ArevaEPRDCPEm Resource

From:	WELLS Russell D (AREVA NP INC) [Russell.Wells@areva.com]
Sent:	Thursday, July 02, 2009 4:49 PM
To:	Tesfaye, Getachew; Miernicki, Michael
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. 133, FSAR Ch 19, Supplement 6
Attachments:	RAI 133 Supplement 6 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 7 of the 17 questions of RAI No. 133 on December 8, 2008. AREVA NP submitted Supplement 1 to the response on January 30, 2009 to address 3 of the remaining questions. AREVA NP submitted Supplement 2 to the response on February 11, 2009 to address 4 of the remaining questions. AREVA NP submitted Supplement 3 to the response on March 6, 2009 to address 1 of the remaining questions. AREVA NP submitted Supplement 4 to the response on March 27, 2009 to provide a revised date regarding Revision 1 to ANP-10290, "Environmental Report Standard Design Certification." AREVA NP submitted Supplement 5 to the response on June 19, 2009 to address one of the remaining questions.

The attached file, “RAI 133 Supplement 6 Response US EPR DC.pdf” provides a technically correct and complete response to the one remaining question.

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 133 Question 19-230.

The following table indicates the respective pages in the response document, “RAI 133 Supplement 6 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 133 — 19-230	2	27

This concludes the formal AREVA NP response to RAI 133, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

(Russ Wells on behalf of)
Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
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From: WELLS Russell D (AREVA NP INC)
Sent: Friday, June 19, 2009 2:49 PM
To: 'Getachew Tesfaye'

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. 133, FSAR Ch 19, Supplement 5

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 7 of the 17 questions of RAI No. 133 on December 8, 2008. AREVA NP submitted Supplement 1 to the response on January 30, 2009 to address 3 of the remaining questions. AREVA NP submitted Supplement 2 to the response on February 11, 2009 to address 4 of the remaining questions. AREVA NP submitted Supplement 3 to the response on March 6, 2009 to address 1 of the remaining questions. AREVA NP submitted Supplement 4 to the response on March 27, 2009 to provide a revised date regarding Revision 1 to ANP-10290, "Environmental Report Standard Design Certification."

The attached file, “RAI 133 Supplement 5 Response US EPR DC.pdf” provides a technically correct and complete response to one of the remaining 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 133 Question 19-243.

The following table indicates the respective pages in the response document, “RAI 133 Supplement 5 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 133 — 19-243	2	4
RAI 133 — 19-243 Appendix A	A-1	A-49

The schedule for a technically correct and complete response to the one remaining question is revised from June 19, 2009 to July 2, 2009, as indicated in the table provided below:

Question #	Response Date
RAI 133 — 19-230	July 2, 2009

Sincerely,

(Russ Wells on behalf of)
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From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, March 27, 2009 3:06 PM
To: 'Getachew Tesfaye'
Cc: MCINTYRE Brian (AREVA NP INC); SLOAN Sandra M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); NOXON David B (AREVA NP INC); SZYMCZAK William J (AREVA NP INC); SANDERS Harris I (AREVA NP INC); SANDERS Mitchell K. (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 4

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 7 of the 17 questions of RAI No. 133 on December 8, 2008. AREVA NP submitted Supplement 1 to the response on January 30, 2009 to address 3 of the remaining questions. AREVA NP submitted Supplement 2 to the response on February 11, 2009 to address 4 of the remaining questions. AREVA NP submitted Supplement 3 to the response on March 6, 2009 to address 1 of the remaining questions.

IN AREVA NP's responses to RAI 133, Questions 19-236, 19-237 and 19-238, AREVA NP stated that a revision to ANP-10290, "Environmental Report Standard Design Certification" would be provided by March 30, 2009. On March 3, 2009 AREVA NP provided a draft revision of ANP-10290 for NRC staff review and comment.

Today, PRA staff indicated that NRC's plan is to provide comments to AREVA NP by April 10, 2009. Therefore, AREVA NP will not be providing Revision 1 by March 30, 2009. Rather, AREVA NP will provide a revised commitment date upon review of the NRC staff's comments.

Thank you,

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From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, March 06, 2009 11:56 AM
To: 'Getachew Tesfaye'
Cc: NOXON David B (AREVA NP INC); DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 3

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 7 of the 17 questions of RAI No. 133 on December 8, 2008. AREVA NP submitted Supplement 1 to the response on January 30, 2009 to address 3 of the remaining questions. AREVA NP submitted Supplement 2 to the response on February 11, 2009 to address 4 of the remaining questions.

Attached please find Supplement 3 to AREVA NP's response to RAI No. 133. The attached file, "RAI 133 Supplement 3 Response US EPR DC.pdf" provides technically correct and complete responses to one of the remaining questions.

The following table provides the pages in the response document, "RAI 133 Supplement 3 Response US EPR DC.pdf" containing the response to each question.

Question #	Start Page	End Page
RAI 133 — 19-244	2	52

A complete answer is not provided for 2 of the 17 questions. The schedule for a technically correct and complete response to the remaining 2 questions is unchanged and provided below.

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-243	June 19, 2009

Sincerely,

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From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, February 11, 2009 4:53 PM
To: 'Getachew Tesfaye'
Cc: NOXON David B (AREVA NP INC); DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 2

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 7 of the 17 questions of RAI No. 133 on December 8, 2008. AREVA NP submitted Supplement 1 to the response on January 30, 2009 to address 3 of the 17 questions.

The attached file, “RAI 133 Supplement 2 Response US EPR DC.pdf” provides technically correct and complete responses to 4 of the remaining 7 questions.

The following table indicates the respective pages in the response document, “RAI 133 Supplement 2 Response US EPR DC.pdf” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 133 — 19-232	2	6
RAI 133 — 19-233	7	15
RAI 133 — 19-238	16	17
RAI 133 — 19-240	18	23

The schedule for technically correct and complete responses to the remaining 3 questions is unchanged and provided below.

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-243	June 19, 2009
RAI 133 — 19-244	March 6, 2009

Sincerely,

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From: WELLS Russell D (AREVA NP INC)
Sent: Friday, January 30, 2009 4:21 PM
To: 'Getachew Tesfaye'
Cc: Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); SLIVA Dana (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 1

Getachew,

The proprietary and non-proprietary versions of the response to RAI No. 133, Supplement 1 are submitted via AREVA NP Inc. letter, "Response to U.S. EPR Design Certification Application RAI No. 133, Supplement 1 " NRC 09:008, dated January 30, 2009. The enclosure to that letter provides technically correct and complete responses to 3 of the remaining 10 questions in RAI No. 133. An affidavit to support withholding of information from public disclosure, per 10CFR2.390(b), is provided as an enclosure to that letter.

The schedule for technically correct and complete responses to the remaining questions in RAI No. 133 is provided below:

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-232 (c)	February 13, 2009
RAI 133 — 19-233	February 13, 2009
RAI 133 — 19-238	February 13, 2009
RAI 133 — 19-240 (2)	February 13, 2009
RAI 133 — 19-243	June 19, 2009
RAI 133 — 19-244	March 6, 2009

Sincerely,

(Russ Wells on behalf of)

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From: WELLS Russell D (AREVA NP INC)
Sent: Monday, December 08, 2008 6:43 PM
To: 'Getachew Tesfaye'
Cc: 'John Rycyna'; 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. 133, FSAR Ch 19

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 133 Response US EPR DC.pdf" provides technically correct and complete responses to 7 of the 17 questions.

The following table indicates the respective pages in the response document, "RAI 133 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 133 — 19-230	2	2
RAI 133 — 19-231	3	3
RAI 133 — 19-232	4	4
RAI 133 — 19-233	5	5
RAI 133 — 19-234	6	6
RAI 133 — 19-235	7	7
RAI 133 — 19-236	8	8
RAI 133 — 19-237	9	9
RAI 133 — 19-238	10	10
RAI 133 — 19-239	11	12
RAI 133 — 19-240	13	45
RAI 133 — 19-241	46	47
RAI 133 — 19-242	48	48
RAI 133 — 19-243	49	49
RAI 133 — 19-244	50	50
RAI 133 — 19-245	51	52
RAI 133 — 19-246	53	79

A complete answer is not provided for 10 of the 17 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-232 (c)	February 13, 2009
RAI 133 — 19-233	February 13, 2009
RAI 133 — 19-236	January 30, 2009

RAI 133 — 19-237	January 30, 2009
RAI 133 — 19-238	February 13, 2009
RAI 133 — 19-240 (2)	February 13, 2009
RAI 133 — 19-242	January 30, 2009
RAI 133 — 19-243	June 19, 2009
RAI 133 — 19-244	March 6, 2009

(Russ Wells on behalf of)

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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Thursday, November 06, 2008 8:42 PM
To: ZZ-DL-A-USEPR-DL
Cc: Edward Fuller; Theresa Clark; Hanh Phan; Hossein Hamzehee; Lynn Mrowca; John Rycyna; Joseph Colaccino
Subject: U.S. EPR Design Certification Application RAI No. 133 (1456), FSARCh. 19

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 29, 2008, and discussed with your staff on November 6, 2008. Draft RAI Questions 19-229 was deleted and Draft RAI Questions 19-238 was 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,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 618

Mail Envelope Properties (1F1CC1BBDC66B842A46CAC03D6B1CD4101A934C6)

Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 6
Sent Date: 7/2/2009 4:48:45 PM
Received Date: 7/2/2009 4:48:50 PM
From: WELLS Russell D (AREVA NP INC)

Created By: Russell.Wells@areva.com

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Files	Size	Date & Time
MESSAGE	14588	7/2/2009 4:48:50 PM
RAI 133 Supplement 6 Response US EPR DC.pdf		633019

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

Response to

Request for Additional Information No. 133. Supplement 6

11/07/2008

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

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

Question 19-230:

(Follow-up to Question 19-147) In response to Question 19-147, the applicant provided a subjective probability distribution for structural capacity of the EPR reactor pit for dynamic pressure loads. Table 19-147-1 shows that the reactor pit is expected to fail with certainty for loads exceeding 20 kPa-s. NRC assessment of ex-vessel steam explosion loads (NUREG/CR-6849) under similar conditions show maximum loads resulting from FCI energetics ranging from about as low as 10 to as high as a several hundred kPa-s, depending on the melt pour and analysis assumptions and conditions.

a. The approach described in the response appears to be subjective. One acceptable approach to this problem is to determine the threshold impulse load at which the pit structure will have zero probability of failure (i.e., this approach is typically considered as bounding. For instance, in other recent submittals, the DYNA 3D model was used to establish the impulse threshold for the portion of containment that was subject to steam explosion-induced impulse loads). Please perform a mechanistic analysis that supports the assigned uncertainty distribution.

b. Please discuss the implication of the NUREG/CR-6849 results for U.S. EPR in light of the assumed reactor pit capacity.

c. Please provide the technical justification for arriving at ex-vessel FCI loads that are much lower than has been estimated for other plants under similar conditions (e.g. AP1000). This should include plant-specific analysis using methods that are similar to those that are being used in other contemporary studies (e.g. see Westinghouse AP1000 DCD, GEH ESBWR DCD).

d. Please provide the range of expected loads on the RPV, and if there is any potential for RPV uplift impacting containment penetrations.

e. Please provide an analysis of the impact of the reactor pit failure on severe accident progression for U.S. EPR.

Response to Question 19-230:

The evaluation of the consequences of ex-vessel steam explosion on the U.S. EPR reactor pit, as presented in U.S. EPR FSAR Tier 2, Section 19.1.4.2.1.2 and in the Response to RAI 22, Question 19-147, was a representation of the state of knowledge on the pit structural capacity based on limited U.S. EPR specific information. Given the limitations of this initial analysis, plant-specific analysis of the ex-vessel steam explosion loads and the structural capacity of the U.S. EPR reactor pit have been performed and the probabilistic risk assessment (PRA) evaluation of the ex-vessel steam explosion phenomenon was revised accordingly.

1. The best-estimate analysis of the steam explosion phenomenon includes:

   – The results of the ex-vessel steam explosion analysis performed by the University of Stuttgart Institute for Nuclear Technology and Energy Systems (IKE) for the generic EPR design (directly applicable to the U.S. EPR design).

   – A discussion of the differences with the predicted loads presented in NUREG/CR-6849 (Reference 1).

   – A civil/structural evaluation of the capacity of the U.S. EPR reactor pit.

   – A re-evaluation of the U.S. EPR pit failure probability factoring in the new analysis.

2. An uncertainty analysis is provided to evaluate the impact on the reactor pit of loads comparable to those predicted in NUREG/CR-6849. The analysis considers the uncertainty on the boundary conditions used in the calculation of the steam explosion loads.
3. To address items (d) and (e) of this question, the reactor pressure vessel (RPV) uplift and the accident progression are discussed. The impact of the new steam explosion analysis on the Level 2 PRA results and insights are also provided.

The results are presented in the order in which the analysis was performed; therefore, responses to questions (a) through (e) are reorganized to follow the outline detailed above.

Response to Question 19-230(c):

The evaluation of steam explosion loads in the U.S. EPR reactor pit is revised. The results described in this analysis are the findings of the steam explosion study performed by IKE for the generic EPR design.

First, the steam explosion scenarios are addressed, followed by a brief description of the computer codes used. Then the boundary conditions are described followed by the steam explosion loads results.

Ex-vessel Steam Explosion Scenarios

The ex-vessel steam explosion scenario involving molten corium in contact with a stable water pool in the reactor pit has been explicitly addressed in the U.S. EPR design. The U.S. EPR design addresses this scenario by the elimination of water sources and pathways that would allow water to pool in the reactor pit. Only a limited number of low probability reactor coolant system (RCS) failure modes could lead to a wet pit.

An evaluation of the relevant RCS failure modes potentially leading to steam explosion conditions concluded that, for a stable water pool to be present in the reactor pit at the time of RPV failure, the only credible scenario is induced hot leg rupture while the RCS is at high pressure (i.e., manual actuation of the U.S. EPR primary depressurization system is assumed to have failed). The subsequent depressurization would allow accumulator inventory to enter the RCS. Some fluid would be swept through the core and escape to the reactor cavity; however, most of it would be retained in the RCS and RPV. With the induced hot leg rupture, RPV failure would occur at low pressure. The total frequency of relevant sequences with hot leg rupture and vessel failure is approximately 8E-08/yr, or about 15% of the at-power core damage frequency (CDF).

The Level 2 PRA evaluation of phenomenology at vessel failure concluded that a lateral leak is the most likely failure mode for the RPV. This is due to the focusing effect at the junction of the oxidic and metallic layers of the corium pool, leading to high heat densities in proximity of the RPV wall. Illustrations of lateral and central RPV failures are shown in Figures 19-230-1 and 19-230-2, respectively. The Level 2 PRA support study concluded that:

- The lateral failure mode represents 94 percent of the RPV failure modes.
- The central failure scenario represents 5 percent of the RPV failure modes.

The remaining 1 percent represents complete circumferential failure modes which have no impact on steam explosion scenarios and are beyond the scope of this RAI response.

Ex-vessel Steam Explosion Models and Computer Codes

For each of the two RPV failure scenarios, a deterministic analysis of the ex-vessel steam explosion was performed. The analysis was conducted in two steps. The first step is the modeling of the corium jet fragmentation and premixing in the water pool. This analysis was investigated using the IKEMIX/IKEJET codes (References 2 and 3). The second step, the explosion phase, was modeled with the IDEMO code (References 2 and 3). The codes used have been validated against experiments such as FARO (Reference 4) and KROTOS

(Reference 5) and benchmarked against other fuel coolant interactions (FCI) computer codes in the framework of the OECD-SERENA project (References 2 and 3). More details on the modeling codes are provided in the Response to Question 19-230(b).

Ex-vessel Steam Explosion Boundary Conditions

The ranges of boundary conditions selected in the different scenarios evaluated by IKE represent realistic values observed in previous experimental studies on the RPV failure modes, such as the FOREVER experiment (Reference 6). The ranges considered for the two scenarios (i.e., lateral and central melt leaks) are summarized in Table 19-230-1.

The rationale for selecting the parameter ranges used in the deterministic analysis is as follows:

- Jet Diameter: The jet diameter is taken to be equal to the breach diameter in both central and lateral scenarios. The jet diameter impacts the melt flow rate out of the RPV, which is a key parameter in the jet breakup in the premixing phase. A sensitivity analysis on the jet diameter, performed in the IKE analysis, concluded that for the jet break up phenomena relevant for ex-vessel steam explosion effects, breach diameter sizes of 10 to 40 cm bound the realistic scenarios.
- Melt Composition: It has been shown in several studies including NUREG/CR-6849 that immediately after core relocation to the lower head of the RPV, a stratified corium pool forms. The corium pool is composed of a thin top layer of unoxidized metallic melt and a bottom layer consisting of an oxidic or mixed (oxidic, metallic) melt composition. Decay heat and natural convection in the oxidic melt results in high heat densities at the upper edge of the pool, and the high thermal conductivity of the metallic layer redirects the heat flux towards the side of the RPV wall. Therefore, the composition of the melt initially escaping a lateral leak is considered to be mostly metallic while a central melt has an oxidic or mixed composition.
- Water Pool Depth: The depths considered for the water pool are 1.1 m[1] and 2.1 m; these depths correspond to two configurations that favor RPV central and lateral failure modes, respectively[2]. A water pool of 1.1 m depth would not provide adequate cooling to the lower head, favoring a central RPV failure mode. Any water depth over 1.1 m would provide cooling to the lower head and preclude, in most cases, central RPV failure modes. Subsequently, this configuration is more favorable to lateral leaks that are expected to occur at the welded junction of the two RPV structures (RPV wall and lower head) located at approximately 2.1 m from the reactor pit floor.

Deeper water pools (above the breach location) in the case of the U.S. EPR analysis would not impact the premixing of the melt in water because the melt column in the pool is limited by the breach location. The U.S. EPR design limits potential water heights preventing long

[1] All studies referenced in this analysis, including the IKE analysis, the SERENA paper and NUREG/CR-6849 are presented in SI units only. Therefore this unit system is used throughout this response and conversion to Imperial units was not performed.
[2] These are conservative values given that with the induced hot leg rupture scenario described; most of the accumulator water would be expected to remain in the RCS and RPV.

propagations of the explosion waves, and therefore reduces the potential strength of a steam explosion.

Ex-vessel Steam Explosion Loads

The results of the best estimate analysis using IKEMIX/IKEJET and IDEMO codes of bounding steam explosion loads in the U.S. EPR reactor pit are summarized in Tables 19-230-2 and 19-230-3 for a lateral leak, and in Table 19-230-4 for a central leak.

Tables 19-230-2 and 19-230-3 present the steam explosion loads resulting from a lateral melt outflow for oxidic and metallic melt compositions, respectively. The peak and average values of these pressure loads, as well as the impulse loads for the lateral melt outflow, are relevant for evaluating the reactor pit wall integrity.

The steam explosion loads from a central melt outflow are evaluated only for a water pool depth of 1.1 m. Also, because the failure occurs at the bottom of the RPV where the corium pool is dominated by an oxidic melt composition, the case of an oxidic melt is considered only for two jet diameters 10 and 40 cm. The average steam explosion loads on the melt plug resulting from a central leak are summarized in Table 19-230-4. The peak pressure loads evaluated in the steam explosion analysis at three different locations were averaged over the relevant areas to obtain the pressure loads on the melt plug at different times.

The pressure loads on the pit wall resulting from central RPV failures and pressure loads on the melt plug resulting from lateral RPV failures are of minimal impact, and are therefore not addressed.

Discussion of the Differences between the NUREG/CR-6849 and the U.S. EPR Steam Explosion Loads

The U.S. EPR plant-specific ex-vessel steam explosion analysis is based on the work performed by IKE. This analysis uses codes and methods that are comparable to those used in the study described in NUREG/CR-6849.

The study presented in NUREG/CR-6849 calculates the loads from an energetic fuel coolant interaction, using the codes PM-ALPHA (premixing) and ESPROSE.m (explosion). The base case (metallic melt, side failure of 0.4m diameter) predicts peak cavity wall pressures of 90 MPa, and up to 300 MPa in one sensitivity study (ceramic melt). These pressures are significantly higher than those predicted by IKE (maximum local wall pressure of 40MPa).

The differences between the two studies and the observed discrepancy in resulting loads are explained in the following sections. Three types of differences between the two analyses are evaluated:

- Differences in boundary conditions (e.g., water pool height).
- Differences in modeling assumptions (e.g., modeling geometry).
- Intrinsic differences in the codes.

Differences in Boundary Conditions

Differences between the reactor design in NUREG/CR-6849 and the U.S. EPR design and severe accident mitigation strategy result in a different set of boundary conditions at the onset of the steam explosion. By design, the NUREG/CR-6849 cavity is flooded, before vessel failure, with water from the in-containment refueling water storage tank (IRWST). The case analyzed in NUREG/CR-6849 assumes a deeply flooded cavity with a water level of more than 5 m above the cavity floor, and the pool is 50 K subcooled. By contrast, the U.S. EPR pit can only be wet as a result of a hot leg rupture resulting in a lower water depth (between 1 and 2 m) and saturated conditions.

These differences in the boundary conditions would contribute to higher loads in the AP1000 analysis. The higher water column in the reactor design in NUREG/CR-6849 acts as a confinement of the explosion wave and prevents venting to the open containment. The 50 K subcooled water results in a very low void fraction in the mixture, as opposed to the saturated case. Low voiding results in more water/melt interaction and higher explosion efficiency. This effect is magnified by code differences, as explained later in this response.

Differences in Modeling Assumptions

The analyses also use a different geometrical modeling of the steam explosion. NUREG/CR-6849 calculations are performed in a two-dimensional plane geometry, considering a vertical cross-section of the pit, whereas the IKE calculation uses a two-dimensional cylindrical geometry around the jet. This geometry accounts for pressure venting in the orthoradial direction, whereas this venting effect is limited in a geometrical model such as the one used in NUREG/CR-6849. The vertical plane model results in an increased confinement (resulting in higher loads) and accelerated reflection of the pressure peaks (resulting in higher impulses).

Differences Between Computer Codes

Intrinsic differences between the codes used also explain some of the observed differences. The NUREG/CR-6849 analysis uses different computer codes to model FCI than the U.S. EPR analysis. The codes used are PM-ALPHA (to model pre-mixing, counterpart to IKEMIX/IKEJET) and ESPROSE.m (to model the explosion, counterpart to IDEMO). These four codes are benchmarked against each other, against other codes, and against experimental data as part of the international program SERENA. A detailed comparison of the codes used in SERENA is presented in Reference 3. The most significant differences between these codes are in the premixing, which is related to the fuel jet fragmentation model and the water transition phase; this affects the predicted void fraction.

As explained in Reference 3, the physical model used to describe the corium jet fragmentation in PM-ALPHA is the “leading-edge breakup” (i.e., the corium jet is a discrete collection of drops, here with a constant 1 cm diameter). IKEMIX uses a continuous jet column fragmentation model, where small-scale jet fragmentation occurs in a radial direction along the melt jet. Therefore, with PM-ALPHA the entire column participates in the premixing, whereas IKEMIX predicts that only a portion of the melt would be in the mixture. For a given mass of melt in the water pool, this results in higher premixed masses for the NUREG/CR-6849 study than for the U.S. EPR study.

The model used for the water flow also differs between the two studies. The transition flow between bubbly flow (liquid with bubbles of gas) and droplet flow (steam with droplets of liquid) is modeled as a composition of both flows in IKEMIX, and as a churn flow (fixed length gas slugs) in PM-ALPHA. In PM-ALPHA, all corium droplets are in contact with water. As shown in Figure 3 of Reference 3, the amount of fuel in contact with water predicted by PM-ALPHA is higher than that predicted by IKEMIX when the void fraction increases. It is the enhanced fuel-water contact coupled with the lower void fraction that contributes to higher explosion efficiency in the NUREG/CR-6849 study.

It is important to note that each of the codes represents "state-of-the-art modeling" of a phenomenon for which little experimental data is available. Differences in modeling exist as a result of differences in opinion between the experts in charge of their development.

Summary of the Differences between the NUREG/CR-6849 and the U.S. EPR Analyses

Table 19-230-5 provides insights into the impact of the differences previously addressed, by comparing FCI parameters between equivalent cases of the two studies (side failure, metallic / oxidic melt).

To account for the different results between the two studies, three key elements were identified:

- Higher premixed mass is predicted in the NUREG/CR-6849 study due to code modeling differences.
- Higher explosion efficiency is predicted due to lower voiding in the NUREG/CR-6849 study, resulting from the assumed boundary conditions and by differences in codes.
- Higher confinement of the explosion wave occurs in the NUREG/CR-6849 study due to the deeper water pool and the geometrical configuration of the model.

Response to Question 19-230(a):

To support this response, a new structural analysis of the U.S. EPR reactor pit was performed to evaluate its capacity to withstand steam explosion dynamic pressure loads. The structural analysis focused on the two structures adjacent to the steam explosion locations identified. These structures include the pit wall for pressure loads resulting from lateral leaks and the melt plug for pressure loads resulting from a central leak.

Table 19-230-6 summarizes the structural analysis results for the pit wall and melt plug including the median pressure of failure, the structural uncertainty represented by a log normal standard deviation β, and the 10th and 90th percentile of the failure pressure. β was evaluated accounting for the uncertainty on the structural characteristics and the analytical model used.

The fragility curves of the pit wall and melt plug shown respectively in Figures 19-230-3 and 19-230-4 are derived as the cumulative functions of the lognormal distributions with the medians and standard deviations shown in Table 19-230-6.

The probability of rupture of a given structure is obtained by convolving the probability density function representing the fragility curve and the distribution of pressure loads on the structure through Monte Carlo sampling.

- For the pit wall, a uniform distribution of the pressure loads resulting from a lateral failure of the RPV with metallic outflow, a water pool depth of 1.1 m, and two bounding jet diameters 10 and 40 cm was used. The pressure loads selected are summarized in Table 19-230-7.
- For the melt plug, the best estimate pressure loads are summarized in Table 19-230-4.

A water pool depth of 1.1 m was selected for the lateral outflow because it has been found to be the most structurally challenging scenario. Loads from the 1.1 m pool depth scenario, although lower than those resulting from the 2.1 m water pool case, have been shown to impact the pit at a weaker location. Therefore, the net structural impact on the pit is more challenging for the 1.1 m case.

The probabilities of failures resulting from the best estimate analysis of the pit wall and the melt plug weighted by their respective probabilities of occurrence (94 percent and 5 percent) and a triggering probability of 0.86 (see the Response to Question 19-230(b)) are summarized as follows:

- Failure probability of the reactor pit wall: 0.
- Failure probability of the melt plug ~ 2E-03.

The melt plug failure probability is representative of the combined failure probability of the pit because the reactor pit wall does not fail under the best-estimate steam explosion loads. The combined failure probability of the reactor pit is conservatively rounded up to 5E-03.

Response to Question 19-230(b):

The best estimate load analysis provides a realistic evaluation of the steam explosion loads on the pit. An uncertainty analysis was developed to reproduce the expected steam explosion loads as a function of the important boundary conditions that would include the ranges provided in NUREG/CR-6849.

The boundary conditions for the parameters considered to be the most critical for the final loads are identified, and bounding ranges were assigned based on insights from the U.S. EPR specific analysis and from the literature referenced in this analysis. These parameters are addressed as follows:

- Jet Diameter: A bounding range used for the central leak is between 10 and 50 cm while the range for the lateral leak is between 50 cm and 100 cm.
- Conversion Ratio: The conversion ratio represents the steam explosion efficiency in converting the thermal energy contained in the superheated melt into an actual steam pressure load on the pit structure. Based on a literature review (References 7 and 8); the range considered was between 0.15 percent and 20 percent with a probability of 0.99 assigned to the range 0.5 percent to 5 percent.
- Breach Distance from the Wall: Because a steam explosion on the RPV lateral side is the most challenging for the wall, the shortest distance between the pit wall and the RPV side wall is assigned a probability of 0.5. The other possible distances (also assigned a probability of 0.5) are represented by a linear function of the shortest and the longest distance, where the longest distance is taken at the top metallic layer of the molten pool. These distances represent the positions of the lateral breach. The thickness of the metallic layer was estimated from MAAP runs.
- Triggering Probability: A uniform range between 0.72 and 1 was applied resulting in an average triggering probability of 0.86. This value was estimated based on Volume 3 of NUREG/CR-4551 (Reference 9) and is consistent with the discussion of the SERG-1 expert panel documented in Reference 10, as the current state of knowledge does not permit a better evaluation of this complex phenomenon.
- Premixing Mass Correlation: The correlation used was evaluated using a linear trend line to represent the dataset of premixed masses from multiple references, mainly the deterministic analysis performed for the U.S. EPR and the FZK Report (Reference 10).

Review of the premixing masses from the above references for single and multiple jets suggested that the premixing mass was strongly correlated with the interfacial area of the corium column. On this basis, the correlation used for the premixing mass was modeled as a linear function of the interfacial area of the corium column represented by the following equation:

$$\text{Premixed Mass (kg)} = \alpha(\text{kg/m}^2) * \text{Pool Depth(m)} * \text{Melt Jet Diameter (m)}$$

The coefficient α was obtained from a linear interpolation of the datasets previously addressed.

The results of a range of simulations can be represented by the above correlation as the median (best estimate). The lower and higher bounds of the distribution were generated taking into account the conclusions of the FZK report, on the capabilities of some premixing codes to predict different masses.

A variation factor between the premixing codes of 1.4 was noted in the FZK report, therefore, the same factor was applied to the above correlation to represent the lower bound and up to the 95th percentile of the upper bound. The upper bound for the distribution was selected to reflect the approach used in NUREG/CR-6849; a maximum premixed mass equivalent to the total melt mass flow into the pool over a period of 1 second.

The resulting pressure loads for a given set of input parameters were calculated using the blast equations from Reference 12. The premixing and blast correlations were benchmarked against IKE and NUREG/CR-6849 to gain confidence that the pressure loads reported can be reproduced with the appropriate boundary conditions.

Results of the Uncertainty Analysis on the Steam Explosion Loads

The pressure loads resulting from the above uncertainty analysis are:

- For a lateral RPV failure the median pressure load at the wall is 14.5 MPa with a 99.9th percentile load of 150 MPa. This pressure load is to be compared to NUREG/CR-6849 base case (metallic melt, side failure of 0.4 m diameter) for which the predicted peak cavity wall pressure is 90 MPa.
- For a central RPV failure, the median pressure load on the melt plug is 2.9 MPa with a 99.9th percentile of 41 MPa. There is no equivalent case in NUREG/CR-6849 to be compared to this scenario (bottom failure with ceramic melt).

The parameter distributions developed to represent the uncertainty on the steam explosion pressure loads accounted in their boundaries for the ranges considered in NUREG/CR-6849. It was concluded that the pressure loads reported in NUREG/CR-6849 match the higher end of the pressure loads distribution evaluated. It is noted that this observation is consistent with the review comments in the appendices of NUREG/CR-6849 that imply that the choice of modeling parameters used in the NUREG could be considered appropriate for a reasonably bounding analysis.

Under the ex-vessel steam explosion loads developed in the uncertainty analysis, the pit wall has a non zero probability of failure, and the melt plug has a lower probability of failure than the best estimate loads.

When convolving these load distributions (i.e., accounting for the uncertainty on the boundary conditions) with the fragility curves of the pit wall and melt plug, the combined pit probability of failure is evaluated to be 9E-4. This probability is lower than the best estimate failure probability (5E-3), which demonstrates that the boundary conditions considered in the best estimate analysis are bounding for realistic scenarios.

Response to Question 19-230(d):

The upward force exerted by the ex-vessel steam explosion loads on the RPV could result in an uplift of the vessel. The effect on the RPV depends on the location of the steam explosion, as well as the magnitude and duration of the blast. RPV uplift could result in movement of the connecting piping, and has the potential to impact the integrity of the containment penetrations.

To assess the potential for RPV uplift impacting containment penetrations, a structural analysis was performed using a concentrated-mass model of the RPV in ANSYS (Reference 13) to calculate the vertical displacement resulting from a range of credible uplift loads on the RPV due to ex-vessel steam explosion.

The structural analysis evaluates the central leak steam explosion as the limiting case for RPV uplift, given that the potential for vessel uplift from a lateral leak was determined to be minimal. The uplift analysis was performed using loads on the RPV lower head that span a range from 8 to 105 MPa. This range envelops the loads obtained from both the best estimate steam explosion (Question 19-230(c)) and the uncertainty analysis (Question 19-230(b)) applicable to RPV uplift from the central leak. Although the resulting peak force obtained by integrating these loads over the lower head surface is approximately 200 MN, the effective force on the strap-assembly located near the top of the RPV is found to be less than 1 MN. This attenuation is due to the effect of the downward gravitational force of the RPV mass and the absorption of the upward force by deformation of the lower head structure.

The maximum vertical displacement of the lower head predicted by the analysis is less than 4E-3 m. The maximum force exerted on the RPV strap-assembly is found to be less than 1 MN, well within the elastic yield of the strap assembly. Therefore, it is concluded that no significant RPV uplift would occur and that the impact on the strap assemblies is negligible. Thus, no impact is expected on the piping that connects with the primary system and penetrates containment, so there will be no negative impact on containment integrity.

A set of parametric analyses have also been performed to evaluate the impact of varying important input parameters (e.g., RPV mass, temperature, stiffness of the lower head, ultimate displacement of the lower head). The analyses concluded that realistic variations of these input parameters would not change the conclusions.

The loads used as input to the structural evaluation envelop the loads from both the best estimate and the uncertainty evaluations conclude that the impact of RPV uplift on the associated piping and containment penetrations can be neglected.

Response to Question 19-230(e):

The impact of the reactor pit failure on the severe accident progression has been evaluated as part of the Level 2 PRA. The most likely failure mode of the reactor pit is the failure of the melt plug. The reactor pit and melt plug are designed to confirm temporary retention of the melt before the transfer to the corium spreading area. Without a retention period, there could be failure of complete melt transfer to the spreading area, premature flooding of the spreading area, or failure of the melt transfer to the spreading area.

Failure of the melt plug to perform its function following any significant relocation of corium into the pit is not considered credible. Zirconia brick (highly resistant to ablation) protects the structural concrete, whereas melt plug does not. As such, the melt plug is a designed failure location.

If premature flooding of the core melt spreading area occurs, the melt pour could relocate into a water pool, thus creating an opportunity for a fuel-coolant interaction in the spreading area. The most likely scenario is during a severe accident the melt plug fails before all the core material has relocated to the reactor pit. The melt resident in the reactor pit flows into the spreading room, triggering compartment flooding. If at some time following the end of compartment flooding, core material resident in the reactor vessel relocates to the reactor pit, it will immediately flow into the water-filled transfer channel. Because the amount of concrete designed for the reactor pit has considered the anticipated thermal loads from a large melt relocation event, it is likely that only a small fraction of material would participate in this kind of scenario. As described in the U.S. EPR Severe Accident Evaluation Topical Report, ANP-10268P-A (Reference 14), tests performed at FARO (Reference 4) indicate that dry conditions are not required for successful melt spreading and that steam explosion is an unlikely consequence. If a steam explosion occurs, the loads on the reactor pit anticipated by this scenario would be within those described in the Response to Question 19-230(c).

A conservative approach has been adopted in the Level 2 PRA to address the broader issue related to the failure of ex-vessel melt stabilization leading to extensive molten corium-to-concrete interaction (MCCI) and eventual basemat failure. Extensive MCCI could impact the severe accident progression through two mechanisms leading to containment failure. These mechanisms are basemat ablation and overpressurization due to non-condensable gases. These are very slow mechanisms (in the order of 10 days); therefore any other failure mode (e.g., containment isolation failure or hydrogen deflagration/flame acceleration) would occur earlier. For sequences where containment failure occurs prior to or at vessel failure, the occurrence of MCCI modifies the source term. The process used to define release category systematically differentiates between sequences where MCCI occurs and sequences where it does not, and assigns an appropriate source term.

For sequences where containment failure would not have occurred prior to or simultaneously with vessel failure, failure modes, specific to MCCI become relevant. The most likely failure mode is basemat penetration (conditional probability of 0.98 given extended MCCI and no other failure). Overpressurization is a significantly less likely failure mode. These failure modes correspond to a large release, as defined in U.S. EPR FSAR Tier 2, Section 19.1.4.2.1.3.

Conclusion and Impact on the Level 2 PRA Results

As a result of the analysis described in this response, probabilities associated with ex-vessel steam explosion have been re-evaluated. As explained in the Response to Question 19-230(a), the combined failure probability of the pit due to steam explosion is 5E-03 given that the conditions for a steam explosion are met. The Level 2 PRA model was updated to incorporate this probability. The logic was also modified so that this probability is only applied when a hot leg rupture has occurred (this is the only scenario in which corium discharge into a stable pool of water is possible).

The impact of this change on large release category frequencies for internal events, fire, and flooding is shown in Table 19-230-8, which also shows that there is no change in the total large release frequency. Any additional containment failure sequence due to pit failure would occur in a late time frame, and impact release categories (i.e., basemat failure), which do not constitute a large release as defined in U.S. EPR FSAR Tier 2, Section 19.1.4.2.1.3.

Table 19-230-8 shows significant relative changes in the frequency of some of the individual large release categories. This represents source term changes due to an increase in the occurrence of MCCI following containment failure. The largest relative increases are found in RC203 and RC302, which are release categories with MCCI and no containment sprays. Sequences where containment sprays are unavailable generally correspond to core damage sequences with multiple equipment failures. Manual depressurization has failed where multiple equipment failures occur, thus favoring the occurrence of hot leg rupture and the conditions for a steam explosion. However, the total contribution of these sequences to the large release frequency (LRF) is very small (<<1 percent).

The re-evaluation of the ex-vessel steam explosion results in a different distribution of the total LRF between the individual release categories. The net effect on the LRF for internal fire and flood events is zero. Risk insights from the Level 2 PRA, including importance measures, significant initiators, and significant core damage end states, are calculated based on LRF and are therefore not affected.

U.S. EPR FSAR Tier 2 will be revised to include the new ex-vessel steam explosion evaluation methodology and the revised release category frequencies as follows:

- Section 19.1.4.2 will be revised to include the new ex-vessel steam explosion evaluation methodology and the revised release category frequencies for internal event LRF.
- Tables 19.1-24, 19.1-50, 19.1-75 and 19.1-105 will be revised to include the revised release category frequencies for LRF.
- Table 19.1-25 will be revised to include updated significant cutsets for internal events LRF.

Table 19-230-1—Boundary Conditions for the Best Estimate Loads Calculation

Scenarios	Parameters			
Lateral leak of the RPV wall	Water pool depth in the pit	1.1 m / 2.1 m		
	Melt composition	Oxidic / Metallic		
	Jet diameter	10 cm	20 cm	40 cm
Central leak of the lower head	Water pool depth in the pit	1.1 m		
	Melt composition	Oxidic		
	Jet diameter	10 cm	40 cm	

Table 19-230-2—Summary of Pressure Loads on U.S. EPR Reactor Pit Side Wall for Metallic Melt Flow out of Lateral RPV Failure

Jet Diameter (cm)			10		20		40	
Water Level (m)			1.1	2.1	1.1	2.1	1.1	2.1
Maximum Local Pressure (MPa)			10.1	15.8	13.4	17.9	37.2	15.2
Averaging Area 10x10 cm	Impulse at 5 ms (kPa.s)		1.5	3.5	11.2	12.2	13.6	13.9
	Impulse at 10 ms (kPa.s)		2.6	19.2	15.1	20.3	21.9	20.8
	Pressure (MPa)	Averaged over 0.01 ms	5.0	9.7	8.6	12.8	11.4	10.6
		Averaged over 0.1 ms	4.6	9.6	8.4	12.8	11.0	10.2
		Averaged over 0.2 ms	4.2	9.5	8.3	12.5	10.0	10.0
		Averaged over 0.5 ms	3.5	9.0	7.7	11.4	9.1	9.2
		Averaged over 1 ms	3.0	7.9	6.4	9.2	7.5	7.6
Averaging Area 25x25 cm	Impulse at 5 ms (kPa.s)		1.4	3.2	9.8	10.7	12.8	12.5
	Impulse at 10 ms (kPa.s)		2.4	18.1	13.6	19.2	19.1	19.5
	Pressure (MPa)	Averaged over 0.01 ms	2.3	7.3	5.5	8.0	7.7	6.7
		Averaged over 0.1 ms	2.1	7.3	5.3	7.9	7.3	6.6
		Averaged over 0.2 ms	2.0	7.3	5.2	7.9	7.3	6.5
		Averaged over 0.5 ms	1.8	7.1	5.1	7.7	7.0	6.3
		Averaged over 1 ms	1.5	6.6	4.6	7.0	6.2	5.7
Averaging Area 50x50 cm	Impulse at 5 ms (kPa.s)		1.2	2.6	7.6	8.0	10.2	10.2
	Impulse at 10 ms (kPa.s)		2.2	14.8	11.8	16.1	17.4	18.5
	Pressure (MPa)	Averaged over 0.01 ms	0.9	4.9	2.9	4.4	4.4	3.9
		Averaged over 0.1 ms	0.9	4.9	2.8	4.4	4.3	3.8
		Averaged over 0.2 ms	0.9	4.9	2.8	4.3	4.3	3.8
		Averaged over 0.5 ms	0.8	4.8	2.8	4.3	4.1	3.7
		Averaged over 1 ms	0.7	4.6	2.7	4.1	4.0	3.6

Table 19-230-3—Summary of Pressure Loads on U.S. EPR Reactor Pit Side Wall for Oxidic Melt Flow out of Lateral RPV Failure

Jet Diameter (cm)			10		20		40	
Water Level (m)			1.1	2.1	1.1	2.1	1.1	2.1
Maximum Local Pressure (MPa)			21.7	28.1	10.1	22.5	43.7	39.3
Averaging Area 10x10 cm	Impulse at 5 ms (kPa.s)		10.9	15.4	1.6	11.7	18.2	18.4
	Impulse at 10 ms (kPa.s)		14.2	32.8	2.7	15.4	24.4	33.5
	Pressure (MPa)	Averaged over 0.01 ms	8.4	16.3	6.4	9.5	20.1	35.3
		Averaged over 0.1 ms	8.1	16.1	5.2	9.0	17.4	21.2
		Averaged over 0.2 ms	8.1	16.0	4.9	9.0	15.3	18.5
		Averaged over 0.5 ms	7.2	15.0	3.8	8.2	10.8	13.2
		Averaged over 1 ms	5.8	12.9	2.4	6.5	8.6	10.5
Averaging Area 25x25 cm	Impulse at 5 ms (kPa.s)		9.1	13.8	1.5	9.9	17.0	17.9
	Impulse at 10 ms (kPa.s)		12.4	31.3	2.5	13.6	23.2	32.5
	Pressure (MPa)	Averaged over 0.01 ms	4.8	12.9	3.3	5.3	9.0	31.2
		Averaged over 0.1 ms	4.5	12.8	2.6	5.1	8.6	21.2
		Averaged over 0.2 ms	4.5	12.7	2.5	5.0	8.5	17.6
		Averaged over 0.5 ms	4.3	12.2	2.0	4.7	8.2	12.8
		Averaged over 1 ms	3.9	10.9	1.6	4.3	7.3	10.3
Averaging Area 50x50 cm	Impulse at 5 ms (kPa.s)		6.9	9.8	1.4	7.5	13.8	17.3
	Impulse at 10 ms (kPa.s)		10.2	27.1	2.2	11.4	20.0	31.3
	Pressure (MPa)	Averaged over 0.01 ms	2.5	8.1	1.5	2.7	5.6	31.2
		Averaged over 0.1 ms	2.5	8.1	1.3	2.7	5.4	21.2
		Averaged over 0.2 ms	2.5	8.0	1.2	2.7	5.4	17.6
		Averaged over 0.5 ms	2.4	7.9	1.0	2.7	5.2	12.7
		Averaged over 1 ms	2.3	7.5	0.9	2.5	4.9	10.1

Table 19-230-4—Summary of Pressure Loads on U.S. EPR Reactor Pit Melt Plug for Oxidic Melt Flow out of a Central RPV Failure

Jet Diameter (cm)	Time (ms)	Pressure (MPa) (At a given distance from the center of the melt plug)			Average Pressure Loads on the Melt Plug (MPa)
		P(R=0 m)	P(R=0.5 m)	P(R=1 m)	
10	3	5	4.0	2.5	3.6
	4	6.8	2.5	1.5	2.5
40	2	20	8.0	0.0	6.3
	3	14	10.0	4.5	8.5

Table 19-230-5—Comparison of Key FCI Parameters Between the NUREG/CR-6849 and U.S. EPR Studies[1]

	NUREG/CR-6849 Base Case (metallic)	NUREG/CR-6849 Ceramic Melt	U.S. EPR Oxidic Case 2.1m	U.S. EPR Metallic Case 2.1m
Boundary Conditions				
Water pool height	5 m	5 m	2.1 m	2.1 m
Water state	50 K subcooled	50 K subcooled	Saturated	Saturated
Height of melt release	2 m	2 m	2.5 m	2.5 m
Breach diameter	0.4 m	0.4 m	0.4 m	0.4 m
Amount of melt in column	1500 kg	1800 kg	1300 kg	1200 kg
Premixing Parameters				
Premixing Code	PM-ALPHA	PM-ALPHA	IKEMIX	IKEMIX
Jet fragmentation model	Leading edge breakup	Leading edge breakup	Continuous jet fragmentation	Continuous jet fragmentation
Mass in premixing	1500 kg	1800 kg	380 kg	425 kg
Mass in premixing with less than 60% voiding	~1500 kg (no apparent voiding)	~1800 kg (no apparent voiding)	31 kg	60 kg
Results				
Maximum Pressure Load on the cavity walls	90 MPa	290 MPa	35 MPa	11 MPa

Notes:

1. Some of the parameter values presented in the table are estimated from the interpretation of the reference documents when no numerical value was provided.

Table 19-230-6—Structural Capacity of the Pit Wall and Melt Plug

Location	Median Failure Pressure	ß (=σ)	Failure Pressure 10th Percentile	Failure Pressure 90th Percentile
Pit Wall	206.7 MPa	0.15	170.54 MPa	250.5 MPa
Melt Plug	8.97 MPa	0.15	7.4 MPa	10.87 MPa

Table 19-230-7—Best Estimate Pressure Loads on the Pit Wall used for the Failure Probability Calculation

Jet Diameter (cm)		10	40
Pressure (MPa) averaged over an area of 10x10 cm	Averaged over 0.01 ms	5.0	11.4
	Averaged over 0.1 ms	4.6	11.0
	Averaged over 0.2 ms	4.2	10.0
	Averaged over 0.5 ms	3.5	9.1
	Averaged over 1 ms	3.0	7.5

Table 19-230-8—Change in Release Category Frequency due to FCI Re-evaluation for Total LRF (Internal, Fire and Flooding Events)

(Sheet 1 of 2)

Release Category	Description	Current FSAR frequency (1/yr)	Frequency with new FCI analysis (1/yr)	Difference
RC201	Containment fails before vessel breach due to isolation failure, melt retained in vessel	5.0E-10	5.0E-10	0%
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, with containment sprays	4.0E-14	4.0E-14	-1%
RC203	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, without containment sprays	8.5E-13	1.9E-12	127%
RC204	Containment fails before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel with containment sprays	2.4E-11	2.8E-11	15%
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel without containment sprays	4.1E-10	4.1E-10	0%
RC301	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, with containment sprays	1.6E-12	1.7E-12	3%
RC302	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, without containment sprays	1.5E-11	2.2E-11	44%
RC303	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, with containment sprays	2.3E-09	2.3E-09	0%
RC304	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, without containment sprays	1.8E-08	1.8E-08	0%

Table 19-230-8—Change in Release Category Frequency due to FCI Re-evaluation for Total LRF (Internal, Fire, and Flooding Events)

(Sheet 2 of 2)

Release Category	Description	Current FSAR frequency (1/yr)	Frequency with new FCI analysis (1/yr)	Difference
RC702	Steam Generator Tube Rupture without Fission Product Scrubbing	5.4E-09	5.4E-09	0%
RC802	Interfacing System LOCA without Fission Product Scrubbing but building credited	2.6E-10	2.6E-10	0%
Total LRF	Sum of all large release category frequencies	2.6E-08	2.6E-08	0%

Figure 19-230-1—Lateral RPV Failure Configuration

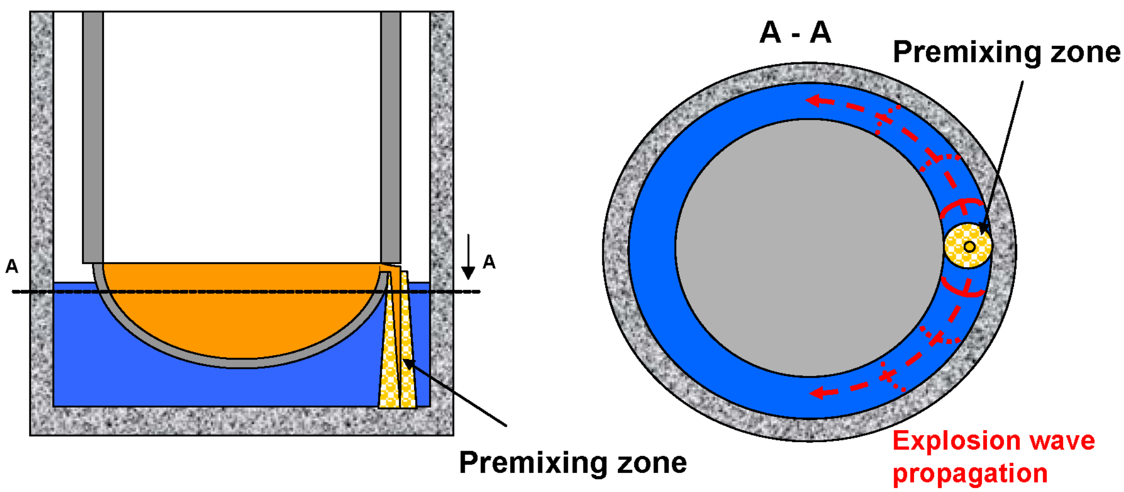


Figure 19-230-2—Central RPV Failure Configuration

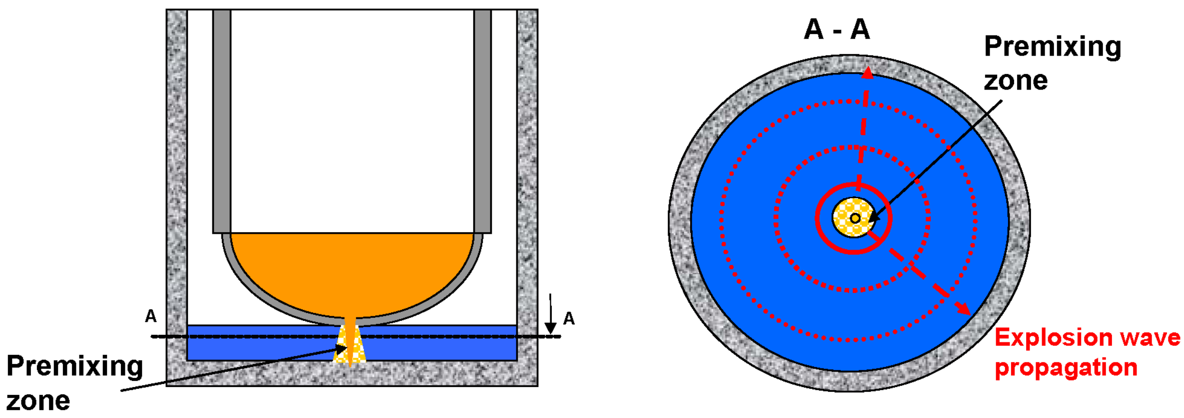


Figure 19-230-3—U.S. EPR Reactor Pit Wall Fragility Curve

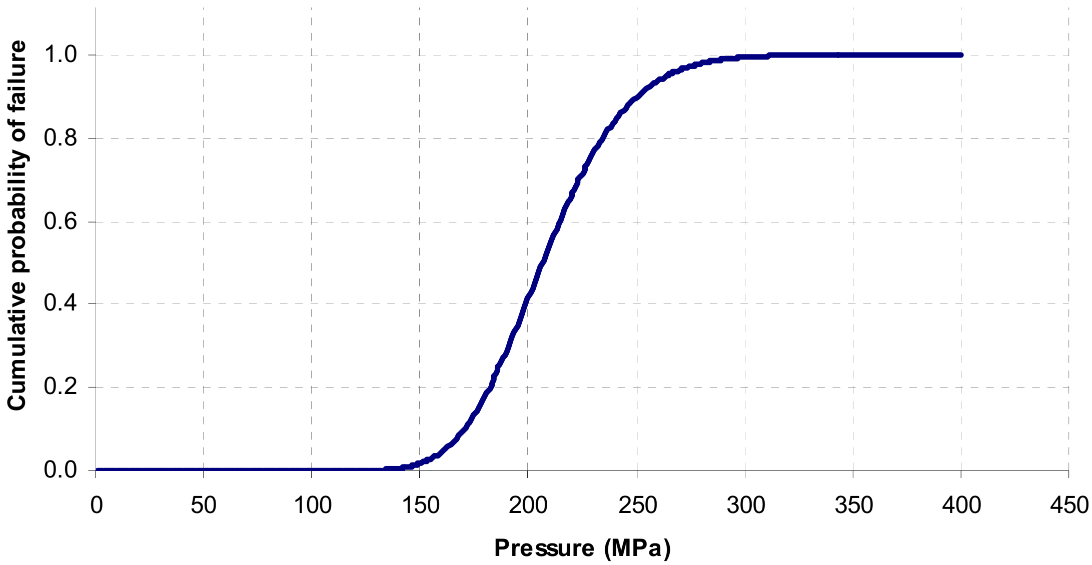
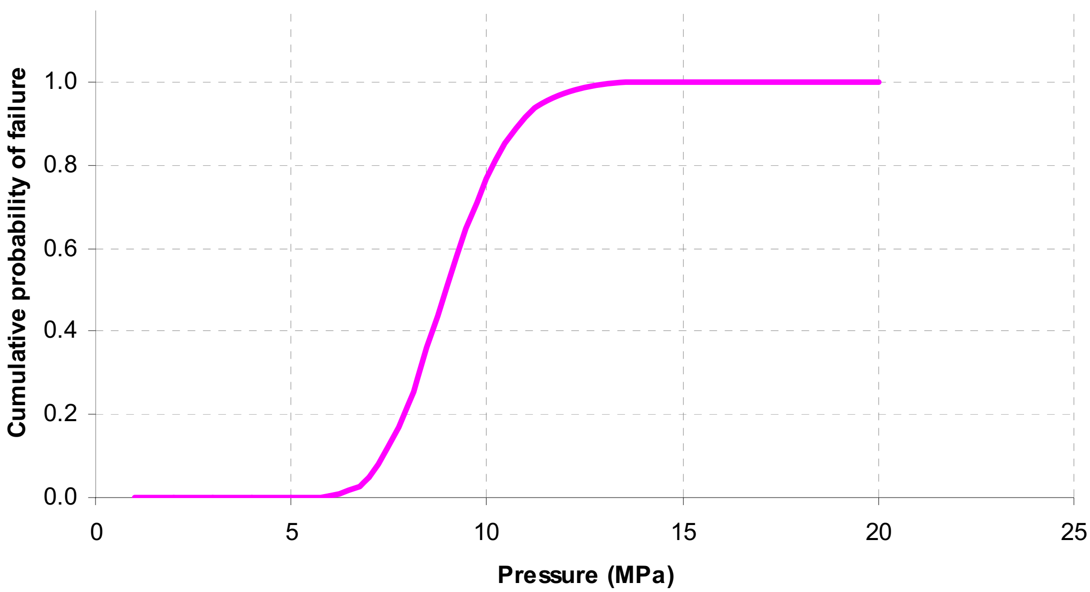


Figure 19-230-4—U.S. EPR Melt Plug Fragility Curve



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FSAR Impact:

U.S. EPR FSAR Tier 2, Section 19.1.4.2.1.2 and Tables 19.1-24, 19.1-25, 19.1-50, 19.1-75, and 19.1-105 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

difference between the two evaluations is that the factor for the fraction of the mechanical energy that is transmitted to the slug that impacts the upper head was not applied for the lower head evaluation. Rather 100 percent of the mechanical energy was assumed to impact the lower head. This assumption is conservative.

The results of the probabilistic evaluation of a steam explosion causing failure of the lower head were approximately as follows:

- A value of 8.4E-04 for a high pressure core melt scenario.
- A value of 2.5E-05 for a low pressure core melt scenario.

Ex-vessel Steam Explosion

19-230

~~Ex-vessel steam explosions were evaluated for scenarios in which molten corium could be released from the vessel into a wet pit. In general, the EPR pit is expected to be dry. However, two scenarios were identified in which this may not be the case and ex-vessel steam explosions were, therefore, considered for the following cases:~~

1. ~~Pour of molten corium into an ex-vessel pool at vessel failure for a sequence that has the RCS depressurized due to an induced hot leg rupture (located at the RV nozzle) leading to the spillage of water into the reactor pit. In this case the flow of corium into the pool is at the rate occurring at the time of vessel failure. MAAP analyses confirmed that in this scenario, with failure at the RV nozzle (this being the most likely failure location), a water pool approximately 4m in depth develops in the reactor pit.~~

2. ~~Pour of molten corium into an ex-vessel water pool in the longer term, after vessel failure, due to the long-term melting of the remaining core material not in the lower head at the time of vessel failure. In this case, the pour may be into an ex-vessel pool that has accumulated because of safety injection water which is lost into the pit after vessel breach. In this case, it is considered likely that the remaining core material in the vessel would freeze rather than melt and fall into the ex-vessel water pool. The pour rates are also anticipated to be lower than in Case 1 above. In view of the results obtained for Case 1 (see below), which predict low probabilities of an ex-vessel steam explosion causing pit damage, it was considered acceptable to bound this scenario (Case 2) using the results of Case 1.~~

~~The ex-vessel steam explosion analysis is based on a comparison of impulse loading on the cavity structures to their strengths. The impulse loading is evaluated in two steps. The first step was to evaluate the mechanical energy release following a similar process to that used for the in-vessel steam explosion. Specifically the mechanical energy release was evaluated by multiplication of:~~

1. ~~The total mass of corium in premixing.~~

19-230

2. ~~The thermal energy stored in the core materials per unit mass of core. (It is assumed that the composition of the molten core in the lower plenum maintains the same proportions of materials in the proportions present in the core as whole.)~~

3. ~~The conversion ratio for thermal to mechanical energy.~~

~~As in the case of the in-vessel steam explosion analyses, the total load was evaluated probabilistically using Monte Carlo simulations. Items (2) and (3) were evaluated using the same distributions as for the in-vessel steam explosion. The total mass of corium in pre-mixing was, however, re-evaluated for the ex-vessel scenario to take into account the expected flows into the ex-vessel water pool and the depth of this pool.~~

~~The second step was to evaluate the impulse loading by translating the mechanical energy release to an impulse. This was performed by use of a correlation relating energy release to peak overpressure and duration.~~

~~Finally, the impulse loading probability distribution was compared to the impulse loading capacity of the reactor pit structures. As in the cases of in-vessel steam explosions, the capacity of the structures was assigned a probability distribution. It is expected that the major structures of the EPR reactor pit (including the plug) are likely to withstand an impulse loading of at least 1.46 psi-s. The probability distribution assigned also contemplates lower values, with capacities in the range 0.73-1.46 psi-s being considered possible (but not probable) and a residual probability being assigned to allow for capacities as low as 0.29-0.73 psi-s. An upper capacity of 2.92 psi-s was taken in developing the probability distribution.~~

~~The evaluation generated a probability of structural damage due to an ex-vessel steam explosion of approximately 2.6E-05. As mentioned above, this value was generated by comparing a distribution of loads to a distribution of pit capacity. This comparison was carried out using Monte Carlo simulation. The result was generated based on the conditions expected for a hot leg rupture with discharge of water into the pit and release of molten corium into that water at vessel failure. To simplify the CET modeling, this value is also conservatively applied to model the effects of an ex-vessel steam explosion occurring due to the release of long term melt into an ex-vessel water pool in the period after vessel failure.~~Ex-vessel steam explosions were evaluated for scenarios in which molten corium is released from the vessel into a stable water pool in the reactor pit cavity. An evaluation of the relevant RCS failure modes concluded that only creep-induced hot leg rupture at the RV nozzle could lead to a stable water pool in the reactor pit at the time of RV failure. A probabilistic evaluation of the consequences of an ex-vessel steam explosion is performed for that specific scenario.

An important parameter for this assessment is the RV rupture location. The probabilistic evaluation of vessel failure described later in this sub-section concluded that among the possible RV failure modes, the lateral failure is the most likely failure

19-230

location. This is due to the focusing effect at the junction of the oxidic and metallic layers of the corium pool, leading to high heat densities in proximity of the RV wall. Based on this evaluation it was concluded that:

- The lateral failure mode represents 94 percent of the RV failure modes. Steam explosion loads from a lateral melt outflow could challenge the structural integrity of the pit wall.
- The central failure scenario represents 5 percent of the RV failure modes. Steam explosion loads from a central melt outflow could fail the melt plug.

The remaining 1 percent represents complete circumferential failure modes that have no impact on steam explosion scenarios.

The impact of an ex-vessel steam explosion on the pit wall and the melt plug was evaluated through a comparison of the dynamic pressure loads on these structures to their respective strengths. This evaluation was performed in two steps; first the best estimate dynamic loads resulting from an ex-vessel steam explosion under realistic conditions were estimated, then these loads were compared to the probability density function representing the fragility of the pit structure.

The dynamic pressure loads used in this evaluation are the result of a deterministic analysis performed by the University of Stuttgart Institute for Nuclear Technology and Energy Systems (IKE). In order to envelop the range of realistic scenarios, the analysis used different sets of initial conditions such as the leak location and size, flow rate, melt temperature and composition, and water pool depth. The resulting pressure loads reached a maximum of 12 MPa on the pit wall with a metallic melt composition and a maximum of 9 MPa on the melt plug with an oxidic melt composition.

The fragility curves used in this evaluation are the result of a structural evaluation of the pit wall and the melt plug responses to the steam explosion loads evaluated above. This evaluation concluded that the maximum steam explosion loads that the pit wall and the melt plug withstand with a zero probability of failure are 161 MPa and 8 MPa, respectively.

The comparison of the pressure loads against the pit wall and melt plug structural strengths was accomplished through a Monte Carlo sampling and resulted in a conditional probability of failure for the pit wall (given a lateral leak) and for the melt plug (given a central leak).

The probabilities of failures of the pit wall and the melt plug are then weighted by their respective probabilities of occurrence (94 percent and 5 percent). This yields a total failure probability of the pit of approximately 2E-03 conservatively rounded up to 5E-03.

19-230

The CET logic reflects the conditions necessary for steam explosion by applying the calculated probability of pit failure only to core damage sequences depressurized by hot leg rupture prior to RV failure.

An analysis of the impact of the reactor pit failure on the severe accident progression has been performed in light of the results of the above analysis that identified the melt plug as the weakest structure in the pit. The purpose of the melt plug sacrificial material is to provide temporary retention of the melt before the transfer to the corium spreading area. Without a retention period, this release would create undefined and potentially unfavorable conditions for subsequent melt spreading. A conservative approach has been adopted in the Level 2 PRA which assumes that an early release of the melt will result in failure of melt stabilization ex-vessel and subsequent molten core concrete interaction (MCCI) with a probability of one.

In-Vessel Core Recovery

The principal cause of core heat-up in a severe accident is the lack of cooling water. Depending on the time when safety injection (SI) is recovered, the accident progression can be stopped or delayed. Thus the SI recovery time has a direct impact on the RCS and containment conditions after injection is initiated to a degraded core. Depending on the injection flow rate, the hot corium can either be quenched or not. Too little flow, and the accident progression is delayed, but reactor vessel failure is not prevented.

The effects of the re-flooding of a damaged core include an enhanced oxidation leading to temperature escalation and high hydrogen peaks. Flooding a damaged core can also lead to the formation of a debris bed due to thermal shock collapse of the upper fuel rods located above the core molten pool, as with the Three Mile Island (TMI) accident.

A severe accident starts with insufficient cooling conditions in the core followed by continuous heat-up of the fuel. The heat transferred from the fuel rods to the steam is not sufficient to remove all decay heat, but is able to heat-up the steam close to the highest temperature of the fuel rods that normally occurs at the top of the core. Core exit temperature of the steam is therefore a measure of the early accident progression and is therefore used as a criterion for dedicated bleed (approximately 1200°F).

To mitigate further accident progression, in particular the consequences of a high pressure core melt scenario, the RCS depressurization strategy aims at opening the depressurization valves to allow injection of available safety injection and accumulators before the start of core melt. If the depressurization and the injection of the SIS accumulator or the LHSI are not successful, fuel element degradation will continue.

The exothermic reaction of the superheated steam with the Zirconium (Zr) of the fuel rods produces hydrogen, which is transported with the remaining steam through the

Table 19.1-24—Internal Events Release Category Results - Large Release Frequency
Sheet 1 of 2

19-230

Release Category	Description	Mean	Contribution to LRF	Conditional Containment Failure Probability
RC201	Containment fails before vessel breach due to isolation failure, melt retained in vessel	4.5E-10	2.1%	0.0016
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, with containment sprays	3.8E-14	0.0%	0.0
RC203	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, without containment sprays	~~5.9E-13~~1.6E-12	0.0%	0.0
RC204	Containment fails before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel with containment sprays	~~2.4E-11~~2.7E-11	0.1%	0.0001
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel without containment sprays	3.3E-10	1.5%	~~0.0011~~0.001
RC301	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, with containment sprays	1.3E-12	0.0%	0.0
RC302	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, without containment sprays	~~3.1E-12~~8.8E-12	0.0%	0.0
RC303	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, with containment sprays	1.7E-09	7.7%	~~0.0058~~0.006

Table 19.1-24—Internal Events Release Category Results - Large Release Frequency
Sheet 2 of 2

Release Category	Description	Mean	Contribution to LRF	Conditional Containment Failure Probability
RC304	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, without containment sprays	1.4E-08	~~66.4%~~66.3%	~~0.049~~0.05
RC702	Steam Generator Tube Rupture without Fission Product Scrubbing	4.6E-09	21.0%	0.016
RC801	Interfacing System LOCA with Fission Product Scrubbing	0.00E+00	0.0%	0.0
RC802	Interfacing System LOCA without Fission Product Scrubbing but building credited	2.6E-10	1.2%	0.0009
	Total LRF:	2.2E-08	100.0%	0.076

19-230

Table 19.1-25—Level 2 Internal Events Large Release Significant Cutsets
Sheet 3 of 14

19-230

Release Category	Freq /yr	Contribution to LRF (%)	Event Identifier	Event Description	Sequence of events that lead to CD and to Containment Failure
Internal RC203 - 1, 2, 3	~~1.55E-14~~1.18E-13	~~0.0001%~~0.0005 %	~~IE SLBI~~IE LOOP	Initiator - ~~Steam Break Inside Containment~~Initiator-Loss of Offsite Power	Level 1: • ~~SLBI initiator with consequential LOOP. EDG 2 fails to run, failure to crosstie results in the loss of all Division 2 power.~~LOOP initiator with non-recovery of OSP • ~~SAC4 is in maintenance, and LOOP fails the maintenance HVAC train~~CCF of ventilation in Division 1 and 4 and failure to recover room cooling locally leads to permanent loss of Division 1 and 4. • ~~Operator fails to recover room cooling locally, so electrical buses in Division 4 fail, failing Division 3 ventilation~~ • ~~The MSRTs close due to the loss of Division 2 and 4. Steam relief via MSSVs require 2 EFWs. Only train 1 is available.~~ • ~~Primary bleed is lost due to loss of Division 4~~
			~~LOOPCON+REC~~REC OSP 2HR	~~Consequential LOOP and Failure of Recovery Within 1 Hour for LOCA IEs~~Failure to Recover Offsite Power within 2 Hours	
			~~XKA20 DFR~~QKA 10GH001 FS B-ALL	~~ELEC, Emergency Diesel Generator XKA20, Fails to Run~~CCF of the Air Cooled SCWS Chiller Units	
			~~OPF-XTDIV-NSC~~SAC04/QKA40 PM4	~~Operator Fails to Xtie Division 1 to Division 2 or Division 4 to Divison 3 During Non-SBO Conditions~~Normal SAC04/QKA40 Train Unavailable due to Preventivie Maintenance	
			~~SAC04/QKA40 PM4~~	~~Normal SAC04/QKA40 Train Unavail due to Preventive Maint~~	
			OPF-SAC-2H	Operator Fails to Recover Room Cooling Locally	

Table 19.1-25—Level 2 Internal Events Large Release Significant Cutsets
Sheet 4 of 14

19-230

Release Category	Freq /yr	Contribution to LRF (%)	Event Identifier	Event Description	Sequence of events that lead to CD and to Containment Failure
			~~L2FLCDES-TR1D~~L2PH CPIHLR-TR.TP=Y	~~Level 2 FLAG: TR1-CDES~~Induced hot-leg rupture. Conditional probability given no ISGTR, TR, TRD, TP, TPD cases	Level 2: • ~~Sequence enters CET1 High Pressure~~ • ~~Operators fail to depressurize in both the EOPs and OSSA/SAGs~~ • ~~Sequence enters CET2 High Pressure~~ • ~~SLBI requires SG blowdown line to isolate on CI signal. One line fails to isolate on loss of Division 2 and 3.~~ • ~~Pit damaged due to overpressure from complete circumferential rupture of the vessel~~ • ~~MCCI occurs due to early melt release from the pit.~~ • ~~SAHRS sprays fail to control source term due to the loss of electrical train 4~~ • Sequence enters CET1 High Pressure • Primary system depressurizes due to hot leg rupture • Sequence enters CET Low Pressure • Containment isolation fails due to loss of Division 1 and 4 power supplies to sets of initially open containment isolation lines
			~~L2FLCET1 HI-PRESSURE~~PROB KTA10 17/18 OP PROB KTD10 24/15 OP PROB KTC10 05/06 OP	~~Level 2 FLAG: CET1 HI-PRESSURE~~Probability that Primary Drain line KTA10, NCS line, or containment sump line KTC is open.	
			~~L2PH LOCA-DEPRESS=N~~KPL85 03/04 HPFL	~~Primary remains pressurized until vessel failure~~Probability that GWP system fails on containment high pressure	
			~~L2FLCET2 HI-PRESSURE~~L2 REC=Y OSP 2-7H	~~Level 2 FLAG: CET2 HI-PRESSURE~~Offsite power recovered between 2 and 7 hours	
			~~L2PH CBV HP~~L2PH STMEXP EX	~~Complete circumferential rupture of vessel~~Ex-vessel steam explosion damages reactor pit	
			~~L2PH CP-PITF-VF(CBV)~~	~~Pit overpressure at high pressure vessel failure fails melt plug given CBV occurs~~	
			L2PH CCI-EARLYREL=Y	MCCI occurs, following early melt release from pit.	

Table 19.1-25—Level 2 Internal Events Large Release Significant Cutsets
Sheet 5 of 14

Release Category	Freq /yr	Contribution to LRF (%)	Event Identifier	Event Description	Sequence of events that lead to CD and to Containment Failure
				19-230	• Offsite power is recovered 2-7 hours, but Division 1 and 4 remain deenergized • Ex vessel steam explosion at vessel failure leads to melt plug failure • MCCI occurs due to early melt release from pit. • SAHRS sprays fail to control source term due to loss of Division 1 and 4. Offsite power recovery does not play a role since the buses are failed.

Table 19.1-25—Level 2 Internal Events Large Release Significant Cutsets
Sheet 9 of 14

Release Category	Freq /yr	Contribution to LRF (%)	Event Identifier	Event Description	Sequence of events that lead to CD and to Containment Failure
Internal RC302	~~1.26E-13~~6.3E-13	~~0.0006%~~0.003%	IE LOOP	Initiator - Loss Of Offsite Power	Level 1: • LOOP Initiator with non recovery of OSP • CCF of ventilation in Division 1 and 4 and failure to recover room cooling result in failure of ventilation in all SBs • All EFW trains fail on loss of ventilation. PBL fails on loss of Division 4.
			REC OSP 2HR	Failure to Recover Offsite Power Within 2 Hours	
	19-230		QKA10GH001_FS_B-ALL	CCF of the Air Cooled SCWS Chiller Units to Start	
			OPF-SAC-2H	Operator Fails to Recover Room Cooling Locally	
			L2PH CPIHLR-TR,TP=Y	Induced hot leg rupture. Conditional probability given no ISGTR. TR, TRD, TP, TPD cases.	Level 2 • Sequence enters CET1 High Pressure • Induced hot leg rupture depressurizes primary • Sequence enters CET Low Pressure • Containment fails before vessel rupture due to hydrogen flame acceleration • Ex vessel steam explosion at vessel failure leads to melt plug failure • MCCI occurs due to early melt release from pit. • SAHRS sprays fail to control source term due to loss of Division 1 and 4. Offsite power recovery does not play a role since the buses are failed.
			L2PH VECF-FA(H)	Very early containment failure due to H2 Flame Acceleration (Hi pressure sequences)	
			L2 REC=Y OSP 2-7H	Offsite power recovered between 2 and 7 hours	
			L2PH STMEXP EX	Ex-vessel steam explosion damages reactor pit	
			L2PH CCI-EARLYREL=Y	MCCI occurs, following early melt release from pit	

Table 19.1-25—Level 2 Internal Events Large Release Significant Cutsets
Sheet 10 of 14

Release Category	Freq /yr	Contribution to LRF (%)	Event Identifier	Event Description	Sequence of events that lead to CD and to Containment Failure
			~~L2FLCDES-TP~~	~~Level 2 FLAG: TP CDES~~	~~Level 2:~~ • ~~Sequence enters CET1 High Pressure~~ • ~~Induced hot leg rupture depressurizes primary~~ • ~~Sequence enters CET Low Pressure~~ • ~~Containment fails before vessel rupture due to hydrogen flame acceleration~~ • ~~Significant CCI occurs with no system failures~~ • ~~SAHRS sprays fail to control source term due to loss of Division 1 and 4. Offsite power recovery does not play a role since the buses are failed.~~
		19-230 →	~~L2FLCET1 HI PRESSURE~~	~~Level 2 FLAG: CET1 HI PRESSURE~~	
			~~L2PH CPIHLR-TR,TP=Y~~	~~Induced hot leg rupture. Conditional probability given no ISGTR. TR, TRD, TP, TPD cases.~~	
			~~L2FLHLR DEPRESS~~	~~Level 2 FLAG: Depressurization of high CDES by HLR~~	
			~~L2FLCET LO PRESSURE~~	~~Level 2 FLAG: CET LO PRESSURE~~	
			~~L2PH VECF-FA(H)~~	~~Very early containment failure due to H2 Flame Acceleration (Hi pressure sequences)~~	
			~~L2PH CCI~~	~~Level 2 phenomena: significant MCCI, no system failures~~	
			~~L2 REC=Y OSP 2-7H~~	~~Offsite power recovered between 2 and 7 hours~~	
			~~L2FLREC OSP 2-7H~~	~~Level 2 FLAG to mark recovery of OSP in 2-7H~~	

Table 19.1-25—Level 2 Internal Events Large Release Significant Cutsets
Sheet 12 of 14

Release Category	Freq /yr	Contribution to LRF (%)	Event Identifier	Event Description	Sequence of events that lead to CD and to Containment Failure
Internal RC304 -1, 2, 3, 4, 5, 6, 7, 8	8.54E-09	39.3767%	IE SLBI	Initiator - Steam Break Inside Containment	Level 1 and 2: • This family of cutsets includes SLBI Initiator plus failure of I&C signals for MSIV and MFW Isolation of at least 3 SGs • This leads to uncontrolled reactivity event due to overcooling
			SG4 PRES CCF-ALL or SG4 PRES CCF-234 or SG4 PRES CCF-123 or SG4 PRES CCF-134 or SG4 PRES CCF-124 or APU4 CCF NS-ALL or ALU-B CCF NS-ALL or CL-PS-B-SWCCF 19-230	CCF of SG4 level sensors (WR & NR) or CCF of SG4 level sensors (WR & NR)or CCF of SG4 level sensors (WR & NR) or CCF of SG4 level sensors (WR & NR) or CCF of SG4 level sensors (WR & NR) or CCF of APU-4 Protection Sys Computer Processors (Non-Self-Monitored) CCF of ALU-B Protection System Computer Processors (Non-Self-Monitored) Software CCF of Protection System diversity group B	
			L2FLCDES-ATI	Level 2 FLAG: ATI CDES	

Table 19.1-50—Level 2 Flooding Events Release Category Results - LRF
Sheet 1 of 2

19-230

Release Category	Description	Mean	Contribution to LRF	Conditional Containment Failure Probability
RC201	Containment fails before vessel breach due to isolation failure, melt retained in vessel	~~1.2E-11~~1.9E-11	~~1.70%~~1.7%	.0003
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, with containment sprays	1.7E-17	0.0%	0.0
RC203	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, without containment sprays	~~1.3E-13~~1.6E-13	0.0%	0.0
RC204	Containment fails before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel with containment sprays	~~1.3E-14~~1.5E-14	0.0%	0.0
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel without containment sprays	4.1E-11	3.7%	0.0007
RC301	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, with containment sprays	~~4.4E-15~~4.3E-15	0.0%	0.0
RC302	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, without containment sprays	~~2.9E-12~~3.0E-12	0.3%	0.0
RC303	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, with containment sprays	1.1E-11	1.0%	0.0002

Table 19.1-75—Level 2 Fire Events Release Category Results - LRF
Sheet 1 of 2

19-230

Release Category	Description	Mean	Contribution to LRF	Conditional Containment Failure Probability
RC201	Containment fails before vessel breach due to isolation failure, melt retained in vessel	2.9E-11	0.8~~0~~%	0.0002
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, with containment sprays	1.8E-15	0.00%	0.0
RC203	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, without containment sprays	~~1.2E-13~~1.5E-13	0.00%	0.0
RC204	Containment fails before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel with containment sprays	~~4.1E-13~~5.4E-13	~~0.01%~~0.0%	0.0
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel without containment sprays	4.2E-11	~~1.17%~~1.2%	0.0002
RC301	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, with containment sprays	3.4E-13	~~0.01%~~0.0%	0.0
RC302	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, without containment sprays	~~9.2E-12~~1.0E-11	~~0.25%~~0.3%	0.0001
RC303	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, with containment sprays	6.1E-10	~~16.88%~~16.9%	0.0034
RC304	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, without containment sprays	2.3E-09	63.5~~8~~%	0.013

Table 19.1-105—U.S. EPR Release Category Contributions to Total LRF from at Power Internal Events, Fire and Flooding
Sheet 1 of 2

Release Category	Description	Mean	Contribution to LRF	Conditional Containment Failure Probability
RC201	Containment fails before vessel breach due to isolation failure, melt retained in vessel	5.0E-10	1.9%	0.001
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, with containment sprays	4.0E-14	0.0%	0.000
RC203	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex vessel, without containment sprays	~~8.5E-13~~1.9E-12	0.0%	0.000
RC204	Containment fails before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel with containment sprays	~~2.4E-11~~2.8E-11	0.1%	0.000
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex vessel without containment sprays	4.1E-10	1.5%	0.001
RC301	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, with containment sprays	~~1.6E-12~~1.7E-12	0.0%	0.000
RC302	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex vessel, without containment sprays	~~1.5E-11~~2.2E-11	0.1%	0.000
RC303	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, with containment sprays	2.3E-09	8.7%	0.004
RC304	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex vessel, without containment sprays	1.8E-08	~~66.5%~~66.4%	0.033

19-230