AREVA

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Washington, D.C. 20555-0001

Response to U.S. EPR Design Certification Application RAI No. 197

Ref. 1: E-mail, Getachew Tesfaye (NRC) to Ronda Pederson, et al (AREVA NP Inc.), "U.S. EPR Design Certification Application RAI No. 197 (2202), FSAR Ch. 19," March 11, 2009.

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 197 Response US EPR DC.pdf" provides technically correct and complete responses to 9 of the 10 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 197 Questions 19-274, 19-277, 19-279, 19-281, 19-282 and 19-283.

The following table indicates the respective pages in the response document, "RAI 197 Response US EPR DC.pdf" that contain AREVA NP's response to the subject questions.

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A complete answer is not provided for 1 of the 10 questions. The schedule for a technically correct and complete response to this question is provided below.

Question #	Response Date
RAI 197 — 19-275	May 15, 2009

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D077
NRO

If you have any questions related to this submittal, please contact me at 434-832-2369 or by e-mail at sandra.sloan@areva.com

Sincerely,

Sandra M. Sloan

Sandra M. Sloan, Manager
New Plants Regulatory Affairs
AREVA NP Inc.

Enclosures

cc: G. Tesfaye
Docket No. 52-020

AFFIDAVIT

COMMONWEALTH OF VIRGINIA	)	
	)	ss.
COUNTY OF CAMPBELL	)	

1. My name is Sandra M. Sloan. I am Manager, New Plant Regulatory Affairs for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA NP information contained in "Response to U.S. EPR Design Certification Application RAI No. 197" and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

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

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

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

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

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

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

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

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

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

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

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

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

Sandra M. Sloan

SUBSCRIBED before me this 10th

day of April, 2009.

Kathleen A. Bennett

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2011

KATHLEEN ANN BENNETT
Notary Public
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My Commission Expires Aug 31, 2011

Response to

Request for Additional Information No. 197 (2202), Revision 0

03/11/2009

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 19-274:

(Follow-up to Question 19-39) The response to Question 10.04.09-1 describes a design change to maintain the emergency feedwater system (EFWS) supply header isolation valves closed during operation, but does not evaluate the effect of this change on the probabilistic risk assessment (PRA). The staff observes that the interconnection of the EFWS headers is implicit in the "EFW PBF" top event (described in the response to Question 19-39) and contributes to the relative importance of the EFWS tanks in the PRA. Evaluate the effect of the design change on the PRA results and insights and either incorporate it in the PRA or discuss it in Section 19.1.2.4 of the Final Safety Analysis Report (FSAR).

Response to Question 19-274:

The impact on the PRA of the design change which maintains the EFWS supply header isolation valves closed is summarized below:

- If one or more EFW train is unavailable, a manual action would be required to interconnect the four EFW tanks to make the entire EFW inventory available. As discussed in the response to Question 19-39, PRA success criteria require the inventory of the four EFW tanks for a 24-hour mission time.
- In case of a pipe break or a tank leakage in the EFWS (internal flooding), the operators no longer have to isolate the leaking train to avoid losing all EFW inventory because the header valves are already closed. However, one tank inventory may still be lost and it would be necessary to refill one of the intact tanks in order to achieve the 24-hour mission time.

A sensitivity case is performed to assess the impact of this change on the PRA results and insights. The "EFW PBF" top event currently models only a pressure boundary failure of the EFWS. In that sensitivity case, this top event is re-defined to include the need for tank interconnection if one or more EFW train is unavailable. Two modeling cases are defined depending on the pressure boundary integrity:

(1) If one (or more) EFW trains are unavailable for reasons other than a pressure boundary failure, a simple tank interconnection is required. Operator error "OPF-EFW-6H" is defined as the failure to interconnect the tanks following the loss of one EFW train. This is a local action, performed from each of the four Safeguard Buildings, a short distance from the control room. This is a simple action, for which expansive time is available (6 hours). The calculated human error probability (HEP) for this action is 2E-5.

(2) If one EFW train is unavailable due to a pressure boundary failure, tank interconnection is not desirable and the available tank(s) should be refilled instead. The demineralized water distribution system (DWDS) or the fire water distribution system (FWDS) both can be used for that purpose. Operator error "OPF-EFW RF-6H" is defined as the failure to initiate tank refill. The same time is available (6 hours) but this action has increased stress and complexity due to multiple cues. The calculated HEP is 8E-4.

Operator failure to perform any of those two actions is assumed to result in a long-term failure of the EFWS due to insufficient water inventory, eventually requiring feed and bleed cooling. The dependency between failure to initiate feed and bleed and prior failure to interconnect or refill the tanks is assessed. Based on a similar crew, different time, different location and additional cues, a low level of dependency is selected.

Table 19-274-1 summarizes the modeling and human reliability analysis differences between having the valves open or closed.

The results of this sensitivity case are shown in Table 19-274-2. Total CDF increases by 5 percent, driven by a rise in internal event and flooding core damage frequency (CDF). The following PRA insights also can be drawn from this design change.

The rise in internal event and flooding CDF can be explained by the increase in CDF from two specific initiating events: loss of balance of plant (LBOP) and flood in Turbine Building. These two initiating events result in a loss of both main feedwater and startup and shutdown systems. With the EFW header valves initially closed, any EFWS train failure requires an operator action to mitigate these events.

Even though the overall internal flooding CDF increases by about 9 percent (driven by the Turbine Building flood), CDF from the EFW break scenario decreases as a result of this design change, from 7.2E-09/yr to 5.1E-09/yr. This is because there is no longer a need for isolating the break.

The following changes will be made to the U.S. EPR FSAR:

<u>Section 19.1.2.4 of the Final Safety Analysis Report (FSAR) will be updated as follows</u>:

- EFWS supply header isolation valves – This design change consists of maintaining the EFWS supply header isolation valves closed. If one or more EFW train is unavailable, a manual action is required to interconnect the four tanks, so that the entire EFW inventory is available. In case of a pipe break or a tank leakage in the EFWS (internal flooding), the operators no longer have to isolate the leaking train to avoid losing all EFW inventory. One tank inventory may still be lost; therefore, it is necessary to refill one of the intact tanks in order to achieve the 24-hour mission time.

- This change results in a measurable increase in internal event and internal flooding CDF, driven by operator failure to perform the interconnection. PRA insights and assumptions regarding manual isolation of an EFWS pressure boundary failure are also affected, as this isolation is no longer needed. This effect is recognized in U.S. EPR FSAR Tier 2, Table 19.1-108 Item 10, and Table 19.1-109 Item 66.

<u>Table 19.1-108, Item 10 will be updated as follows:</u>

"The severity of a flooding event from an EFW tank leak or pipe break will be reduced as a result of the design change identified in Section 19.1.2.4, which maintains the EFW suction header isolation valves closed. Manual isolation of an EFW tank leak or pipe break will not be necessary."

<u>Table 19.1-109, Item 66 will be updated as follows:</u>

"The severity of a flooding event from an EFW tank leak or pipe break will be reduced as a result of the design change identified in Section 19.1.2.4, which maintains the EFW suction header isolation valves closed. Manual isolation of an EFW tank leak or pipe break will not be necessary."

Table 19-274-1

Effect of EFW Header Valve Design Change in Accident Sequence Modeling

	EFW Header valves initially OPEN	**EFW Header valves initially CLOSED**
Response to the loss of 1 EFW train	Not required	Interconnection in 6 hours OPF-EFW-6H (2E-5)
Response to a pressure boundary failure	Isolation in 1 hour OPF-EFW-1H (4E-2)	Tank refill in 6 hours OPF-EFW RF-6H (8E-4)
Feed and Bleed following depletion of EFW inventory	Dependent action on OPF-EFW-1H OPD-FB-90M (0.05)	Dependent action on OPF-EFW-6H and on OPF-EFW RF-6H OPD-FB-90M (0.05)

Table 19-274-2

Results of the Sensitivity Case with EFW Header Valves Initially Closed

Hazard Group	**Base Case CDF (1/yr)**	**Sensitivity Case CDF (1/yr)**	**Delta CDF**
Internal Events	2.9E-07	3.1E-07	6%
Internal Fires	1.8E-07	1.8E-07	1%
Internal Flooding	6.1E-08	6.7E-08	9%
Total	5.3E-07	5.5E-07	5%

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 19.1.2.4, Table 19.1-108 and Table 19.1-109 will be revised as described in the response and indicated on the enclosed markup.

Question 19-275:

(Follow-up to Question 19-158) The new FSAR Section 19.1.6.3.1.4 provided with the response to Question 19-158 states that "the ability to close the hatch is credited" and that "the habitability of the containment (i.e., local temperature no higher than 122°F) is considered to be the most limiting criterion in determining the time available to close the hatch." This assumption is inconsistent with the outage guidelines in NUMARC 91-06, guideline 4.1.1(3), and item 15 in FSAR Table 19.1-8, which state that containment should be closed prior to steaming. The staff requests additional information to clarify the containment closure assumptions.

a. In FSAR Section 19.1.6.3.1.4, document the time to core boiling for plant operating state (POS) CA and CB (based on the outage schedule assumed in the shutdown PRA).
b. Revise and document in the FSAR the probabilities of the operator failing to close containment based on the time available for closure prior to steaming inside containment.

Response to Question 19-275:

A response to this question will be provided by May 15, 2009.

Question 19-276:

(Follow-up to Question 19-158) The new FSAR Section 19.1.6.3.1.2 provided with the response to Question 19-158 states that "[t]he primary system is considered pressurized in states CA and CB"; however, FSAR Table 19.1-87 indicates that the reactor coolant system (RCS) is vented in POS CB. The RCS vent could be large enough to preclude significant RCS re-pressurization given a complete loss of decay heat removal. Discuss the impact of the RCS vent in POS CB on the Level 2 shutdown analysis, and revise the PRA and/or FSAR as appropriate.

Response to Question 19-276:

The RCS openings during plant operating state (POS) CB are considered to reduce the degree of RCS repressurization given a loss of decay heat removal, but not to prevent it, because either they are not large enough or they have flow restrictors. Even though the RCS is identified as vented in U.S. EPR FSAR Tier 2, Table 19.1-87, RCS repressurization is considered in the shutdown Level 1 analysis.

In the shutdown Level 2 analysis, performed in support of the response to Question 19-158, supporting MAAP analyses show significant RCS repressurization in this POS; therefore, initiating events during POS CB are assumed to cause RCS repressurization. This is conservative because of the increased severity of core melts at high pressure. In general, depressurization is considered beneficial because it precludes high pressure RCS ruptures such as induced steam generator tube rupture, direct containment heating or vessel rocketing.

During POS CB, a complete loss of residual heat removal (RHR) could lead to RCS repressurization. While the extent of that repressurization could vary under different plant conditions, the assumption that the RCS is pressurized is conservative.

FSAR Impact:

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

Question 19-277:

(Follow-up to Question 19-158) Revised FSAR Section 19.1.6.3.3.4, provided in response to Question 19-158, states that the passive autocatalytic recombiners (PAR) are used for control of hydrogen and oxygen following a severe accident at shutdown. FSAR Table 19.1-23 states that operation of the PARs is "implicitly assumed." Discuss whether this assumption applies to the Level 2 shutdown PRA as well, or whether failures of the PARs are modeled. If the PARs are assumed not to fail, discuss the availability controls that ensure they will function when needed (e.g., not be disabled by a protective covering while maintenance is done in containment). As appropriate, include the importance of the PARs as a risk insight in the FSAR.

Response to Question 19-277:

In the shutdown Level 2 PRA a 100 percent availability of the PARs is assumed for containment failure probabilities calculation. The scenarios leading to partial unavailability of the PARs are described in the following discussion and the impact of low PAR availability on the shutdown Level 2 PRA is discussed.

The probabilities of containment failures due to hydrogen loads are either unchanged or slightly increased under certain conditions with partial availability of the PARs. However, the overall increase in the shutdown large release frequency (LRF) does not exceed 4 percent under the most limiting condition of very low availability of the PARs.

PARs Availability:

Several scenarios can lead to a fraction of the PARs being unavailable during an accident, when the release of hydrogen is the highest. During shutdown, the following cases can lead to a reduced PARs inventory at the beginning of the accident scenario:

- Scheduled maintenance; PARs plates removed, or
- PARs disabled during containment maintenance (e.g. protective covering during maintenance).

Scheduled Maintenance on PARs:

It is expected that no more than three catalytic plates from 20–25 percent of the PARs would be removed for maintenance. There are six small PARS containing several hundred plates and 41 large containing more than a thousand plates. The maximum expected unavailability resulting from such maintenance is less than 5 percent of the total catalytic plates during shutdown. Such a low unavailability is judged to have negligible impact on recombiner efficiency.

PARs Disabled due to Containment Maintenance:

The shutdown Level 2 PRA did not assume that PARs could be partially disabled during containment maintenance due, for example, to a protective covering. Because there are multiple PARs units installed at different locations inside the containment, simultaneous unavailability of multiple PARs due to containment maintenance is judged unlikely and is not modeled in the PRA. The following assumption will be added to U.S. EPR FSAR Tier 2, Table 19.1-109:

> "The efficiency of the Passive Autocatalytic Recombiners (PAR) during shutdown is assumed to be nominal. Maintenance unavailability, if any, is assumed to be limited to a small fraction of the PARs and would not affect the overall efficiency of the system."

Impact of the PARs Availability on the Shutdown Level 2 PRA:

Deflagration Loads

Two scenarios were chosen to illustrate the impact of PARs availability. MAAP case Ca_4_1c is representative of a transient in state Ca and Ca_4_3 is representative of a loss of coolant accident (LOCA) in state Ca. Both cases were run with 100 percent, 75 percent and 50 percent PARs availability, as shown in Figures 19-277-1 and 19-277-2.

MAAP cases with 75 percent and 50 percent PARs availability show higher quantities of hydrogen in containment than the shutdown base cases with 100 percent PARs availability. These higher quantities are bounded by the mass of hydrogen used in the shutdown Level 2 PRA for the following reasons:

- With 50 percent PARs availability, the maximum quantities of hydrogen present in the containment for transient and LOCA cases are about 700 and 600 kg respectively, as seen in Figures 19-277-1 and 19-277-2.
- The Hydrogen mass of 940 kg used for deflagration load calculations in the shutdown Level 2 analysis was derived following the full power methodology. The full power study considered a bounding range of Zircaloy oxidation between 48 percent and 82 percent and, on the basis of the 82 percent upper bound, assumed a maximum mass of Hydrogen in containment. On this same basis of 82 percent upper bound, the assumed peak mass of hydrogen in containment at shutdown is 940 kg.

The containment failure probabilities following hydrogen deflagration loads derived in the shutdown Level 2 are unchanged in the case of partial unavailability of the PARs.

Flame Acceleration Loads

The probability of containment failure following flame acceleration loads is the product of multiple partial probabilities. One of these is the probability of ignition during the period of high concentration, which relates to the duration of the flame susceptibility period and is impacted by the availability of the PARs.

To account for lower availability of the PARs, the initial probability of ignition (calculated with 100 percent PARs available) is scaled to the longer duration of hydrogen concentration above the flame acceleration limit.

Increasing the partial probability of ignition from 0.86, the value used in the shutdown Level 2 PRA, to 1 (reflecting the maximum value possible with no PARS available) leads to an increase of the containment failure probability during the in-vessel period of 16.3 percent and an increase in the shutdown LRF of 3.3 percent.

The initial values of the shutdown LRF and the early containment failure due to hydrogen flame acceleration are respectively reported in U.S. EPR FSAR Tier 2, Table 19.1-122 and Table 19.1-122.

The sensitivity calculation presented in this discussion demonstrated that the impact of the PARs availability on the LRF is not significant. Very low availability of the PARs can lead to an increase of less than 4 percent in the shutdown LRF.

Figure 19-277-1

Hydrogen Mass in Containment for Transient Case Ca4_1c

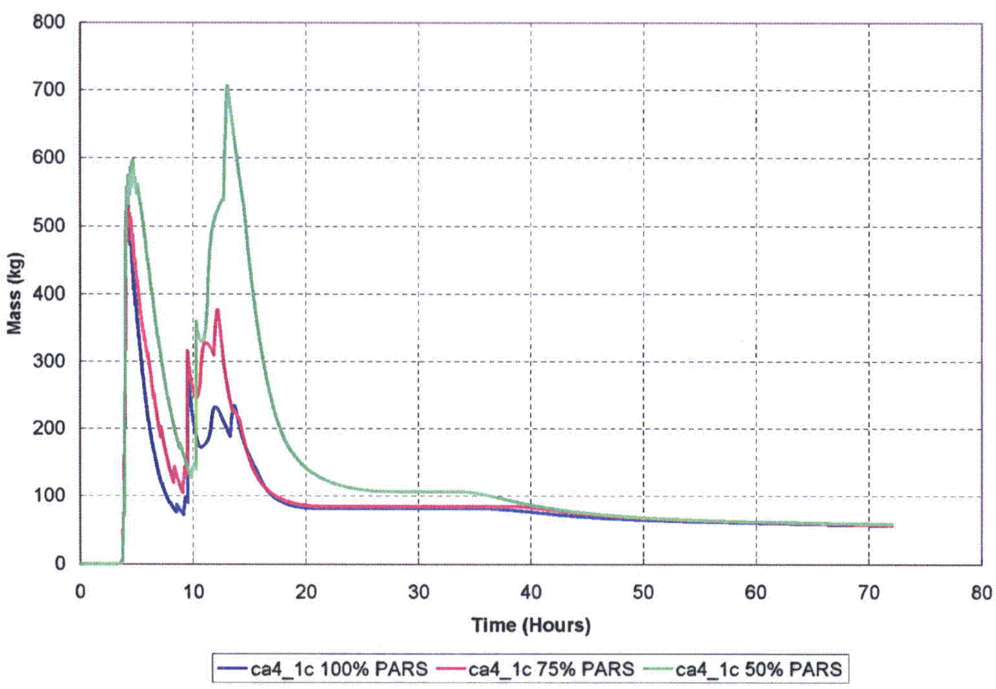
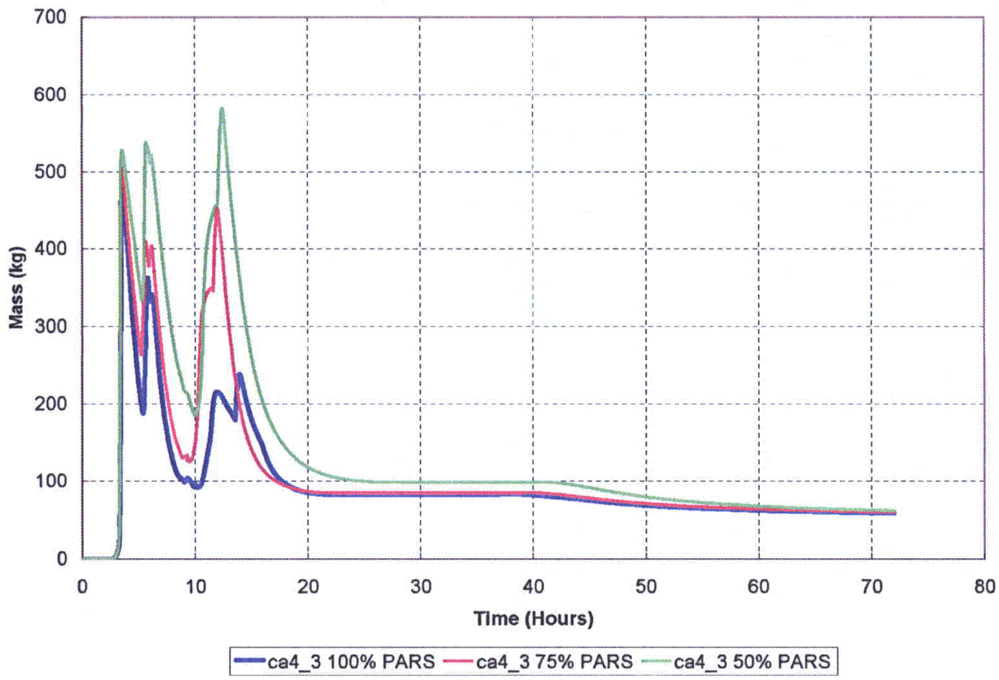


Figure 19-277-2

Hydrogen Mass in Containment for LOCA case Ca4_3



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.

Question 19-278:

(Follow-up to Question 19-162) In response to Question 19-162, FSAR descriptions of emergency diesel generator (EDG) and station blackout diesel generator (SBODG) diversity were revised. The staff needs additional information related to this diversity assumption to conclude that the PRA treatment of the EDGs and SBODGs is appropriate. Specifically:

a. Since that response, additional Tier 1 information on the EDG bypass exhaust path has been added in response to Question 09.05.08-8. Discuss whether the SBODG has a similar bypass exhaust path and whether common-cause failures (CCF) of the exhaust paths can potentially occur.

b. FSAR Table 14.3-6, which lists PRA input to the development of inspections, tests, analyses, and acceptance criteria (ITAAC), states that "SBO DGs are independent and diverse from the EDGs." However, the ITAAC in Table 2.5.3-2 of Tier 1 require only that the air start systems of the EDGs and SBODGs be independent. Discuss, given the general statement in Table 14.3-6, why only the air start system is addressed specifically, and revise the FSAR as appropriate.

Response to Question 19-278:

Additional information related to SBODG independence and diversity is provided as follows:

a. The SBODG has an exhaust bypass valve and bypass exhaust stack that is similar in function to the EDG exhaust bypass path. However, due to the potential failure modes, common-cause failures of EDG and SBODG exhaust paths requires multiple failures in diverse machines, resulting in an unlikely occurrence.

   Potential failure modes for the exhaust bypass include failing open, and a failure to open when actuated during diesel operation combined with unacceptable exhaust backpressure conditions. Exhaust bypass valve failing open does not adversely affect diesel engine operation for either the EDG or SBODG engines, so this failure mode is not considered.

   The actuation of the exhaust bypass valve occurs when diesel engine exhaust backpressure reaches an unacceptable level. For common-cause failure to adversely affect both an EDG and an SBODG, there would need to be exhaust system failures in the silencer, emission control equipment or piping that could restrict the exhaust gas flow and the subsequent failure of the exhaust bypass valve. Additionally, it is important to note that even following actuation of the exhaust bypass valve and failure of the valve to operate, the existing exhaust path is expected to remain in operation. An exhaust system failure is expected to be a slow degradation of system performance and not a catastrophic failure. In this event an alarm will alert the operators of the abnormal condition, exhaust system back pressure indication will show the severity of the exhaust system degradation and indicate to the operators to manually operate the exhaust system bypass or shutdown the EDG or SBODG in accordance with station procedures. If a catastrophic failure did occur (e.g., external missile) it is not probable it would affect both an EDG and SBODG, and the exhaust bypass valve of the affected EDG would not be affected as it is located within the Seismic Category I Emergency Power Generating Building structure. This physical separation of the EDGs and SBODGs eliminates this common-cause failure potential.

b. The combination of existing ITAAC, and ITAAC added in response to RAIs cited below demonstrate the independence and diversity of the SBODGs from the EDGs in accordance with regulatory guidance. U.S. EPR FSAR Tier 2, Section 8.4 indicates the SBODGs conform to diversity guidance provided by RG 1.155, Appendix B, with reference to RG 1.155 Regulatory Position 3.3.5. RG 1.155 Regulatory Position Part C and RG 1.155 Table 1 indicate NUMARC-8700 is acceptable for satisfying 10 CFR 50.63 requirements. NUMARC-8700, Appendix B, Section B.8 indicates system features to minimize potential for common-cause failures of the alternate AC (AAC) source. U.S. EPR SBODG and EDG system features that minimize the potential for common cause failure by following the guidance provided in NUMARC-8700, Appendix B, Section B.8 and the associated ITAAC that verify these features are as follows:

(1) "The AAC power system shall be equipped with a DC power source that is electrically independent from the blacked-out unit's preferred and Class 1E power system."

The EDG DC power source is from the Class 1E uninterruptible power supply system (EUPS), while the SBODG DC power source is from the 12-hour uninterruptible power supply system (12UPS). The EDG DC control power is verified from the respective division in U.S. EPR FSAR Tier 1, Section 2.5.4, Item 5.1. Verification of this item demonstrates independence between the EDG and SBODG related to DC power sources.

(2) "The AAC power system shall be equipped with an air start system, as applicable, that is independent of the preferred and the blacked-out unit's preferred and Class 1E power supply."

The SBODG air start system is independent from the EDG air start systems. Each diesel engine has a separate air start system associated with that engine. This independence is verified by U.S. EPR FSAR Tier 1, Section 2.5.3, Items 3.1 and 4.5.

(3) "The AAC power system shall be provided with a fuel oil supply, as applicable, that is separate from the fuel oil supply for the onsite emergency AC power system. A separate day tank supplied from a common storage tank is acceptable provided the fuel oil is sampled and analyzed consistent with applicable standards prior to transfer to the day tank."

The SBODG fuel oil system is independent from the EDG fuel oil system. Each SBODG has an independent fuel oil storage tank and day tank. The U.S. EPR FSAR was revised in response to RAI 116, Question 14.3.6-7 to verify this independence by adding Tier 1, Section 2.5.3 Items 3.2 and 3.3. Additionally, U.S. EPR FSAR Tier 1, Section 2.5.4, Items 3.9 and 3.10 verifies that each EDG has a fuel oil storage tank and fuel oil day tank.

(4) "If the AAC power source is an identical machine to the emergency onsite AC power source, active failures of the emergency AC power source shall be evaluated for applicability and corrective action taken to reduce subsequent failures."

The response to RAI 11, Question 08.04-3 describes the large difference in nominal size between the SBODG and the EDG, so the two types of diesel generators will be different models and not identical machines. The difference in size between the EDG and SBODG will also result in a different size exhaust bypass valve, further reducing probability of common-cause failure of this equipment.

(5) "No single point vulnerability shall exist whereby a likely weather-related event or single active failure could disable any portion of the onsite emergency AC power sources or the preferred power sources, and simultaneously fail the AAC power source(s)."

Minimal potential for common cause failure as a result of a likely weather-related event or single active failure is demonstrated by the physical separation of the SBODGs from the EDGs and the normal separation of the SBODGs from the emergency power supply system (EPSS). These attributes are verified by U.S. EPR FSAR Tier 1, Section 2.5.4, Item 2.2 and Section 2.5.3, Item 4.1.

(6) "The AAC power system shall be capable of operating during and after a station blackout without any support systems powered from the preferred power supply, or the blacked-out unit's Class 1E power sources affected by the event."

The SBODGs are capable of operating during and after an SBO without any support systems powered from the preferred power supply, or the EPSS or EUPS. The SBODG support systems are powered from the normal power supply system or the 12UPS. The portions of these distribution systems that support SBODG operation are reenergized by the SBODG during operation, so there is no reliance on the preferred power supply or the EPSS for SBODG operation.

(7) "The portions of the AAC power system subjected to maintenance activities shall be tested prior to returning the AAC power system to service."

This subsection is related to pos- maintenance testing and not related to design activities identified for ITAAC verification.

Information shown demonstrates SBODGs are independent and diverse from the EDGs. The ITAAC supports U.S. EPR FSAR Tier 2, Table 14.3-6, Item 6-19.

FSAR Impact:

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

Question 19-279:

(Follow-up to Question 19-247) The response to Question 19-247 estimates the amount by which core damage frequency (CDF) would decrease if the design change related to reactor coolant pump (RCP) thermal barrier cooling were incorporated in the U.S. EPR PRA. The sensitivity suggests that total CDF would decrease by 16 percent, largely driven by the fire and flood contribution, as a result of the design change. Therefore, while the change represents a decrease in overall risk, insights from the PRA may be affected. In response to Question 19-270, FSAR Section 19.1.2.4.1 was revised to state that, when a design change results in a cumulative impact of more than 10 percent on total CDF or large release frequency (LRF), PRA insights are assessed to see if they remain valid. If the insights are no longer valid, the PRA is updated.

a. Discuss the expected effect of the design change on the PRA insights (such as those documented in Table 19.1-108 of the FSAR), as well as the input to the reliability assurance program (RAP) and the important operator actions provided to the developers of procedures, training, and other human-factors programs.

b. State the planned schedule for incorporating this design change in the PRA.

c. Discuss how this schedule relates to the planned schedule for implementing other programs (e.g., RAP, procedure development) that may be affected by the change.

Response to Question 19-279:

Response to Question 19-279a:

The basic effect of the thermal barrier cooling design change on the PRA insights is the decrease in RCP seal loss of coolant accident (LOCA) contribution to internal events, fire and flooding CDF. For each initiating event group, the percentage contribution of RCP seal LOCA to the CDF can be measured by the Fussel Vesely (FV) importance of the parameter "PROB SEAL LOCA". The change in RCP seal LOCA contribution brought by this design change is shown in Table 19-279-1, by comparing the base case (based on the U.S. EPR PRA) and the sensitivity case performed in the response to Question 19-247.

Table 19-279-1

Seal LOCA Contribution to CDF

Event	Base Case	Sensitivity case performed for RAI 19-247 (CCWS CH cross-tie considered)
Internal Event CDF	11%	7%
Internal Fire CDF	43%	18%
Internal Flooding CDF	30%	2%
Total at-power CDF	24%	10%

One insight in the fire and flood PRA sections of the U.S. EPR FSAR Tier 2, Sections 19.1.5.2.2.8 and 19.1.5.3.2.8, which will change as result of this design change identifies seal LOCAs as dominant contributors to the fire and flood CDF. The most significant decrease is in the contribution of seal LOCAs to flooding CDF, from 30 percent to 2 percent. Flooding in SB1 or SB4 is no longer dominated by seal LOCAs, but instead by ventilation failures.

Seal LOCA contribution to fire CDF is also reduced, but it remains a significant contributor (18 percent). This could be explained by insights from the fire CDF cutsets shown in Group 4 of the U.S. EPR FSAR Tier 2, Table 19.1-66, which include a failure of CH2 and an independent failure of the battery-backed 480V motor control center 31BRA. This electrical failure prevents successful switchover of thermal barrier cooling to CH1.

The thermal barrier design change will affect the PRA insights as discussed previously. Importance measures used as input to the reliability assurance program (RAP) and other programs also will be affected.

The following material will be added to U.S. EPR FSAR Tier 2, Table 19.1-108:

24	<u>**Seal LOCA contribution to fire and flooding CDF**</u>: RCP seal LOCAs are identified as important contributors to both the internal fire and the internal flooding CDF. The thermal barrier cooling design change, identified in Section 19.1.2.4, exhibits a prospective reduction in seal LOCA contribution to fire and flooding CDF.	Tier 2, Section 19.1.5.2.2.8 and 19.1.5.3.2.8

Response to Question 19-279b:

PRA updates will be performed in accordance with the PRA maintenance and upgrade process described in U.S. EPR FSAR Tier 2, Section 19.1.2.4.

Response to Question 19-279c:

PRA provides input to the RAP as well as to the design certification environmental report. PRA input to the RAP and the environmental report will be revised in conjunction with PRA updates performed in accordance with the PRA maintenance and upgrade process described in U.S. EPR FSAR Tier 2, Section 19.1.2.4.

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 19.1-108 will be revised as described in the response and indicated on the enclosed markup.

Question 19-280:

(Follow-up to Question 19-248) The response to Question 19-248(d) justifies excluding the switchover to hot leg injection from the PRA, but only addresses boron precipitation. FSAR Section 6.3.2.8, as well as ANP-10299, "Applicability of AREVA NP Containment Response Evaluation Methodology to the U.S. EPR™ for Large Break LOCA Analysis," Revision 0, state that hot leg injection is used to terminate core steaming. Justify the exclusion of this operator action with respect to termination of steaming. In addition, if the operator action (whether modeled in the PRA or not) is important to maintaining a low CDF and/or LRF, this risk insight should be included in the FSAR.

Response to Question 19-280:

Hot leg injection is not credited in the PRA as a means to terminate steaming in a large LOCA. The analyses performed in FSAR Section 6 and in ANP-10299 used conservative assumptions, as required by the Standard Review Plan. Calculations performed with both GOTHIC and MAAP4, without the Standard Review Plan conservatisms for design basis analysis predicted containment pressures below the design pressure limits. Based on these calculations, the PRA estimates that, with or without hot leg injection, containment overpressurization is not likely to occur following a large LOCA.

Incorporation of hot leg injection at 90 minutes in the design basis analysis accelerated steam suppression such that the second containment pressure peak was mitigated by the safety injection switchover to the hot leg at 90 minutes.

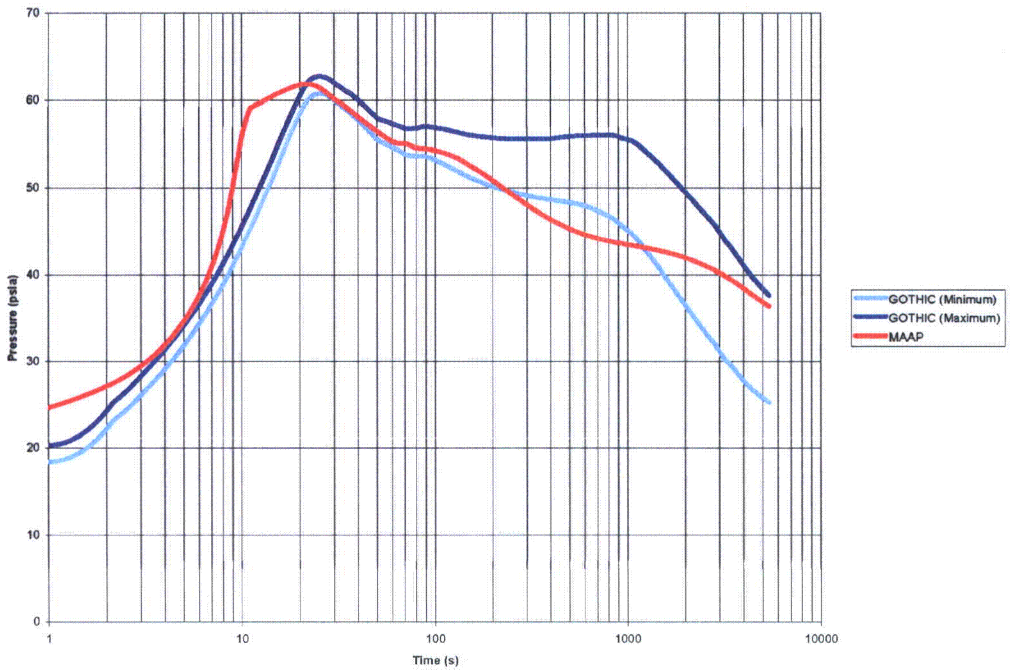
When the conservative assumptions are removed, containment pressure response with cold leg injection alone did not result in the long term containment pressurization predicted in the design basis analysis, as discussed below.

Similar S-RELAP5/GOTHIC and MAAP4 calculations were prepared with many best-estimate assumptions. One difference was that that S-RELAP5 calculation was prepared assuming three operational ECCS trains injecting into the cold legs (one going to the broken loop). MAAP4 is typically not appropriate for LLOCA short term peak clad temperature calculations; however, for long-term, decay heat driven analysis, MAAP4 is acceptable. The MAAP4 calculation modeled a large cold break LOCA with one LHSI train injecting into the intact cold leg (a lumped component in MAAP4). The subsequent containment pressure response from both analyses is shown in Figure 19-280-1. The figure shows comparable pressure responses between the two calculations. The maximum pressurization occurs during the early pressure peak, and is about 63 psia (49 psig), below the design pressure. Beyond the reflood phase (within a few hundred seconds), the mass and energy releases are dominated by decay heat. Decay heat was modeled consistently in these calculations. The difference in ECCS flows affects the amount of steaming and sensible heat addition and explains why the long term containment response predicted by MAAP is at the higher end of the range of responses predicted by GOTHIC.

It should also be noted that even if the containment design pressure of 62 psig was to be exceeded, this is not likely to result in a failure of the containment. The containment fragility evaluation performed for the Level 2 PRA showed that the probability of containment failure at 62 psig is approximately 1E-6. To attain a probability of failure of one percent, the pressure would need to rise up to approximately 105 psig.

Figure 19-280-1

Containment Pressurization for Representative LLOCA run



FSAR Impact:

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

Question 19-281:

(Follow-up to Question 19-251) The response to Question 19-251 includes information necessary for the staff to conclude that shutdown-specific fire and flood initiating events do not pose a significant risk. As requested in Question 19-251, add a summary of this information to the FSAR.

Response to Question 19-281:

The U.S. EPR FSAR will be updated to include a summary of the information contained in the response to Question 19-251.

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 19-282:

(Follow-up to Question 19-251) The remote shutdown station (RSS) is only required to be operable during MODES 1, 2, and 3 per technical specification (TS) 3.3.3; however, the at-power PRA considers transfer to the RSS following a control room fire. State whether the assessment of internal fires during shutdown assumes that the RSS is available during shutdown. As appropriate, revise FSAR Table 19.1-109 to include this assumption. If applicable, discuss the administrative controls that will ensure the RSS remains available during shutdown.

Response to Question 19-282:

The PRA did not model internal fires in shutdown as separate initiating events. Instead, it is assumed that the at-power fire PRA envelops shutdown conditions. In addition, the response to RAI Question 19-251 considers some potential shutdown-specific internal fire scenarios requested by the NRC. The PRA models a fire in the control room as one of the fire scenarios at power. In that case, the PRA credits operator action to transfer the plant control to the remote shutdown station (RSS). Since the assumption is made that the at-power fire analysis envelops shutdown, it implies that the PRA also credits this transfer in shutdown.

U.S. EPR FSAR Tier 2, Table 19.1-109 will be revised to include this assumption. A new sentence will be added to Assumption 74 in Table 19.1-109, as follows:

No.	Category	PRA General Assumptions
74	Fire	The RSS is assumed to be available in all POS where fuel is loaded in the core.

COL information item 19.1-9, from U.S. EPR FSAR Tier 2, Table 1.8-2, provides that assumptions used in the PRA—including PRA inputs to the reliability assurance program (RAP)—remain valid for the as-built plant.

As indicated in the response to RAI Question 17.04-7, the RSS is included in the RAP. As described in U.S. EPR FSAR Tier 2, Section 17.4.1, inclusion in the RAP provides assurance that the RSS availability will be integrated into relevant existing programs, such as maintenance rule. As indicated in the COL information items in U.S. EPR FSAR Tier 2, Table 1.8-2 that are associated with U.S. EPR FSAR Tier 2, Chapter 17, the COL applicant will provide information on the description, scope, and implementation of these programs.

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.

Question 19-283:

(Follow-up to Question 19-254) In response to Question 19-254, FSAR Table 19.1-109 was revised to include part of the assumption about nozzle dams identified in the response to Question 19-174. However, the revision does not address other assumptions identified in Question 19-174, specifically:

a. Plant procedures that cover reduced inventory operation govern the installation of nozzle dams and the establishment of adequate venting to prevent pressurization of the RPV upper plenum due to a postulated loss of decay heat removal.

b. Freeze seals are not expected to be used; they will not be part of the maintenance procedures for the U.S. EPR.

c. There are no other temporary reactor coolant system pressure boundaries as defined by NUREG-1449 and NUREG-1512.

Discuss why these assumptions were excluded from the revision to Table 19.1-109, and revise the FSAR as needed.

Response to Question 19-283:

The U.S. EPR FSAR Tier 2, Table 19.1-109, will be revised as follows:

Additional material will be added to No. 82 (LPSD).

> "Plant procedures that cover reduced inventory operation will govern the installation of nozzle dams and the establishment of adequate venting to prevent pressurization of the RPV upper plenum due to postulated loss of decay heat removal.
>
> Nozzle dams are the only U.S. EPR related temporary reactor coolant system boundary as specified by NUREG-1449 and NUREG-1512. Freeze seals are not expected to be used; they will not be part of the maintenance procedures for the U.S. EPR."

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.

U.S. EPR Final Safety Analysis Report Markups

The PRA model credits the alternate feed from Division 1 to Division 2 and from Division 4 to Division 3 in SBO conditions and in non-SBO conditions. This change does not modify the availability of those functions or the context in which they will be performed, but modifies the way that they will be executed.

The failure probability of those functions, as modeled in the PRA, is dominated by human errors. Those human errors were assigned HEP values judged to be conservative for this alternate feed configuration. Therefore, this design change is judged not to have a significant impact on the current conclusions of the PRA.

6. Component cooling water (CCW) common header cooling to reactor coolant pump (RCP) thermal barriers – This design change consists of having one CCW common header cooling all four RCP thermal barriers, instead of each common header cooling two RCP thermal barriers. In case of a loss of cooling from one header, a manual switchover to the second header can be performed. This change has been quantitatively evaluated and results in a small decrease in seal LOCA contribution to internal event CDF. A larger decrease in internal fire and flood event CDF can be attributed to the conservative treatment of these events, which is likely to change as a result of more realistic fire and flood PRA updates. Overall, this design change is judged not to have a significant impact on the current conclusions of the PRA.

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7. EFWS supply header isolation valves – This design change consists of maintaining the EFWS supply header isolation valves closed. If one or more EFW train is unavailable, a manual action is required to interconnect the four tanks so that the entire EFW inventory is available. In case of a pipe break or a tank leakage in the EFWS (internal flooding), the operators no longer have to isolate the leaking train to avoid losing all EFW inventory. One tank inventory still may be lost; therefore, it is necessary to refill one of the intact tanks in order to achieve the 24-hour mission time.

   This change results in a measurable increase in internal event and internal flooding CDF, driven by operator failure to perform the interconnection. PRA insights and assumptions regarding manual isolation of an EFWS pressure boundary failure are also affected, as this isolation is no longer needed. This effect is recognized in Table 19.1-108, Item 10, and Table 19.1-109, Item 66.

19.1.2.4.1 Description of PRA Maintenance and Update Program

The U.S. EPR PRA model and supporting documentation are maintained so that they continue to reflect the as-designed characteristics of the plant. Consistent with the ASME PRA Standard, Reference 5, and RG 1.200, a process is in place to perform the following as applicable to the certified design:

- Monitor PRA inputs and collect any new information relevant to the PRA.
- Maintain and upgrade the PRA to be consistent with the design.
- Consider cumulative impacts of pending changes when applying the PRA.

The probability of plugging the IRWST suction strainers is modeled the same as at-power operation (i.e., CCF). Maintenance work during shutdown could result in a higher probability of plugging. However, the IRWST design is somewhat unique in comparison to the PWR plants operating in the USA. The structure is very large with separation between suction lines to the four SB; three levels of filters are also provided: trash racks, retaining baskets, and six strainers with a back flush capability. This probability of plugging is also dependent on maintenance procedures that will be in place to control foreign material, but are not available in this phase. As a result, the present modeling of the IRWST suction strainers was not changed.

Preventive maintenance modeling was revised for LPSD because of obvious differences in risk management strategies from power operation. Assumptions on maintenance strategies are as follows:

- Maintenance on the SG systems is assumed to be performed on two SGs that are not available in states CAD and CBD.
- Maintenance on the other trains is assumed to occur in state E. One division is assumed to be out for maintenance during that state.

Available mitigating systems in different POSs are defined in Table 19.1-89—System Availability During Shutdown.

19.1.6.1.8 Fire & Flooding Events in Shutdown

Limited evaluation of fire and flooding initiators is performed in the LPSD PRA. Fire and flooding events are evaluated with bounding analyses similar to the analysis performed at-power. Since there is physical separation between RHR trains, and at least two are operating during shutdown, fires and floods can only impact one operating train. Because of the physical separation between operating and standby trains, the impact of the possible degradation in the fire and flood barriers during shutdown is assumed to be not significant. Transient combustibles and maintenance activities may result in a higher fire/flood frequency during shutdown in certain parts of the plant, but are judged to be not significant for the protected RHR trains providing decay heat removal.

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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.

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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.

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Based on ~~these judgments~~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 Initiating Events Contributions - Level 1 Shutdown. Only those initiating events that contribute more than one percent to the total internal events CDF are listed in the table. All initiating events and their contributions are illustrated in Figure 19.1-21—U.S. EPR Initiating Event Contributions - Level 1 Shutdown. As can be seen from Table 19.1-90 and Figure 19.1-21, the shutdown initiating events which dominate shutdown core damage frequency are uncontrolled level drop in states CBD and DU, LOCA in state CBD, and loss of RHR in state CBD. Note that the LOOP event is included in the loss of RHR initiating event. Based on the FV importance measures from the shutdown model, the LOOP events during shutdown contribute approximately 37 percent of the total risk.

The total contribution of each POS is illustrated in Table 19.1-91—U.S. EPR Shutdown State (POS) Contributions - Level 1 Shutdown. This table shows the estimated POS duration, the CDF and CDF/day for each POS. The highest contribution is from POS CBD and DU which is to be expected because these are states where RCS is being drained to mid-loop and an uncontrolled level drop could occur.

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Table 19.1-108—U.S. EPR PRA Based Insights Sheet 3 of 5

No	U.S. EPR PRA Based Insight	Disposition
10	**Isolation of EFW tank leaks or pipe breaks is assumed possible for any break location.** Pipe breaks in the EFWS are treated as flooding events with the potential to drain all four EFW tanks. It is assumed that the operators would have the ability to manually isolate an EFW pipe break occurring in any of the four SB with isolation valves in another unaffected SB and to initiate DWS makeup to the tanks of the intact EFW trains. The severity of a flooding event from an EFW tank leak will be reduced as a result of the design change identified in Section 19.1.2.4, which maintains the EFW suction header isolation valves closed. Manual isolation of an EFW tank leak or pipe break will not be necessary.	Tier 2, Section 3.4.3.4; Tier 2, Section 10.4.9.2.1
11	**Flooding event would not affect the electrical and I&C rooms of a safeguard building.** Flood paths are provided in the safeguard buildings, such that water from a break anywhere in the building would be stored in the lower elevation of the building. In particular, a flooding event would not affect the electrical and I&C rooms of a safeguard building. All electrical / I&C equipment is located above the maximum postulated flood level.	Tier 1, Section 2.1.1; Tier 2, Section 3.4.3.4
12	**Cable separation in the MCR Cable Spreading Area** Due to divisional separation measures in the MCR Cable Spreading Area, a fire in the cable spreading area is assumed to disable only one electrical safety division. Non-safety division cables are also assumed to be separated from the safety divisions.	Tier 2, Section 9.5.1.2.1
13	**Shutdown management guidelines** The shutdown guidelines as described in the Shutdown Management Guidelines, NUMARC 91-06, should be considered when developing the plant specific operations procedures.	Tier 2, Section 13.5.2; COLA Item 13.1-1; COLA Item 13.4-1; COLA Item 13.5-1
14	**The low probability that the IRWST suction strainers are plugged during shutdown.** The IRWST design (e.g., large, separation between suction lines, debris retaining capability) and plant procedures (e.g., foreign material control) are expected to ensure that this probability is low.	Tier 2, Section 6.3.2.2.2; COLA Item 6.3-1
15	**Closing containment hatches and penetrations** The ability to close containment hatches and penetrations during Modes 5 & 6 prior to steaming to containment is important. It is assumed that procedures and training will be developed that encompass this item.	Tier 2, Section 13.5.2; COLA Item 13.1-1; COLA Item 13.4-1; COLA Item 13.5-1
16	**Low pressure reducing station auto isolation** In shutdown operation, low pressure reducing station auto isolation on low loop level is important to prevent possible RCS flow diversion through CVCS.	Tier 2, Section 9.3.4.2.2

Table 19.1-108—U.S. EPR PRA Based Insights
Sheet 5 of 5

No	U.S. EPR PRA Based Insight	Disposition
24	**Seal Loca contribution to fire and flooding CDF** RCP seal LOCAs are identified as important contributors to both the internal fire and the internal flooding CDF. The design change to thermal barrier cooling, identified in Section 19.1.2.4, exhibits a prospective reduction in seal LOCA contribution to fire and flooding CDF.	Tier 2, Section 19.1.5.2.2.8 Tier 2, Section 19.1.5.3.2.8

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Table 19.1-109—U.S. EPR PRA General Assumptions
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No.	Category[1]	PRA General Assumptions[2]
64	Flood	Floods caused by a break in a system with very large flooding potential (ESWS or DWS) are assumed to be contained below ground level of the affected buildings (SB or FB). This assumption is based on the ability to automatically isolate those systems upon high sump level. Moreover, the amount of time needed to flood a building up to ground level is lengthy which supports detection and isolation by the operator if automatic isolation failed. This manual isolation is credited because an alarm exists in the Control Room, and the operation can be performed with high reliability.
65	Flood	A flood in an SB is assumed to affect the CCW switchover valves. This is a conservative assumption, since those valves are located exactly at ground level, while all flooding events considered are contained below ground level. Failure of either Train 1 or 4 of CCW requires a switchover to be performed in order to ensure continuous supply to the CCW common header. This assumption results in asymmetrical results for SAB1/4 versus SAB2/3.
66 19-274	Flood	Pipe breaks in the EFWS are treated as flooding events with the potential to drain all four EFW tanks. It is assumed that the operators would have the ability to manually isolate an EFW pipe break occurring in any of the four SB with isolation valves located in the unaffected SBs, and to initiate DWS ~~makeup to the tanks of the intact EFW trains.~~ ~~The severity of a flooding event~~ from an EFW tank leak or pipe break will be reduced as a result of the design change identified in Section 19.1.2.4, which maintains the EFW suction header isolation valves closed. Manual isolation of an EFW tank leak or pipe break will not be necessary.
67	Flood	If a flood in the annulus from a fire water distribution system (FWDS) pipe break is left unisolated, water level will reach the level of the doors before it reaches the level of the electrical penetration. The doors are not designed to withstand water pressure applied from the Annulus side, therefore, their opening is considered. If both doors fail to open, water will reach the electrical penetrations level. All instrumentation to the core is affected, leading to a possible loss of control and/or spurious signals. It is difficult to assess the consequences, but they could be severe and, conservatively, core damage was assumed. The probability that the connection boxes of the electrical penetrations that run through the annulus will fail if submerged is estimated to be 0.5. This number represents the limited state of knowledge regarding the design of those penetrations. This assumption has a very high importance, because the failure of the penetrations is assumed to lead directly to core damage.

Table 19.1-109—U.S. EPR PRA General Assumptions Sheet 14 of 16

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No.	Category[1]	PRA General Assumptions[2]
73	Fire	The U.S. EPR RCPs will be fitted with an oil-collection system designed to prevent RCP oil leakage from reaching any ignition source. Because of this improved design, it is assumed that fire ignition due to RCP oil leakage reaching an ignition source does not occur.
74	Fire	A fire in the MCR is assumed to disable the entirety of the MCR if it is not suppressed. This will happen if a fire affects either the functional capability of the MCR (destroying cables or workstations) or if it degrades the habitability to an extent where operators have to evacuate the control room. A corresponding operator action is associated with the entire process, including the decision to evacuate the MCR and the action of switching controls. It is assumed that once the operators resume control of the plant from the RSS, the status of the plant will be similar as that following a Loss of Balance of Plants (LBOP) since the fire in the MCR could result in a loss of control of secondary side balance of plant systems. Failure of the operators to transfer to the RSS is assumed to lead directly to core damage. The RSS is assumed to be available in all POS where fuel is loaded to the core.
75	Fire	For the CSR and MCR, the generic room fire ignition frequency is modified by using the 0.5 correction factor to account for the fact that most of the cables routed through the CSR and MCR will be fiber optic cables that are not susceptible to ignition under any condition.
76	Fire	The consequences of the spurious opening of an MSRIV are dependent on the position of the MSIV, with higher consequences corresponding to an open MSIV. The MSIVs are designed to fail closed in the case that their associated SOVs are de-energized. However, hot shorts may still cause one or more MSIVs to remain open. It is conservatively assumed that if there is a fire in the valve room that causes a spurious opening of an MSRIV, it could affect MSIV on the same location, even though there is approximately 14 feet of spatial separation between the MSRIV and MSIV. Based on engineering judgment, it is assumed that a fire affecting an MSRIV would cause its associated MSIV to fail open with a probability of 0.5 and independently cause the other MSIV in the same Valve Room to fail open with a probability of 0.1. Since this modeling was finalized, fire barriers were added in each of the two main steam/main feedwater valve rooms to separate Division 1 from Division 2 and Division 3 from Division 4. This separation would prevent any fire impact on the second MSIV.
77	Fire	Detailed designs for the Turbine Building and the Switchgear Building were not available at the time of the fire risk evaluation. Therefore, it was conservatively assumed that both the TB and SWGR building are one contiguous fire area. Given that the type of communications that will exist between the Switchgear Building and the TB is not known, it was consider reasonable to assume that electrical penetrations and doors, if any, will have a fire rating of three hours.

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Table 19.1-109—U.S. EPR PRA General Assumptions
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No.	Category[1]	PRA General Assumptions[2]
82	LPSD	Nozzle dams are not required during a plant shutdown, but may be used infrequently during mid-cycle maintenance, when full core off-load is not desirable. Appropriate RCS operating conditions will be considered in the specification of nozzle dams to provide reasonable assurance that nozzle dams will not fail. Plant procedures that cover reduced inventory operation will govern the installation of nozzle dams and the establishment of adequate venting to prevent pressurization of the RPV upper plenum due to a postulated loss of decay heat removal. Nozzle dams are the only U.S. EPR related temporary reactor coolant system boundary as specified by NUREG-1449 and NUREG-1512. Freeze seals are not expected to be used; they will not be part of the maintenance procedures for the U.S. EPR.
83	LPSD	The efficiency of the Passive Autocatalytic Recombiners (PAR) during shutdown is assumed to be nominal. Maintenance unavailability, if any, is assumed to be limited to a small fraction of the PARs and would not affect the overall efficiency of the system.

Notes:

1. | Category | Description |
   |---|---|
   | Model | Modeling Assumption |
   | IE | Initiating Event |
   | CC | Common Cause |
   | PM | Preventive Maintenance |
   | HRA | Human Reliability Analysis |
   | SYS | System Modeling |
   | I&C | Instrumentation and Controls |
   | LPSD | Low Power/ Shutdown Modeling |
   | Flood | Flood Analysis |
   | Fire | Fire Analysis |
   | Seismic | Seismic Analysis |

2. The PRA assumptions will be reevaluated as part of the PRA maintenance and update process. The PRA maintenance and upgrade process is described in U.S. EPR FSAR Tier 2, Section 19.1.2.4. COL item 19.1-9 listed in U.S. EPR FSAR Tier 2, Table 1.8-2—U.S. EPR Combined License Information Items is provided to confirm that assumptions used in the PRA remain valid for the as-to-be-operated plant.

Next File