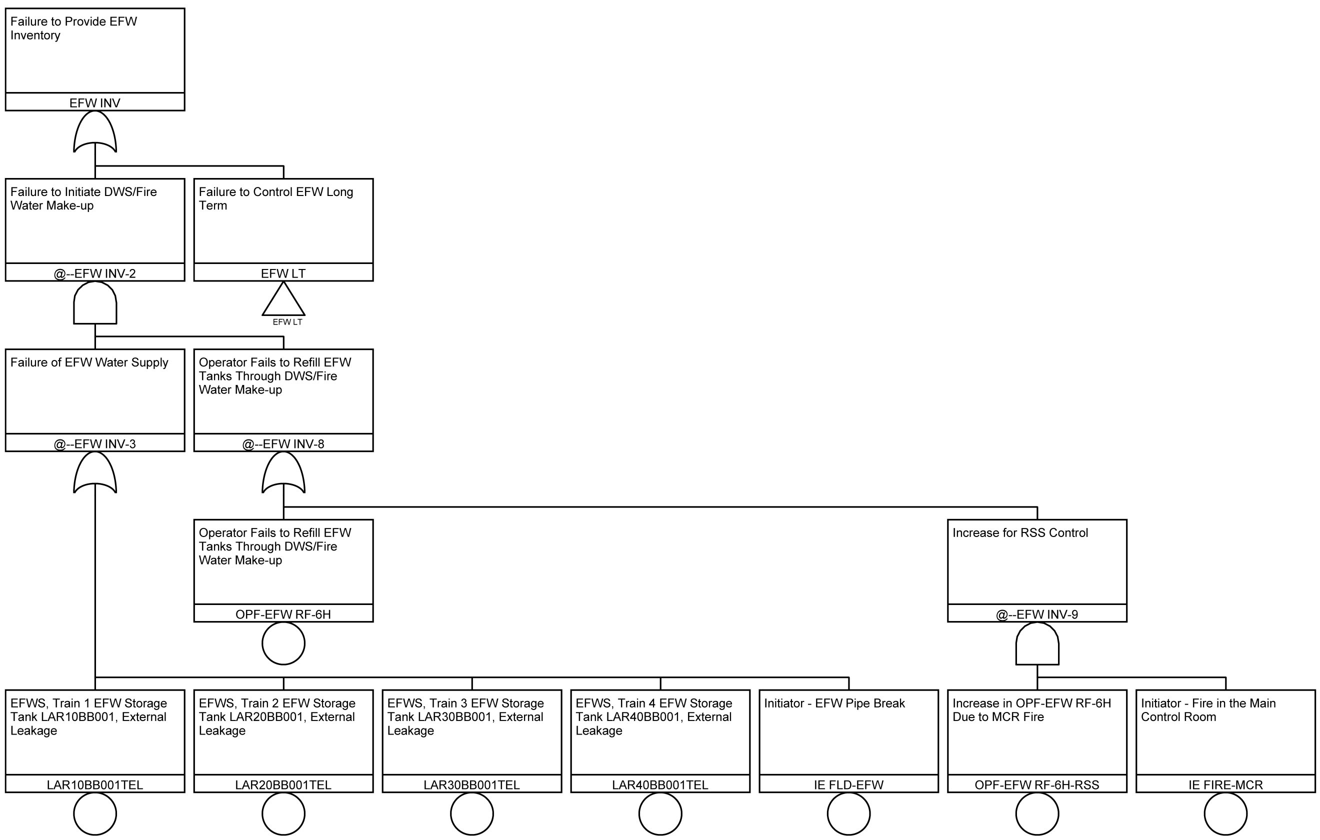
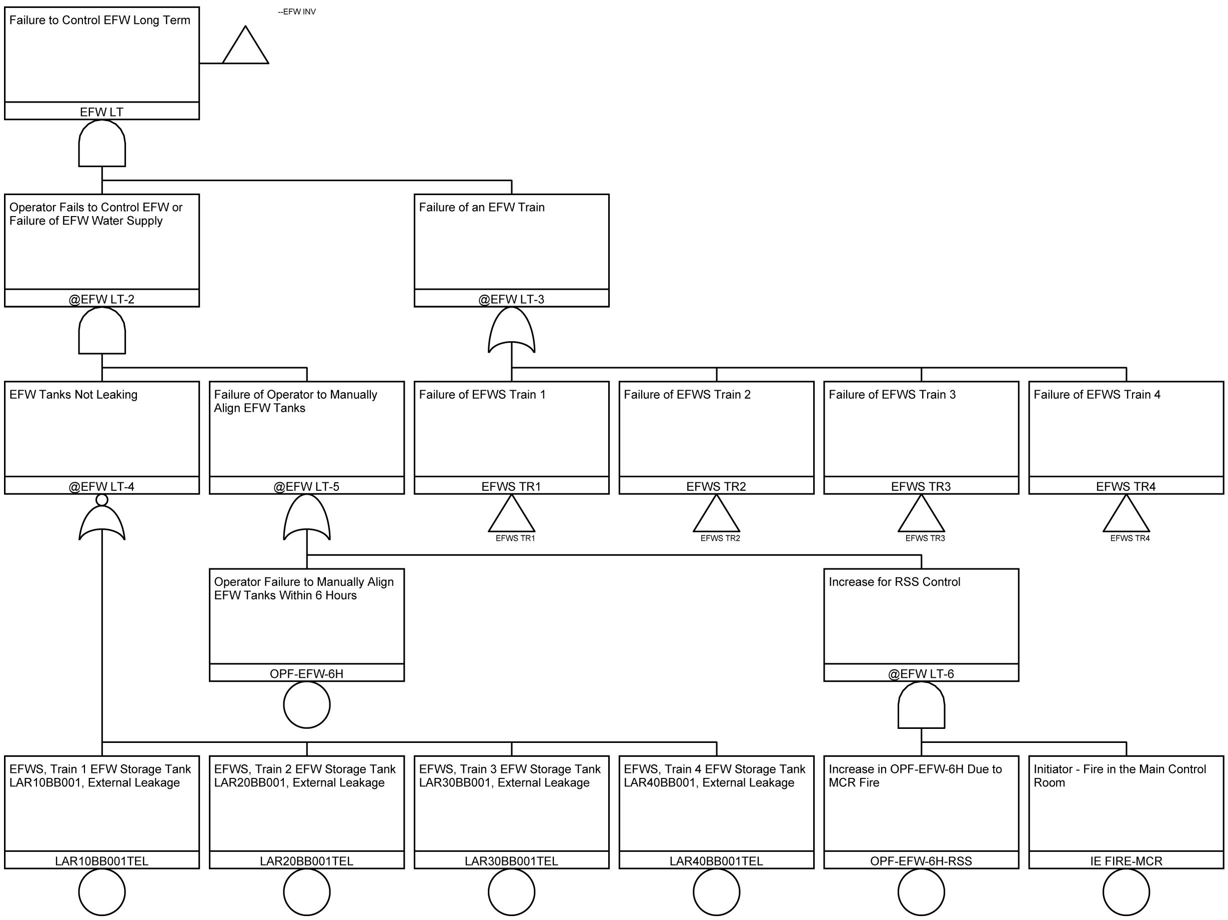


Figure 19-297-2 Fault tree "EFW LT": Failure to Supply Long-Term EFW Inventory when one EFW Train is Unavailable

EFW LT **RP192742**



Question 19-298:

(Follow-up to Question 19-277) In response to Question 19-277, AREVA evaluated hydrogen deflagration loads in containment assuming 75 and 50 percent passive autocatalytic recombiner (PAR) availability. The staff also noted that AREVA also added an insight to FSAR Table 19.1-109 recommending PAR availability during shutdown. However, PAR availability is not required by technical specifications. Thus, the staff requests information concerning how many PAR units have to be out of service to reach the containment hydrogen mass of 940 kg which was used for deflagration load calculations in the shutdown Level 2 analysis.

Response to Question 19-298:

A response to this question will be provided by August 28, 2009.

Question 19-299:

(Follow-up to Question 19-282) In response to Question 19-282, the assumption that the remote shutdown station (RSS) is available was added to FSAR Table 19.1-109. However, because fires during shutdown are not modeled in the PRA, the risk impact of MCR fires during shutdown is unclear. The staff observes that the low CDF from main control room (MCR) fires occurring at power depends on a fire frequency of 4.2E-4/yr and an HEP for the transfer action of 7E-5. The response to Question 19.01-34 states that MCR fires are assumed to cause a loss of balance of plant (LBOP), after which 90 minutes are available to take action; the HEP is based on these assumptions. The staff requests additional information on MCR fires during shutdown, specifically:

a. Discuss whether the at-power HEP for transfer to the RSS is also applicable to shutdown scenarios, assuming that the RSS is available during shutdown. In the response, discuss the initiating event assumed to occur following a MCR fire during shutdown and how long the operator has to respond.
b. Discuss whether the MCR fire frequency assumed in the at-power model is also applicable to shutdown.
c. Justify the lack of technical specifications (TS) for the RSS during shutdown MODES 4, 5, and 6. The staff observes that other systems were added to TS during shutdown based on risk arguments (see the response to Question 19-269b). Although the mitigation credit (based on realistic availability of the RSS and the associated HEP) is not currently known, MCR fires without the RSS available would lead directly to core damage using the assumptions from the at-power PRA.

Revise the FSAR to include a summary of this information, which is needed for the staff to conclude that the risk from MCR fires during shutdown is insignificant or enveloped by the at-power estimate.

Response to Question 19-299:

Response to Question 19-299a:

Fires in the MCR are treated conservatively in the at-power fire probabilistic risk assessment (PRA) by assuming that an initiating event (loss of balance of plant is modeled) would occur following the fire. This is a conservative assumption because the plant is likely to be manually shutdown with the main feedwater and startup and shutdown systems still available. The time available to evacuate the control room is based on the time in which the steam generator inventory would provide cooling in case of a loss of the emergency feedwater system following the LBOP initiating event, or 90 minutes, because during this time no operator action would be required.

In shutdown, the loss of the MCR would not, by itself, result in an initiating event, because the running residual heat removal (RHR) train(s) would continue to operate without need for operator intervention. A fire in the MCR could increase the chance of operator error during the drain-down to mid-loop in plant operating state (POS) CBd or POS Du and impact the frequency of uncontrolled level drop (ULD) initiating event. A fire could also disable the MCR until controls are restored, either by recovering the MCR or transferring to the RSS. During that time, the MCR would not be available to perform operator actions that may be needed to respond to an initiating event in shutdown.

Two scenarios are evaluated in order to assess the importance of RSS in shutdown:

1) Fire in the MCR increasing the frequency of the ULD initiating event in states CBd and Du

2) Fire in the MCR disabling operator actions in all shutdown (SD) plant operating states (POS).

These two scenarios are further discussed in the response to Question 19-299c.

The different initiating events modeled in the shutdown PRA are reviewed to identify the cases where operator actions would be required in the absence of additional system failures. Only the loss of RHR in state D and the uncontrolled level drop would always necessitate an operator action for successful mitigation (low head safety injection (LHSI) pump start in state D following a loss of RHR and isolation of the chemical and volume control system (CVCS) low pressure reducing station). The time available for these two actions is more than two hours (LHSI pump start) and more than eight hours (isolation of the CVCS). Thus, the 90-minute time window for evacuation of the MCR is conservative for shutdown fires in the MCR.

Response to Question 19-299b

The MCR fire frequency of 3.6E-4 per year used in the model is based on the RES/OERAB/S02-01 frequency for at-power main control room fires multiplied by a factor of 0.1 for suppression and 0.5 to take limited credit for fiber optic cables and digital workstations.

This number is comparable to the frequency of 2.6E-4 per year shown in the response to RAI 97 Question 19-223, which is based on NUREG/CR-6850 method. The relevant fire ignition sources from NUREG/CR-6850 (main control board and transients) show the same frequency independent of the mode of operation. Therefore, a frequency of 3.6E-4 per year could be considered appropriate for both at-power and shutdown.

Response to Question 19-299c

To assess the risk significance of having the remote shutdown station unavailable during shutdown, two sensitivity cases are used to evaluate the risk of fires in the MCR assuming no evacuation is possible. These two sensitivity cases are based on the two possible outcomes defined in Response to Question 19-299a.

1) Fire in the MCR increasing the frequency of ULD initiating event in states CBd and Du: The sensitivity case assumes that a fire in the MCR would occur with a daily frequency of 3.6E-4 / 365 = 9.9E-7. This frequency is applied to each of the two drain down phases modeled in the representative outage. This is conservative since the actual time spent actively draining down would be less than a day. The human error that dominates the ULD initiating event frequency is "OPF-ULD", with an estimated HEP of 1E-2. It represents the operator failure to stop draining while in the process of draining down to mid-loop. If a fire related abnormal situation occurs in the control room while the operator is draining down, the most likely response would be to immediately stop draining and isolate the reducing station. However, a factor of 10 increase is assumed to account for extreme stress and abnormal situation, resulting in a modified HEP of 0.1 for

OPF-ULD. The resulting core damage frequency (CDF) for this sensitivity case is 7.7E-10 per year.

2) Fire in the MCR resulting in the unavailability of controls. Scenarios which do not result in an initiating event are not usually modeled in the fire PRAs. This scenario is included for completeness in the sensitivity study. For the purpose of this evaluation, it was assumed that the time to restore controls, given an MCR fire, will also be 24 hours, by either recovering the MCR or using the RSS. During that time the ability to perform any operator action is assumed to be lost. Conservatively, no local operator actions are credited. The shutdown PRA model is quantified with all operator actions set to failure. The resulting delta CDF for this sensitivity case is 8.9E-11 per year

The total resulting delta CDF for both sensitivity cases is 8.6E-10 per year. This number conservatively represents the shutdown CDF due to fires in the MCR if the RSS is always unavailable, and corresponds approximately to 1.5 percent of the shutdown CDF of 5.8E-8 per year.

Based on the above results, having the RSS unavailable during an outage would not cause an unacceptable increase in shutdown risk and therefore it would not warrant inclusion in TS based on a risk argument.

Response to Question 19-299d

In the response to RAI 197, Question 19-282, the assumption that the RSS is available in all POS has been added to FSAR Tier 2, Table 19.1-109. This assumption reflects an engineering judgment regarding the most likely configuration in shutdown, but is not necessary to demonstrate that the risk at-power envelops the shutdown risk.

In fact, the CDF from this scenario at-power (2.5E-8 per year), scaled down to the time assumed spent in shutdown (21 days), yields a shutdown CDF of 1.4E-9 per year, which is higher than the CDF of the sensitivity case shown above in the response to Question 19-299c. This shows that even if the RSS was unavailable, the at-power treatment would still be conservative for shutdown.

FSAR Tier 2, Section 19.1.6.1.8 (Fire and Flooding Events in Shutdown) will be modified to add the following sentence. “The risk from a fire in the main control room at-power also envelops the risk in shutdown. The assumption made at-power of core damage if the operators fail to evacuate is conservative for shutdown, where loss of the MCR would not directly result in an initiating event.”

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 19.1.6.1.8 will be revised as described in the response and indicated on the enclosed markup.

Question 19-300:

Follow-up to Question 19-223 of RAI 97 - The following findings and questions relate to the sensitivity study discussed in the response to Question 19-223.

1. Please provide the number of each ignition source in each PFA and the total number of items per equipment type in the generic locations that were used to establish the ignition source weighting factor.
2. Provide the basis for the exclusion of air compressors, dryers, hydrogen tanks/fires from the fire frequency estimate and justify the impacts of these components on the fire frequency.
3. Justify the exclusion of cable fires (including self-ignited cable fires) from the fire frequencies and why cable fire impacts can be neglected from the fire PRA.
4. Which PFA does the fire scenario “Cabinet fires resulting in a loss of single bus” in Table 19-223-2 belong to?
5. Provide details on how the weighting factors for ignition frequency bins involving transient combustibles/ activities were estimated and distributed.

Response to Question 19-300:

Response to Question 19-300-1:

The elements used to calculate the ignition source weighting factors are shown in Table 19.300-1. The numbers shown in Table 19.300-1 are the basis for the percentages shown in RAI 97, Table 19.223-1, as summarized below:

- Percentages were rounded in Table 19.223-1 so that the accuracy shown is consistent with the level of knowledge.

- Transients and transients caused by welding and cutting were grouped in a single category called “Transients” in Table 19.223-1. The percentages allocated to each PFA are a weighted average of the normalized ratings for the two types of transients.

- The Turbine Building (TB) ignition sources (boiler, main feedwater pumps, turbine generator excitor, turbine generator hydrogen, turbine generator oil, transients) were grouped in a single category “total Turbine Building” in Table 19.223-1. The fire ignition frequency for that group is the sum of the fire ignition frequencies for each element of the group. This grouping is possible because the TB is a single PFA.

- The percentages for the “rest of the plant”, such as areas that are not part of any PFA, shown in Table 19.223-1, are calculated by subtracting the sum of the percentages for all the PFAs from 100 percent.

Response to Question 19-300-2:

Hydrogen tanks and fires are included in the fire frequency for the TB because they are part of the bin “turbine generator hydrogen”. This bin was subsumed in the general category “Turbine Building” in RAI 97, Table 19.223-1.

The air compressors which supply the plant instrument air are expected to be located in the TB. Since the scope of the air compressor bin is limited to the main instrument air compressors, it is reasonable to assume that the frequency of the bin (2.4E-3 per year) can be added to the TB. This would change the fire frequency in PFA-TB from 5.7E-2 per year to 5.9E-2 per year and likewise the CDF from that scenario from 6.8E-10 per year to 7.0E-10 per year. The impact on the fire CDF shown in response to RAI 97, Question 19-223 is negligible.

The “dryers” bin in NUREG/CR-6850 corresponds to clothes dryer, for which design information will be developed later in the design process. It is unlikely that these components would be located in any of the areas modeled in the fire PRA. The impact of their exclusion is judged to be insignificant.

Response to Question 19-300-3:

Cable fires are not included because details of cable routing, which is essential to derive the cable load weighting factors, will be developed later in the design process. Moreover, the U.S. EPR cables will be qualified by the Institute of Electrical and Electronics Engineers, and will therefore not be susceptible to self-ignition. Therefore, the only relevant fire ignition frequency for cables would be the cable fires caused by welding and cutting, which have a frequency of 1.6E-3 per year (Nuclear Auxiliary Building) and 2E-3 per year (plant-wide).

The distribution of these frequencies would be influenced by the cable loading and the transient ratings of the PFAs. The transient ratings for welding and cutting are discussed below in the response to Question 19-300-5 and are shown in Table 19.300-3. Table 19.300-3 shows that all the PFAs have a welding and cutting transient rating of less than five percent, except for the Fuel Building, which has a very low contribution to the fire risk. Therefore, the impact of this exclusion is judged to be small.

Response to Question 19-300-4

The fire scenario “Cable fires resulting in the loss of a single bus” shown in RAI 97, Table 19.223-2 models a fire occurring in one of the safeguard building switchgear rooms (PFA-SB4-AC or PFA-SB2-AC). It represents a low heat release fire that only affects the source electrical cabinet. This scenario is modeled by applying the conditional core damage probability of internal initiating event “31BDA” to the fire frequency of this scenario. This is equivalent to assuming that the disabled bus is the main emergency switchgear of Division 1.

Response to Question 19-300-5

Transient fire ratings are calculated following the relative ranking approach presented in NUREG/CR-6850, Section 6.7.5.2. Transient fire ignition frequencies are distributed between all the buildings of the plant, not only the ones modeled in the fire PRA. Influence factors such as maintenance, occupancy, and storage are assessed for each fire area described in the fire hazard analysis, based on engineering judgment.

The assigned influence factors for each fire area, and the resulting transient normalized ratings for each building, or group of buildings (safeguard building mechanical areas), are shown in Table 19.300-2. The normalized transient ratings are the product of the number of buildings in the group times the number of fire areas in each building times the normalized transient rating of each area.

The normalized transient rating for a given PFA can be obtained by multiplying the rating of their building by the fraction of the building fire areas that are located in the PFA. The calculation of the normalized transient ratings for each PFA is shown in Table 19.300-3.

The following assumption will be added to Table 19.1-109:

> "It is assumed that when the final number of fire ignition sources is known for each PRA fire area, the conclusion that fire ignition frequencies obtained using RES/OERAB/S02-01 are comparable to those obtained by using NUREG/CR-6850 will remain valid."

FSAR Impact:

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

Table 19-300-1: Number of Fire Ignition Sources Considered in Evaluating the Weighting Factors for each PFA

Location	**Battery Room**	**Control room**	**Control/Aux/Reactor Building**	**Control/Aux/Reactor Building**	**Diesel Generator Room**	**Plant-Wide Components**	**Plant-Wide Components**	**Plant-Wide Components**	**Plant-Wide Components**	**Plant-Wide Components**	**Plant-Wide Components**	**Plant-Wide Components**	**Plant-Wide Components**	**Turbine Building**	**Turbine Building**	**Turbine Building**	**Turbine Building**	**Turbine Building**	**Turbine Building**	**Turbine Building**
Ignition Source	Batteries	Main Control Board	Transient fires caused by welding and cutting	Transients	Diesel Generators	Battery Chargers	Electric Motors	Electrical Cabinets	Pumps	Transformers (Dry)	Transient fires caused by welding and cutting	Transients	Ventilation Subsystems	Boiler	Main Feedwater Pumps	Turbine Generator Excitor	Turbine Generator Hydrogen	Turbine Generator Oil	Transient fires caused by welding and cutting	Transients
Generic Frequency [1/yr]	7.5E-04	2.5E-03	9.7E-03	3.9E-03	2.1E-02	1.8E-03	4.6E-03	4.5E-02	2.1E-02	9.9E-03	4.9E-03	9.9E-03	7.4E-03	1.1E-03	1.3E-02	3.9E-03	6.5E-03	9.5E-03	8.2E-03	8.5E-03
Total in Plant	8	1	1	1	6	65	2013	1361	375	49	1	1	871	1	4	1	1	1	1	1
SB4-AC			4.67E-03	3.66E-03		0		168		5										
SB2-AC			4.67E-03	3.66E-03		0		108		4										
SB4-DC			4.67E-03	3.66E-03		11		32		0										
SB2-DC			4.67E-03	3.66E-03		13		29		0										
SB4-MECH			1.40E-02	1.28E-02			89	4	20				16							
BATT-4	1		4.67E-03	3.66E-03																
FB			1.96E-01	1.79E-01			245	11	26				175							
ESW4			1.40E-02	1.28E-02			12	18	5	1			3							
TB							97	8	77				22	1	4	1	1	1	1	1
SWGR	4				2	17		250		14	4.35E-02	4.67E-02								
MCR		1	4.67E-03	2.19E-02																

Table 19-300-2: Transient Normalized Rating for Different Plant Buildings

Building ID	Building description	# of buildings	# of fire areas (FHA)	Mainte-nance	Occu-pancy	Storage	Transients caused by welding and cutting factor	Transients factor
Aux/Reactor/Control Buildings								
RB	Containment	1	1	0	0	0	0	0
SB-Mech	SB Mech. Area	4	4	3	1	3	2.2E-01	2.0E-01
SB-Elec	SB Electrical Area	4	3	1	1	0	5.6E-02	4.4E-02
MS/MF VR	MS/MF valve room	2	1	0	0	0	0	0
FB	Fuel Building	1	14	3	1	3	2.0E-01	1.8E-01
MCR	Control Room	1	1	1	10	1	4.7E-03	2.2E-02
NAB	Nuclear Aux Building	1	13	3	3	3	1.8E-01	2.1E-01
EPGB	EPG Building	4	3	3	1	3	1.7E-01	1.5E-01
RadWaste	Radwaste Building	1	8	3	3	3	1.1E-01	1.3E-01
ESW PH	ESW Pump structure	4	1	3	1	3	5.6E-02	5.1E-02
Total Aux/Reactor Building				**214**	**131**	**202**	**214**	**547**
Turbine Building								
TB	Turbine Building	1	1	10	3	10	1	1
Total Turbine Building				**10**	**3**	**10**	**10**	**23**
Other Areas								
SWGR	Switchgear Building	1	4	1	1	3	8.7E-02	9.3E-02
Access	Access Building	1	11	3	10	3	7.2E-01	8.2E-01
Gen xFers	Generator Transformer Areas	1	4	1	1	0	8.7E-02	3.7E-02
Aux xFers	Auxiliary Power Transformer Areas	1	5	1	1	0	1.1E-01	4.7E-02
Total other areas				**46**	**123**	**45**	**46**	**214**
Total plant-wide				**270**	**257**	**257**	**270**	**784**

Table 19-300-3: Transient Normalized Ratings for PRA Fire Areas (PFA)

	Building	Bin	Ratio of FAs in the PFA / FAs in Bldg	Transient (welding & cutting) Factor	Transient factor
SB4-AC	SB-Elec	Aux. Bldg	1/3	4.7E-03	3.7E-03
SB2-AC	SB-Elec	Aux. Bldg	1/3	4.7E-03	3.7E-03
SB4-DC	SB-Elec	Aux. Bldg	1/3	4.7E-03	3.7E-03
SB2-DC	SB-Elec	Aux. Bldg	1/3	4.7E-03	3.7E-03
SB4-MECH	SB-Mech	Aux. Bldg	1/4	1.4E-02	1.3E-02
BATT-4	SB-Elec	Aux. Bldg	1/3	4.7E-03	3.7E-03
FB	Fuel Building	Aux. Bldg	14/14	2.0E-01	1.8E-01
ESW4	ESW Pump Structure	Aux. Bldg	1/1	1.4E-02	1.3E-02
TB	Turbine Building	TB	1/1	1	1
SWGR	Switchgear Building	Other	2/4	4.3E-02	4.7E-02
MCR	Main Control Room	Aux. Bg	1/1	4.7E-03	2.2E-02

Question 19-301:

The use of either NUREG/CR-6850 main control board fire frequency or RES/OERAB/S02-01 control room fire frequency to represent EPR control room fire may not be appropriate. The fire ignition frequencies provided in these documents are primarily derived from the existing power plants equipped with analog technology. However, per design, the EPR main control room is a compact cockpit-style; a workstation which is entirely driven by digital computers and visual display monitors rather than analog hardware. Thus, please demonstrate that using MCR ignition frequency of either 7.2E-3/yr or 2.6E-3/yr is realistic and practical for EPR design.

Response to Question 19-301:

As shown in the response to Question 19-299b, the U.S. EPR fire PRA takes limited credit for the digital nature of the main control room in evaluating the fire ignition frequency. A factor of 0.5 is applied to the RES/OERAB/S02-01 control room fire frequency of 7.2E-3 per year to account for the digital design of the control room, including fiber optic cables which are not susceptible to self-ignition and the presence of computers instead of analog control panels.

There is no industry data available regarding the fire ignition frequency for digital control rooms; and the use of existing data for analog control rooms would be conservative. Applying the 0.5 factor allows some credit to be taken for the improvement in fire risk brought by the digital design, while still producing a conservative frequency.

FSAR Impact:

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

Question 19-302:

Follow-up to Question 19.01-29 of RAI 66 - The response to Question 19.01-29 states that “Due to the large size and small combustible loading of the PFA, a fire that would affect all components is not postulated. Instead, a specific analysis of vulnerable locations is performed. Reactor coolant pump fires due to oil leakage have been the source of most fires in-containment in operating history. Due to the specific oil collecting system described in U.S. EPR FSAR Tier 2, Section 5.4.1.2.2, it was concluded that this event could not occur in the U.S. EPR. Reactor coolant pump fires are therefore not analyzed as a credible fire scenario.” Per your conclusion, please explain how it was determined that coolant pump fires could not occur.

In addition, the RCP fire frequency of 6.1E-3/yr provided in NUREG/CR-6850 was derived from the RCPs that are also equipped with oil collection systems. Thus, neglecting this frequency and using 1.9E-5/yr for containment fires is not practical; please revise fire area PFA-CNTMT frequency to correctly address RCP and containment fires or provide further justification for the exclusion.

In addition, provide the justification for excluding containment transients and hotwork (Bin 3 of NUREG/CR-6850 Table 6-1). Note that, the response to Question 19.01-29 does not address justification for the exclusion of transient ignition frequency as requested.

Response to Question 19-302:

The sentence from the response to RAI 66, Question 19.01-29 was not intended to mean that reactor coolant pump (RCP) fires could not occur. Instead, the meaning of that sentence was that RCP oil fires with a high heat release were extremely unlikely, and therefore were not considered as a credible fire scenario in the containment. Small fires such as motor fire, limited oil fire, and pump casing insulation fire could occur, but would only affect the pump itself. The consequences of such an event would be limited to a reactor trip with one RCP unavailable, and its frequency would be negligible compared to the general transient frequency.

The oil collection system described in U.S. EPR FSAR Tier 2, Section 5.4.1.2.2 is designed such that an oil leakage from any location of the pump is captured and directed to an oil collection tank. Therefore, an oil spill onto the containment floor, which is a prerequisite to a significant fire, is precluded. The oil collection system is a passive component; the probability of its failure combined with an oil leak and a fire ignition would be extremely low.

To justify the exclusion of the RCP fires from the fire scenarios considered in the containment, a few possible RCP fire scenarios analyzing fires of the pump itself, and pump fires including possible failures of the oil collection system are evaluated as a screening analysis. Two failure modes of the oil collection system are considered with conservative likelihoods assigned: a minor leak with an estimated likelihood of 0.1, and a major oil spill with an estimated likelihood of 0.01. The RCP fire ignition frequency taken from NUREG/CR-6850 is the following: total frequency for the RCP bin: 6.1E-3 per year, 14 percent from electrical fires (8.5E-4 per year) and 86 percent from oil fires (5.2E-3 per year). The results of the evaluation are presented in Table 19.302-1. Fire suppression, even if it is available for the RCP fires, was not credited in the screening analysis. As it can be seen from this screening table, the RCP fires scenarios contribute 0.2 percent to the total fire CDF and can be screened from further analysis.

As explained in the response to RAI 66, Question 19.01-29, a fire occurring within the pressurizer compartment was judged to be the limiting fire scenario in the containment. With a frequency of 1.9E-5 per year, this scenario has a lower frequency than analyzed RCP fire scenarios, but higher consequences (CDF of 1.7E-9 per year).

Table 19-302-1: Screening Analysis of the RCP Fires Scenarios

RCP Fire Scenario	Consequences	Frequency (1/yr)	CCDP	CDF (1/yr)	% of Fire CDF
Pump Fire	Loss of one pump	6.1E-03	3.6E-08	2.2E-10	0.12%
Pump Oil Fire with a Failure of Lube Oil Collection System (limited leak)	Loss of one SG	5.2E-04	2.1E-07	1.1E-10	0.06%
Pump Oil Fire with a Catastrophic Failure of Lube Oil Collection System (major spill)	Loss of two SGs	5.2E-05	1.1E-06	5.7E-11	0.03%

FSAR Impact:

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

Question 19-303:

Please clarify whether the biological events are included in the loss of condenser heat sink frequency, loss of balance of plant frequency, and/or loss of main feedwater frequency used in the US EPR PRA or not.

Response to Question 19-303:

As stated in U.S. EPR FSAR Tier 2, Table 19.1-4, the loss of condenser (LOC) frequency and loss of main feedwater (LOMFW) are selected from NUREG/CR-6928. In NUREG/CR-6928, Section D.2.8.1 Initiating Event Description, it is stated that the LOC heat sink frequency for pressurized water reactors includes events where one or more conditions are met. One of these conditions is:

> "A decrease in condenser vacuum that leads to an automatic or manual reactor trip, or manual reactor trip; or a complete loss of condenser vacuum that prevents the condenser from removing decay heat after a reactor trip. In addition, reactor trips that are the indirect result of a low condenser vacuum, such as a loss of feedwater caused by condensate pumps tripping on high condensate temperature because of loss of vacuum are counted."

NUREG/CR-5750 page 49 includes additional information on the scope of the loss of condenser heat sink frequency. Specifically, the loss of condenser vacuum (a subset of loss of condenser heat sink) is described as having two major causes, which are:

- Problems directly related to the condenser.
- Problems associated with the circulating water systems (i.e., pump trip, traveling screen blockage).

Based on the NUREG descriptions above, biological events are considered to be included in the LOC frequency. This is because biological events could result in traveling screen blockage and loss of condenser vacuum.

The loss of balance of plant (LBOP) frequency is calculated using a fault tree, where one contributor is a failure of the normal heat sink, an undeveloped event whose unavailability corresponds to a failure frequency of 1E-02 per year. This frequency is based on the loss of condenser vacuum frequency discussed above and described in NUREG/CR-5750. Because this frequency is also related to a loss of condenser heat sink frequency, biological events are considered to be included in the LBOP frequency.

The LOMFW initiating event description is provided in NUREG/CR-6928 Section D.2.12.1. However, main feedwater pump trip due to biological effects is expected to occur by clogging cooling to the main condenser, resulting in low condenser vacuum and subsequent main feedwater pump trip. These types of initiating events are not included in the LOMFW initiating event and are instead included in the Loss of Condenser Heat Sink frequency, as shown in the NUREG/CR-6928 Section D.2.8.1 quote above.

Based on the above discussion, initiating events caused by biological effects (affecting a loss of condenser vacuum) are included in the LOC and LBOP frequencies. Both of these events are assumed to disable MFW.

FSAR Impact:

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

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of the plant, but are judged to be not significant for the protected RHR trains providing decay heat removal. The risk from a fire in the main control room at-power also envelops the risk in shutdown. The assumption made at-power of core damage if the operators fail to evacuate is conservative for shutdown, where loss of the MCR would not directly result in an initiating event.

Additionally, the following fire and flooding events that could cause scenarios specific to shutdown are identified:

- Flooding in the annulus that propagates to two Safeguard Buildings (SB), disabling both running residual heat removal (RHR) trains.
- Fire-induced hot short that causes an uncontrolled level drop.
- Fire-induced hot short that causes a flow diversion due to spurious operation of a motor-operated valve.

The frequency of each of these three scenarios is evaluated. In each case, it is found to be at least two orders of magnitude less than the frequency of the equivalent initiating event in the internal event LPSD PRA (i.e., loss of RHR, uncontrolled level drop and flow diversion LOCA).

The effect of each of these three scenarios on mitigating systems is also evaluated, and sensitivity studies are performed to evaluate the increase in shutdown risk posed by these initiators. The relative change in CDF is found to be negligible for loss of RHR and uncontrolled level drop, and very small (2 percent) for the RHR flow diversion. This is due to the low frequency of these events and their limited impact on mitigating systems.

Based on the bounding nature of the at-power fire and flood evaluations and on the low risk impact of shutdown-specific internal hazards, the risk from fire and flood events during at-power operation is assumed to envelop the risk during shutdown.

19.1.6.2 Results from the Low-Power and Shutdown Operations PRA.

19.1.6.2.1 Risk Metrics

The total CDF from shutdown events is 5.8E-08/yr, well below the NRC safety goal of 1E-04/yr (SECY-90-016) and the U.S. EPR probabilistic design goal of 1E-05/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.6.2.7.

19.1.6.2.2 Significant Initiating Events

The significant shutdown initiating events and their contribution to shutdown core damage frequency are given in Table 19.1-90—U.S. EPR Significant Initiating Events Contributions - Level 1 Shutdown. Only those initiating events that contribute more

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No.	Category[1]	PRA General Assumptions[2]
78	Fire	The entire Transformer Yard is considered a single fire area and is physically separated from other plant structures. Separation will be assured by non-rated exterior barriers and distance. These factors will prevent a fire in the Transformer Yard from propagating to other plant structures. In the fire risk evaluation, it is also assumed that fire protection features will be designed to prevent fire propagation between transformers.
78a	Fire	PRA General Assumption: It is assumed that when the final number of fire ignition sources is known for each PRA fire area, the conclusion that fire ignition frequencies obtained using RES/OERAB/S02-01 are comparable to those obtained by using NUREG/CR-6850 will remain valid.
79	Seismic	When equipment is not seismically qualified by analysis or testing or anchorage design is not complete, the seismic analysis is based on the seismic design criteria and qualification methods normally followed in the nuclear industry.
80	Seismic	Seismic-induced LOOP, LOCA and ATWS events are assumed to dominate all potential initiating events. Equipment and structures that are not seismically qualified are not credited in the model. The key assumptions regarding system availability and operator response are given below: • Seismic-induced LOOP is assumed not to be recoverable. • Station Blackout (SBO) Diesels are assumed to fail as a result of a SSE. • All systems that depend on normal AC power such as main feedwater, main condenser, Startup and Shutdown System (SSS) pump, and their support systems are assumed to fail as a result of a SSE. • Operator actions in response to seismic events are not credited. • RCP seal injection with CVCS is assumed to be lost due to a seismic event. • CVCS makeup to the Reactor Pressure Vessel (RPV) and auxiliary pressurizer spray are assumed to fail as a result of a SSE. • Dedicated Relief Valves (DRV) are assumed to fail as a result of a SSE. • Severe Accident Heat Removal (SAHR) is assumed to fail as a result of a SSE.
81	Seismic	The PRA-based seismic margin assessment assumes that equipment will be installed as designed and that there are no potential spatial interaction concerns in the as-built configuration (e.g., adjacent cabinets are bolted together, collapse of non-seismically designed equipment or masonry wall onto safety-related equipment is precluded, and no likelihood of seismically-induced fire or flood impacting safety-related equipment).

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