Response to
Request for Additional Information No. 6, Revision 0
6/05/2008

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: 19.1
SPLB Branch

Question 19-78:

Severe accident "relevant" accident scenarios are defined as those having a PRA Level 1 CDF greater than a single specified frequency. Please provide information on the rationale for selecting this scenario selection criterion, including a discussion of how these scenarios would bound various scenarios that could have significantly different levels of consequences and public risk factors.

Response to Question 19-78:

The term “Relevant scenario” is used in the AREVA NP severe accident evaluation methodology, described in ANP-10268 (PA), “Severe Accident Evaluations,” which addresses the safety issues associated with severe accident response in nuclear power plants. The methodology was prepared to demonstrate how the design of the U.S. EPR complies with the expectation of SECY-93-087 for issues of hydrogen control, core debris coolability, high pressure core melt ejection, containment performance, and equipment survivability. The methodology is also the basis for the content appearing in the U.S. EPR FSAR Tier 2 Section 19.2. Analyses derived from this methodology are distinct from PRA; nonetheless, PRA (i.e., PRA Level 1) is used to identify relevant scenarios demonstrating U.S. EPR severe accident response features. It should be noted that the AREVA NP PRA methodology does not limit the scope of analysis to “relevant scenarios”.

The first step in preparing the calculation matrix is the identification of the set of “more likely” or relevant scenarios. The expression “more likely,” which comes from the SECY-93-087 report, is interpreted to mean that there exists a threshold of relevance for which certain events or combination of events becomes so unlikely that detailed analytical consideration is unnecessary. A prerequisite of the AREVA NP severe accident evaluation methodology is to identify that threshold so that meaningful analysis can be prepared.

For severe accident analysis, an applicable measure for such a threshold is the core damage frequency (CDF). Such a relevance threshold has been proposed previously (e.g., 1.0E 7/yr appears in EPRI’s Utility Requirements Document). Defining such a threshold is not entirely a subjective task. For modern reactor designs, new safety features have driven the CDF very low. As such, even a threshold of 1.0E-7/yr might exclude so many events that the completeness of a severe accident analysis comes into question. For this reason, relevant scenarios in the U.S. EPR are defined as those having a CDF greater than 1.0E-8/yr. In the AREVA NP scenario identification analysis, this 1.0E-8/yr threshold captured categories of events covering over 90% of the CDF.

The objective of the analyses derived under the AREVA severe accident evaluation methodology is to demonstrate that the plant’s best-estimate response to the more likely severe accidents preserves containment integrity for a minimum of 24 hours, or that containment failure probability from a particular event is very low, thus significantly limiting public exposure and risk. This conclusion is drawn from a large suite of deterministic severe accident simulations using the MAAP4.0.7 computer code. The AREVA NP severe accident evaluation methodology includes an uncertainty analysis that addresses various uncertainties associated with the response of the U.S. EPR’s severe accident response features. These were identified during the RAI process for ANP-10268. Consequences of varying levels of severity up to containment integrity are addressed within this uncertainty analysis. The subsequent results are bounded by an estimate of a particular tolerance limit (“95/95”) with the considered uncertainty domain. The nature of certain events, such as steam explosion and high pressure melt ejection, include inherently large uncertainties. As such, a suite of phenomenological studies, performed as part of PRA Level 2, are evaluated to determine the probability of containment failure or bypass from these events. As with the uncertainty analysis supporting containment integrity, these phenomenological studies statistically convolve several uncertainties to derive a bounding containment failure probability. The results from these analyses are considered in the evaluation PRA Level 2 and Level 3 to quantify consequence and risk to the public and they are also credited as a supplement to the analyses addressing SECY-93-087.

For the U.S. EPR, the principle objective of the deterministic analyses is to demonstrate that the integrity of the containment is maintained for a minimum of 24 hours. This condition is demonstrated using deterministic analyses for the issues of combustible gas control, core debris coolability, containment overpressurization, high pressure melt ejection (partially), and equipment survivability (containment failure probabilities were evaluated for steam explosion and high pressure melt ejection). As such, public risk in these events is considered negligible. In the same way in which design basis events were identified as precursors for relevant severe accident scenarios, events with notable public risk identified in PRA Level 3 are drawn from the same set of initiating events considered to derive at the relevant scenarios. The difference is that additional failures must be assumed to result in serious consequences for the public. When convolving the failure probabilities associated with a relevant scenario with the probabilities of the additional failures, the likelihood of significant consequences to the public is so small that they do not qualify as a “more likely” or “relevant” scenario. As such, the best measure for defining the threshold for the relevant scenarios is the CDF.

FSAR Impact:

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

Question 19-79:

Please provide the description, technical basis, and decomposition of phenomena pertaining to induced rupture of the reactor system pressure boundary during severe accidents.

- Provide the analyses for hot leg and surge line rupture. Include base-case analyses and sensitivity studies. Discuss the consequences of hot leg failure, the impact of material properties, the sensitivity of phenomena to natural circulation flow rates, the impacts of different initiators, and the effects of small breaks or seal leaks. Also identify the representative transients.
- Provide the analyses for steam generator tube rupture. Include base case analyses and sensitivity studies. Discuss the impact of degraded tubes, from either wear at anti-vibration bars, wear from foreign objects above the tube support plate, or from stress corrosion cracking (note that the tubes will be fabricated from Inconel 690). Include considerations of pressure-induced rupture upon secondary side depressurization prior to core damage, and high temperature-induced creep rupture after core damage.
- Provide a summary of the relevant MAAP cases and key results, including failure times and/or damage fractions, plots of hot leg and steam generator tube temperatures, plots of natural circulation flow rates (including core-to-upper plenum, countercurrent and unidirectional flow through the hot legs, and countercurrent and unidirectional flow through the steam generator tubes).
- What are the total probabilities of induced primary system rupture for the spectrum of break sizes considered?
- Describe how the core damage end states were defined. Show the methodology used for applying these to the Level 1 core damage sequences. For the core damage end state TP, please explain the basis for calculating the times to hot leg or surge line rupture, steam generator tube rupture, and vessel failure.

Response to Question 19-79:

A response to this question will be provided by August 8, 2008.

Question 19-80:

Please discuss any modifications to the MAAP 4 code that are associated with the core catcher, the heavy reflector, or the severe accident depressurization valves, that were not discussed in the topical report ANP-10268P.

Response to Question 19-80:

The modifications to MAAP4 for the core catcher and heavy reflector were described in the AREVA NP Severe Accident Topical Report ANP-10268PA. There were no modifications specifically for the severe accident depressurization valves. There was a modification to allow the pressurizer relief tank (PRT) to discharge into two separate compartments in parallel. This modification is also described in ANP-10268PA.

FSAR Impact:

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

Question 19-81:

For the Level 2 phenomenological basic events (i.e., those events with identifiers "L2PH…"), please supply the associated discrete uncertainty probability distributions and the supporting bases for these values. Also supply the CET-supporting fault trees that utilize these events. Examples of such basic events would be MCCI probabilities in various circumstances, containment overpressure failure probability during debris quench, in-vessel and ex-vessel steam explosion consequences, and "rocket" mode RPV failure. In addition, please provide the numerical values for any other branch probabilities of events in the containment event trees documented in Appendix 19C of the FSAR that are not covered by the above request.

Response to Question 19-81:

In general, Level 2 phenomenological events follow double-delta uncertainty distributions. The true value of the basic events is either 1 or 0, but it is uncertain, given the present state of knowledge of the event phenomenology.

The implementation of these uncertainty distributions is complicated by some limitations of the Risk Spectrum® software. The data has to be entered as cumulative functions. Also, the software does not allow two consecutive points to have the exact same cumulative probability value. Therefore, to represent the functions correctly, the data was entered using four points, as follows:

- (Point 1) Value = 0.0, Cumulative probability = 0.
- (Point 2) Value = 1E10, Cumulative probability = success probability for the event.
- (Point 3) Value = 1 – 1E-10, Cumulative probability = success probability for the event + 1E-7.
- (Point 4) Value = 1.0, Cumulative probability = 1.0.

The above approach leads to two peaks of probability on the uncertainty distribution, centered at P = 0 and P = 1.

Table 19-81-1 shows the discrete distribution functions used in quantifying the Risk Spectrum® model. These were calculated according to the previous mentioned method, using a spreadsheet.

- The mean value in the table is the failure probability of the basic event.
- The X value is the value of the event.
- The Y value is the cumulative probability (discrete distribution).

The CET-supporting fault trees that utilize these phenomenological events are shown in Figure 19-81-1 Sheets 1 thru 19.

Table 19-81-2 provides the numerical values for any other branch probabilities of events in the containment event trees (documented in Tier2 Appendix 19C of the FSAR).

FSAR Impact:

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

Table 19-81-1—Discrete Probability Distributions for Phenomenological Level 2 Basic Events

ID	Mean	X	Y
L2PH CBV HP	0.08	0.0000000000	0.0000000000
		0.0000000001	0.9200000000
		0.9999999999	0.9200001000
		1.0000000000	1.0000000000
L2PH CCI	0.001	0.0000000000	0.0000000000
		0.0000000001	0.9990000000
		0.9999999999	0.9990001000
		1.0000000000	1.0000000000
L2PH CF QUENCH SPIKE	0.000003	0.0000000000	0.0000000000
		0.0000000001	0.9999970000
		0.9999999999	0.9999971000
		1.0000000000	1.0000000000
L2PH CP-PITF-VF(CBV)	0.02	0.0000000000	0.0000000000
		0.0000000001	0.9800000000
		0.9999999999	0.9800001000
		1.0000000000	1.0000000000
L2PH CPIHLR-TR,TP=N	0.05	0.0000000000	0.0000000000
		0.0000000001	0.9500000000
		0.9999999999	0.9500001000
		1.0000000000	1.0000000000
L2PH CPIHLR-TR,TP=Y	0.95	0.0000000000	0.0000000000
		0.0000000001	0.0500000000
		0.9999999999	0.0500001000
		1.0000000000	1.0000000000
L2PH EARLYCF DCH(HP)	0.00055	0.0000000000	0.0000000000
		0.0000000001	0.9994500000
		0.9999999999	0.9994501000
		1.0000000000	1.0000000000
L2PH EARLYCF FA(HP)	0.0056	0.0000000000	0.0000000000
		0.0000000001	0.9944000000
		0.9999999999	0.9944001000
		1.0000000000	1.0000000000

Table 19-81-1—Discrete Probability Distributions for Phenomenological Level 2 Basic Events

ID	Mean	X	Y
L2PH INC MELT TR	0.01	0.0000000000	0.0000000000
		0.0000000001	0.9900000000
		0.9999999999	0.9900001000
		1.0000000000	1.0000000000
L2PH INC MELT TR HLR	0.5	0.0000000000	0.0000000000
		0.0000000001	0.5000000000
		0.9999999999	0.5000001000
		1.0000000000	1.0000000000
L2PH INVREC(LOOP)=N	0.5	0.0000000000	0.0000000000
		0.0000000001	0.5000000000
		0.9999999999	0.5000001000
		1.0000000000	1.0000000000
L2PH INVREC(LOOP)=Y	0.5	0.0000000000	0.0000000000
		0.0000000001	0.5000000000
		0.9999999999	0.5000001000
		1.0000000000	1.0000000000
L2PH INVREC(S-DEP)=N	0.1	0.0000000000	0.0000000000
		0.0000000001	0.9000000000
		0.9999999999	0.9000001000
		1.0000000000	1.0000000000
L2PH INVREC(S-DEP)=Y	0.9	0.0000000000	0.0000000000
		0.0000000001	0.1000000000
		0.9999999999	0.1000001000
		1.0000000000	1.0000000000
L2PH INVREC(T-DEP)=N	0.3	0.0000000000	0.0000000000
		0.0000000001	0.7000000000
		0.9999999999	0.7000001000
		1.0000000000	1.0000000000
L2PH INVREC(T-DEP)=Y	0.7	0.0000000000	0.0000000000
		0.0000000001	0.3000000000
		0.9999999999	0.3000001000
		1.0000000000	1.0000000000

Table 19-81-1—Discrete Probability Distributions for Phenomenological Level 2 Basic Events

ID	Mean	X	Y
L2PH ISGTR-SS0.6D=N	0.714	0.0000000000	0.0000000000
		0.0000000001	0.2860000000
		0.9999999999	0.2860001000
		1.0000000000	1.0000000000
L2PH ISGTR-SS0.6D=Y	0.286	0.0000000000	0.0000000000
		0.0000000001	0.7140000000
		0.9999999999	0.7140001000
		1.0000000000	1.0000000000
L2PH ISGTR-SS2D=N	0.16	0.0000000000	0.0000000000
		0.0000000001	0.8400000000
		0.9999999999	0.8400001000
		1.0000000000	1.0000000000
L2PH ISGTR-SS2D=Y	0.84	0.0000000000	0.0000000000
		0.0000000001	0.1600000000
		0.9999999999	0.1600001000
		1.0000000000	1.0000000000
L2PH ISGTR-TRD=Y	0.0004	0.0000000000	0.0000000000
		0.0000000001	0.9996000000
		0.9999999999	0.9996001000
		1.0000000000	1.0000000000
L2PH LATE FA MCCI+DR	0.0005	0.0000000000	0.0000000000
		0.0000000001	0.9995000000
		0.9999999999	0.9995001000
		1.0000000000	1.0000000000
L2PH LATE-CCI-CF=BMT	0.98	0.0000000000	0.0000000000
		0.0000000001	0.0200000000
		0.9999999999	0.0200001000
		1.0000000000	1.0000000000
L2PH LATE-CCI-CF=OP	0.01	0.0000000000	0.0000000000
		0.0000000001	0.9900000000
		0.9999999999	0.9900001000
		1.0000000000	1.0000000000

Table 19-81-1—Discrete Probability Distributions for Phenomenological Level 2 Basic Events

ID	Mean	X	Y
L2PH LATE-CF-FA	0.00045	0.0000000000	0.0000000000
		0.0000000001	0.9995500000
		0.9999999999	0.9995501000
		1.0000000000	1.0000000000
L2PH LATECF-FA(MCCI)	0.0001	0.0000000000	0.0000000000
		0.0000000001	0.9999000000
		0.9999999999	0.9999001000
		1.0000000000	1.0000000000
L2PH PITF-VF(NO-CBV)	0.00000207	0.0000000000	0.0000000000
		0.0000000001	0.9999979300
		0.9999999999	0.9999980300
		1.0000000000	1.0000000000
L2PH RECDAMAGE(I)	0.016	0.0000000000	0.0000000000
		0.0000000001	0.9840000000
		0.9999999999	0.9840001000
		1.0000000000	1.0000000000
L2PH RECDAMAGE(II)	0.00125	0.0000000000	0.0000000000
		0.0000000001	0.9987500000
		0.9999999999	0.9987501000
		1.0000000000	1.0000000000
L2PH RECDAMAGE(III)	0.00045	0.0000000000	0.0000000000
		0.0000000001	0.9995500000
		0.9999999999	0.9995501000
		1.0000000000	1.0000000000
L2PH STM EXP EXV	0.0000255	0.0000000000	0.0000000000
		0.0000000001	0.9999745000
		0.9999999999	0.9999746000
		1.0000000000	1.0000000000
L2PH STM EXP INV HP	0.0000229	0.0000000000	0.0000000000
		0.0000000001	0.9999771000
		0.9999999999	0.9999772000
		1.0000000000	1.0000000000

Table 19-81-1—Discrete Probability Distributions for Phenomenological Level 2 Basic Events

ID	Mean	X	Y
L2PH STM EXP INV LP	0.00000562	0.0000000000	0.0000000000
		0.0000000001	0.9999943800
		0.9999999999	0.9999944800
		1.0000000000	1.0000000000
L2PH STM EXP LH HP	0.000863	0.0000000000	0.0000000000
		0.0000000001	0.9991370000
		0.9999999999	0.9991371000
		1.0000000000	1.0000000000
L2PH STM EXP LH LP	0.0000253	0.0000000000	0.0000000000
		0.0000000001	0.9999747000
		0.9999999999	0.9999748000
		1.0000000000	1.0000000000
L2PH TF3-MCCI-CF=N	0.01	0.0000000000	0.0000000000
		0.0000000001	0.9900000000
		0.9999999999	0.9900001000
		1.0000000000	1.0000000000
L2PH VEACF-H2DEF(HL)	0.000138	0.0000000000	0.0000000000
		0.0000000001	0.9998620000
		0.9999999999	0.9998621000
		1.0000000000	1.0000000000
L2PH VECF-FA(H)	0.016	0.0000000000	0.0000000000
		0.0000000001	0.9840000000
		0.9999999999	0.9840001000
		1.0000000000	1.0000000000
L2PH VECF-FA(HL)	0.00125	0.0000000000	0.0000000000
		0.0000000001	0.9987500000
		0.9999999999	0.9987501000
		1.0000000000	1.0000000000
L2PH VECF-H2DEF(H)	0.000002	0.0000000000	0.0000000000
		0.0000000001	0.9999980000
		0.9999999999	0.9999981000
		1.0000000000	1.0000000000

Table 19-81-2—Other CET Branch Probabilities

Basic Event	Description	Mean value
L2 REC OSP 2-7H	Offsite power not recovered between 2 and 7 hours	3.21E-01
L2 REC OSP 7-31H	Offsite power not recovered between 7 and 31 hours	3.04E-01
L2 REC=Y OSP 2-7H	Offsite power recovered between 2 and 7 hours	6.79E-01
L2 REC=Y OSP 7-31H	Offsite power recovered between 7 and 31 hours	6.96E-01
L2CP ISL BL NO WATER	Level 2 conditional probability: break location not under water (ISL)	1.00E+00
L2CP ISL BL WATER	Level 2 conditional probability: break location not under water (ISL)	0.00E+00
L2CP SL0.6"DIAM	Level 2 conditional probability: Small LOCA has 0.6" diameter	5.00E-01
L2CP SL2"DIAM	Level 2 conditional probability: Small LOCA has 2" diameter	5.00E-01
L2CP SS0.6"DIAM	Level 2 conditional probability: Seal LOCA has 0.6" diameter	5.00E-01
L2CP SS2"DIAM	Level 2 conditional probability: Seal LOCA has 2" diameter	5.00E-01
L2FLCDES-AT	Level 2 FLAG: AT CDES	1.00E+00
L2FLCDES-ATI	Level 2 FLAG: ATI CDES	1.00E+00
L2FLCDES-IS	Level 2 FLAG: IS CDES	1.00E+00
L2FLCDES-LL	Level 2 FLAG: LL CDES	1.00E+00
L2FLCDES-LL1	Level 2 FLAG: LL1 CDES	1.00E+00
L2FLCDES-ML	Level 2 FLAG: ML CDES	1.00E+00
L2FLCDES-PL	Level 2 FLAG: PL CDES	1.00E+00
L2FLCDES-RV	Level 2 FLAG: RV CDES	1.00E+00
L2FLCDES-SG	Level 2 FLAG: SG CDES	1.00E+00
L2FLCDES-SG1	Level 2 FLAG: SG1 CDES	1.00E+00
L2FLCDES-SG2	Level 2 FLAG: SG1 CDES	1.00E+00
L2FLCDES-SG3	Level 2 FLAG: SG3 CDES	1.00E+00
L2FLCDES-SL	Level 2 FLAG: SL CDES	1.00E+00
L2FLCDES-SL1	Level 2 FLAG: SL1 CDES	1.00E+00
L2FLCDES-SL1D	Level 2 FLAG: SL1D CDES	1.00E+00
L2FLCDES-SLD	Level 2 FLAG: SLD CDES	1.00E+00
L2FLCDES-SP	Level 2 FLAG: SP CDES	1.00E+00
L2FLCDES-SP1	Level 2 FLAG: SP1 CDES	1.00E+00
L2FLCDES-SP1D	Level 2 FLAG: SP1D CDES	1.00E+00
L2FLCDES-SPD	Level 2 FLAG: SPD CDES	1.00E+00
L2FLCDES-SS	Level 2 FLAG: SS CDES	1.00E+00
L2FLCDES-SS1	Level 2 FLAG: SS1 CDES	1.00E+00
L2FLCDES-SS1D	Level 2 FLAG: SS1D CDES	1.00E+00

Table 19-81-2—Other CET Branch Probabilities

Basic Event	Description	Mean value
L2FLCDES-SSD	Level 2 FLAG: SSD CDES	1.00E+00
L2FLCDES-TP	Level 2 FLAG: TP CDES	1.00E+00
L2FLCDES-TP1	Level 2 FLAG: TP1 CDES	1.00E+00
L2FLCDES-TR	Level 2 FLAG: TR CDES	1.00E+00
L2FLCDES-TR1	Level 2 FLAG: TR1 CDES	1.00E+00
L2FLCDES-TR1D	Level 2 FLAG: TR1D CDES	1.00E+00
L2FLCDES-TRD	Level 2 FLAG: TRD CDES	1.00E+00
L2FLCET CF	Level 2 FLAG: CET CF	1.00E+00
L2FLCET ISL	Level 2 FLAG: CET ISL	1.00E+00
L2FLCET LIMITED CD	Level 2 FLAG: CET LIMITED CD	1.00E+00
L2FLCET LO PRESSURE	Level 2 FLAG: CET LO PRESSURE	1.00E+00
L2FLCET SGTR	Level 2 FLAG: CET SGTR	1.00E+00
L2FLCET SGTR FW	Level 2 FLAG: CET SGTR FW	1.00E+00
L2FLCET1 HI PRESSURE	Level 2 FLAG: CET1 HI PRESSURE	1.00E+00
L2FLCET2 HI PRESSURE	Level 2 FLAG: CET2 HI PRESSURE	1.00E+00
L2FLDELETE	Delete cutsets on these sequences. Dummy event.	0.00E+00
L2FLDUMMY	Level 2 FLAG: Dummy Flag	1.00E+00
L2FLHLR DEPRESS	Level 2 FLAG: Depressurization of high CDES by HLR	1.00E+00
L2FLNATDEPRESS	Level 2 FLAG: Depressurization of high CDES by natural depressurization of small LOCA	1.00E+00
L2FLOP DEPRESS	Level 2 FLAG: Depressurization of high CDES by operator	1.00E+00
L2FLREC OSP 2-7H	Level 2 FLAG to mark recovery of OSP in 2-7H	1.00E+00
L2FLREC OSP=NO	Level 2 FLAG to mark NO recovery of OSP	1.00E+00
L2FLREC OSP7-31H	Level 2 FLAG to mark recovery of OSP in 7-31H	1.00E+00
L2PH CBV HP	Complete circumferential rupture of vessel (gives vessel rocket in HP sequences)	8.00E-02
L2PH CCI	Level 2 phenomena: significant MCCI, no system failures	1.00E-03
L2PH CCI-DRY	Significant MCCI occurs, debris not flooded. P = 1.0	1.00E+00
L2PH CCI-EARLYREL=N	MCCI does not occur following early release from pit	0.00E+00
L2PH CCI-EARLYREL=Y	Level 2 phenom: MCCI occurs, following early melt release from pit.	1.00E+00

Table 19-81-2—Other CET Branch Probabilities

Basic Event	Description	Mean value
L2PH CF QUENCH SPIKE	Containment fails due to ovepressure during debris quench	2.98E-06
L2PH CP-PITF-VF(CBV)	Pit overpressure at high pressure vessel failure fails melt plug given CBV occurs	2.00E-02
L2PH CPIHLR-SS,SL=N	No induced hot leg rupture. Conditional probability given no ISGTR. SS, SL, TR/P-D cases	0.00E+00
L2PH CPIHLR-SS,SL=Y	Induced hot leg rupture. Conditional probability, given no SGTR. SS,SL cases.	1.00E+00
L2PH CPIHLR-TR,TP=N	No induced hot leg rupture. Conditional probability given no ISGTR. TP, TR cases (sec not D)	5.00E-02
L2PH CPIHLR-TR,TP=Y	Induced hot leg rupture. Conditional probability given no ISGTR. TR, TRD, TP, TPD cases.	9.50E-01
L2PH EARLYCF DCH(HP)	Containment fails due to DCH loads at vessel failure	5.50E-04
L2PH EARLYCF FA(HP)	Loads from accelerated flame at vessel failure fails containment. High pressure case.	5.60E-03
L2PH EARLYCF FA(LP)	Loads from accelerated flame at vessel failure fails containment. Low P CDES.	0.00E+00
L2PH INC MELT TR	Incomplete melt transfer occurs - not HLR case	1.00E-02
L2PH INC MELT TR HLR	Incomplete melt transfer occurs following induced hot leg rupture	5.00E-01
L2PH INVREC(LOOP)=N	In-vessel recovery, phenomenological failure given sufficient injection. LOOP	5.00E-01
L2PH INVREC(LOOP)=Y	In-vessel recovery, phenomenological success given sufficient injection. LOOP	5.00E-01
L2PH INVREC(NR)=N	In vessel recovery phenomenological failure. Default, non-recoverable cases	1.00E+00
L2PH INVREC(S-DEP)=N	In-vessel recovery fails - hot leg Rupture or operator depressurization during seal/small LOCA DES	1.00E-01
L2PH INVREC(S-DEP)=Y	In-vessel recovery success - hot leg rupture or operator depressurization during seal/small LOCA DES	9.00E-01
L2PH INVREC(T-DEP)=N	In-vessel recovery fails - hot leg Rupture or operator depressurization during transient CDES	3.00E-01
L2PH INVREC(T-DEP)=Y	In-vessel recovery success - hot leg rupture or operator depressurization during transient CDES	7.00E-01
L2PH ISGTR-SS,SL=N	No ISGTR in SL, SS cases with secondary pressurized	1.00E+00
L2PH ISGTR-SS,SL=Y	Induced SGTR occurs. Any seal LOCA or small LOCA	0.00E+00

Table 19-81-2—Other CET Branch Probabilities

Basic Event	Description	Mean value
L2PH ISGTR-SS0.6D=N	No induced SGTR occurs. 0.6" LOCAs, secondary side depressurized	7.14E-01
L2PH ISGTR-SS0.6D=Y	Induced SGTR occurs. 0.6" LOCAs, secondary side depressurized	2.86E-01
L2PH ISGTR-SS2D=N	No induced SGTR. 2" LOCA, secondary depressurized	1.60E-01
L2PH ISGTR-SS2D=Y	Induced SGTR. 2" LOCA, secondary depressurized	8.40E-01
L2PH ISGTR-TR=Y	Induced SGTR. Transients, secondary not depressurized	0.00E+00
L2PH ISGTR-TRD=N	No induced SGTR. Transients with secondary depressurized	1.00E+00
L2PH ISGTR-TRD=Y	Induced SGTR. Transient, secondary depressurized.	4.00E-04
L2PH LATE FA MCCI+DR	Accelerated flame fails containment after VF. Extensive MCCI and early damage to recombiners	5.00E-04
L2PH LATE-CCI-CF=BMT	Level 2 phenomena. MCCI causes late basemat failure.	9.80E-01
L2PH LATE-CCI-CF=N	Level 2 phenomena. MCCI does not cause late basemat failure	1.00E-02
L2PH LATE-CCI-CF=OP	Level 2 phenomena. Late CF due to steam overpressure.	1.00E-02
L2PH LATE-CF-FA	Accelerated flame after vessel failure fails containment. Case (iv) Low pressure, no long term MCCI.	4.50E-04
L2PH LATECF-FA(MCCI)	Accelerated flame fails containment after VF. Extensive MCC	Q
L2PH LOCA-DEPRESS=N	Level 2 phenomena. Small LOCA remains at high pressure.	1.00E+00
L2PH LOCA-DEPRESS=Y	Level 2 phenomena. Small LOCA depressurizes before vessel failure.	0.00E+00
L2PH PITF-VF(NO-CBV)	Level 2 phenomena. Pit overpressure failure (not CBV case)	2.09E-06
L2PH RECDAMAGE(I)	Level 2 phenomena. Recombiners damaged by accelerated flame. Case I.	1.60E-02
L2PH RECDAMAGE(II)	Level 2 phenomena. Recombiners damaged by accelerated flame. Case II.	1.25E-03
L2PH RECDAMAGE(III)	Level 2 phenomena. Recombiners damaged by accelerated flame. Case III.	4.50E-04
L2PH STM EXP EXV	Level 2 phenomena: steam explosion ex-vessel damages pit. General event all CDES.	2.55E-05

Table 19-81-2—Other CET Branch Probabilities

Basic Event	Description	Mean value
L2PH STM EXP INV HP	Level 2 phenomena: containment failure due to in-vessel steam explosion. High pressure CET sequences	2.29E-05
L2PH STM EXP INV LP	Level 2 phenomena: containment failure due to in-vessel steam explosion. Low pressure CET sequences.	5.60E-06
L2PH STM EXP LH HP	Level 2 phenomena: reactor pit damage after in-vessel steam explosion damage to lowr head. Hi P CDES	8.63E-04
L2PH STM EXP LH LP	Level 2 phenomena: reactor pit damage after in-vessel steam explosion damage to lower head. Low CDES.	2.53E-05
L2PH VECF-FA(H)	Very early containment failure due to H2 Flame Acceleration (Hi pressure sequences)	1.60E-02
L2PH VECF-FA(HL)	Very early flame acceleration loads fail containment following induced Hot Leg Rupture	1.25E-03
L2PH VECF-H2DEF(H)	V early CF due to hydrogen deflagration. High pressure CDES, in-vessel - PRV cycling phase	2.03E-06
L2PH VECF-H2DEF(HL)	V Early CF due to hydrogen deflagration. High pressure CDES with Induced Hot Leg Rupture	1.38E-04

Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 1 of 19)

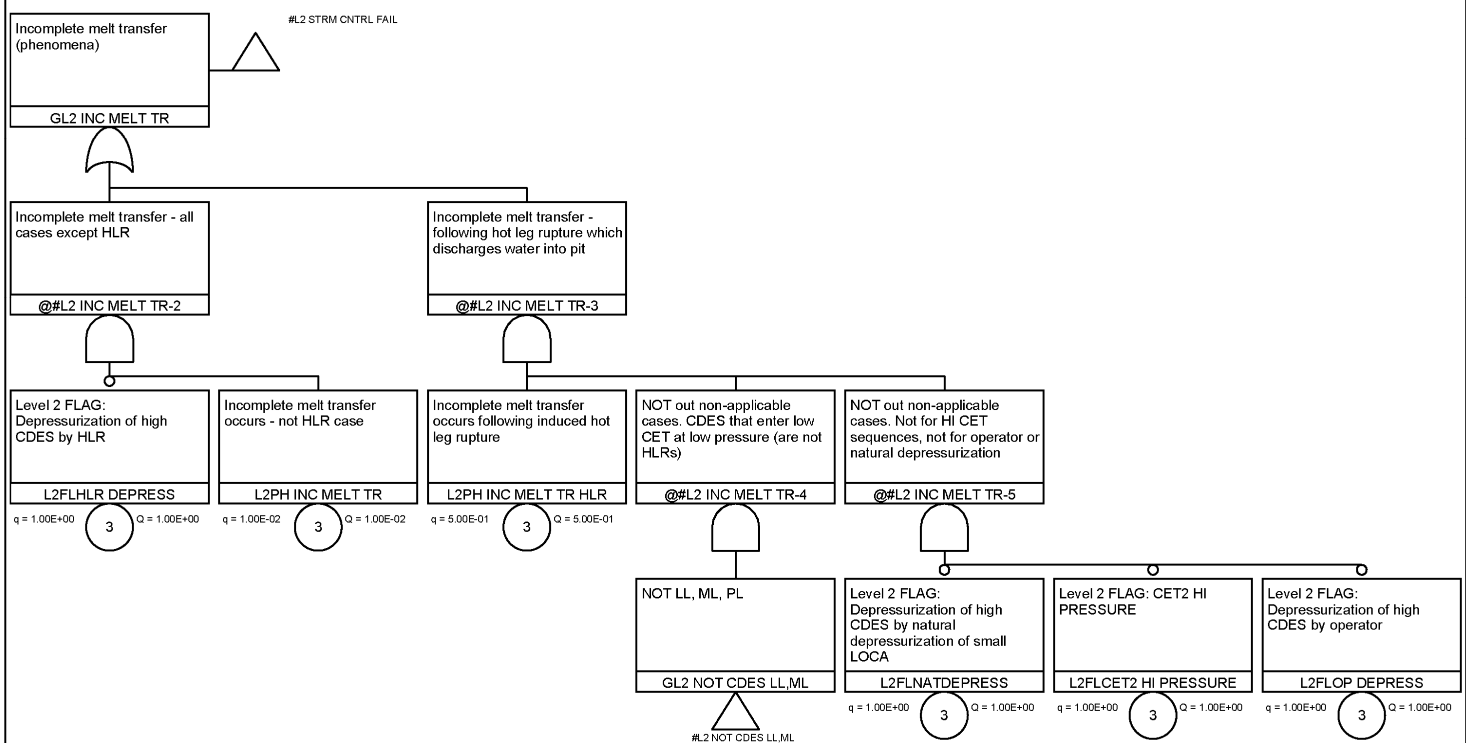


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 2 of 19)

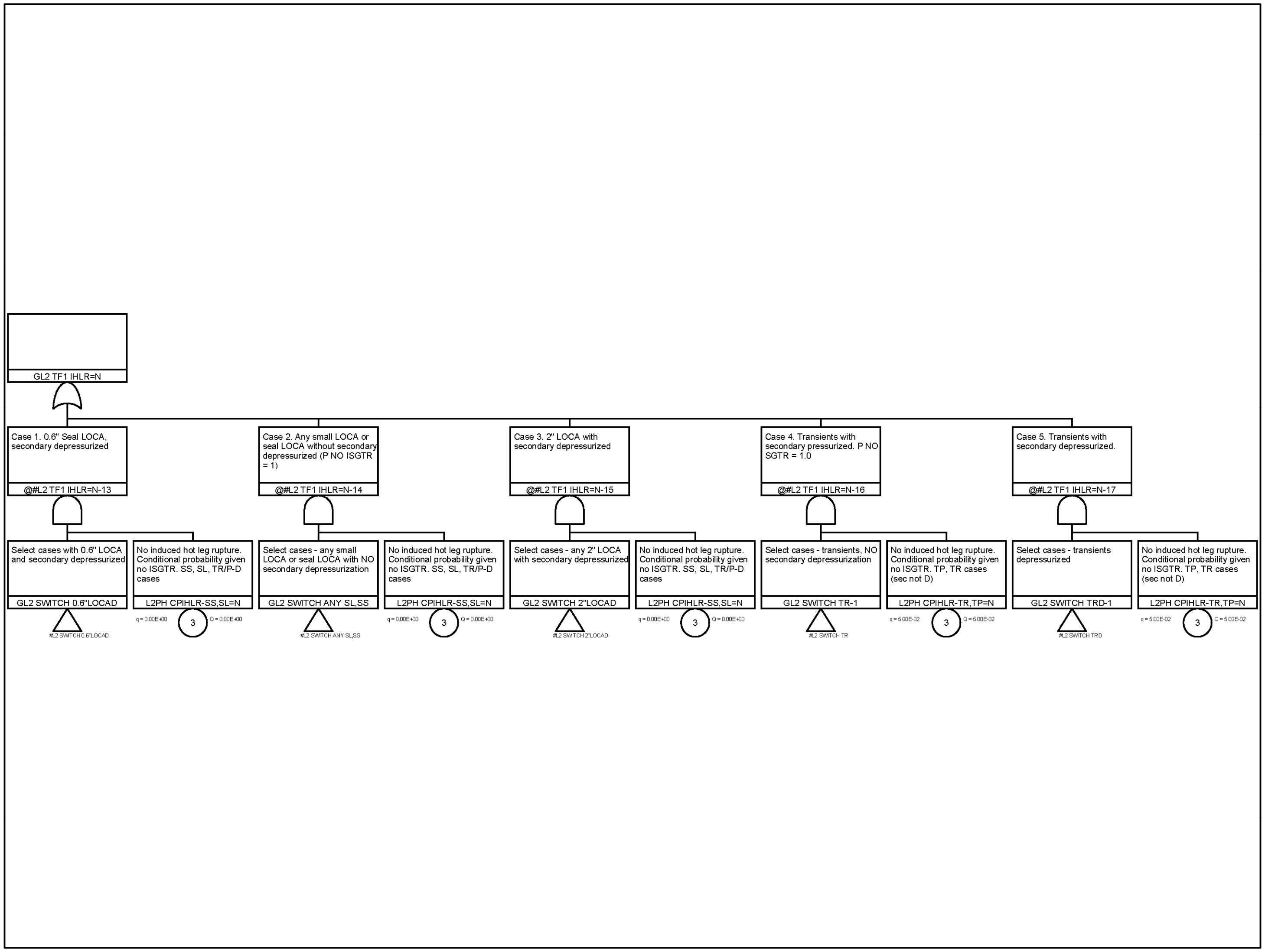


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 3 of 19)

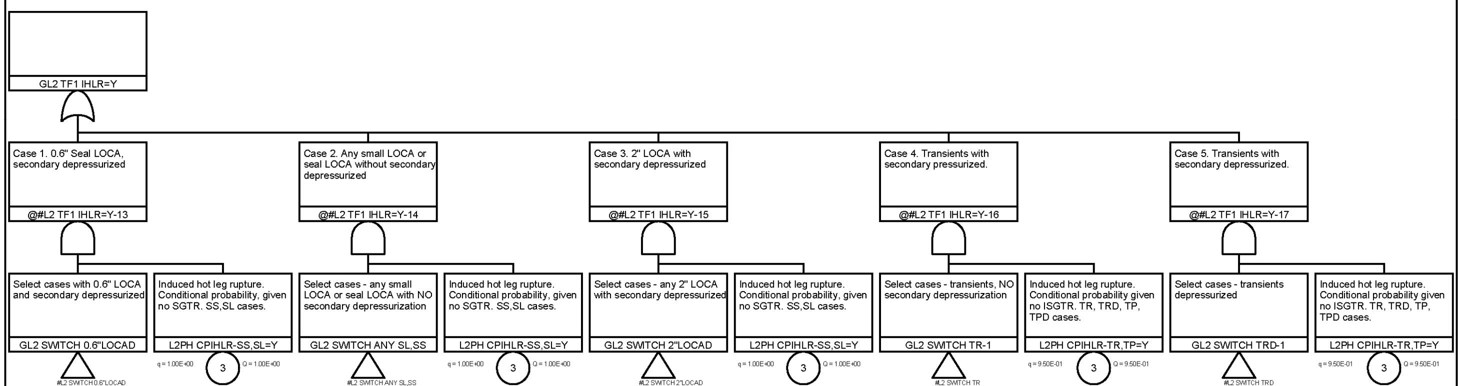


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 4 of 19)

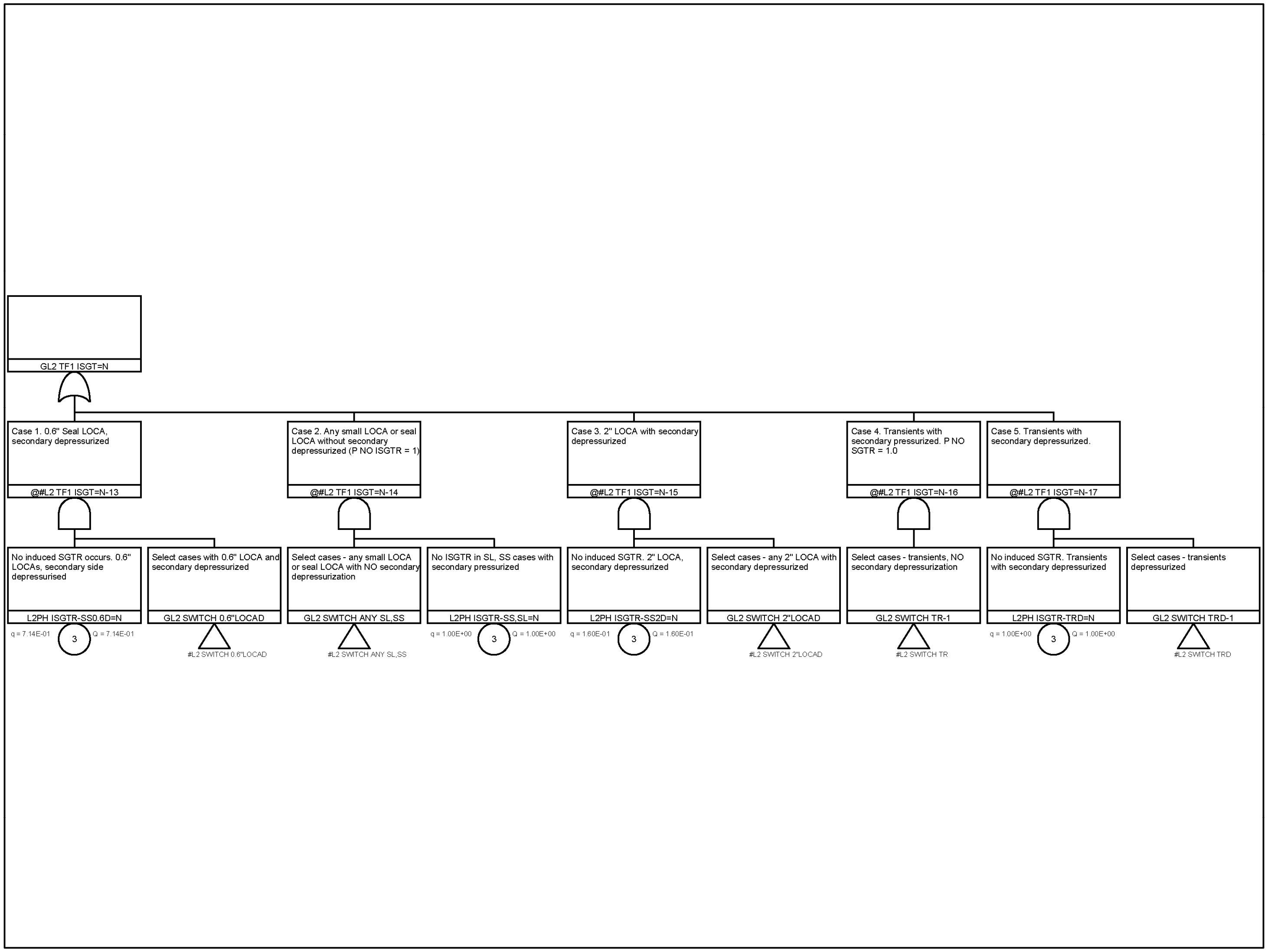


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 5 of 19)

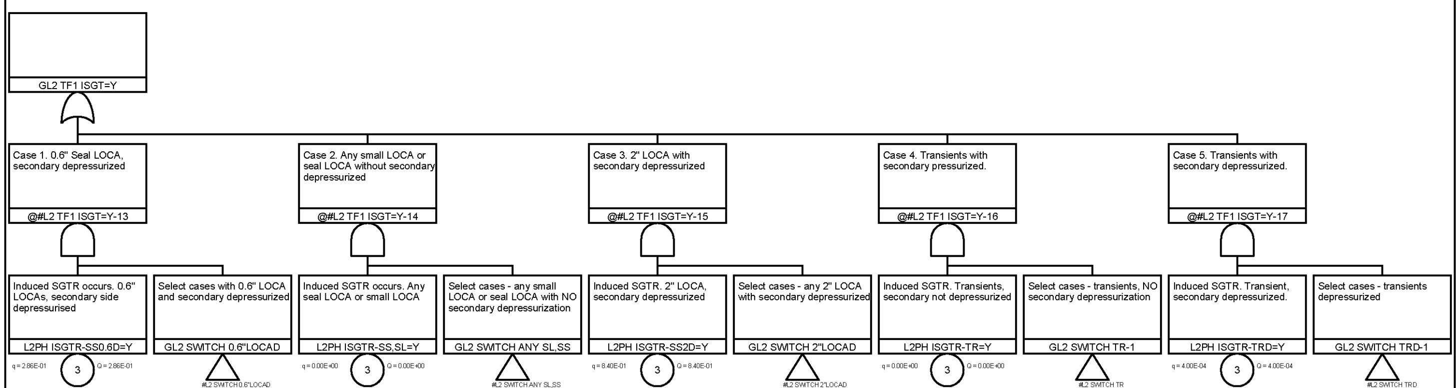


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 6 of 19)

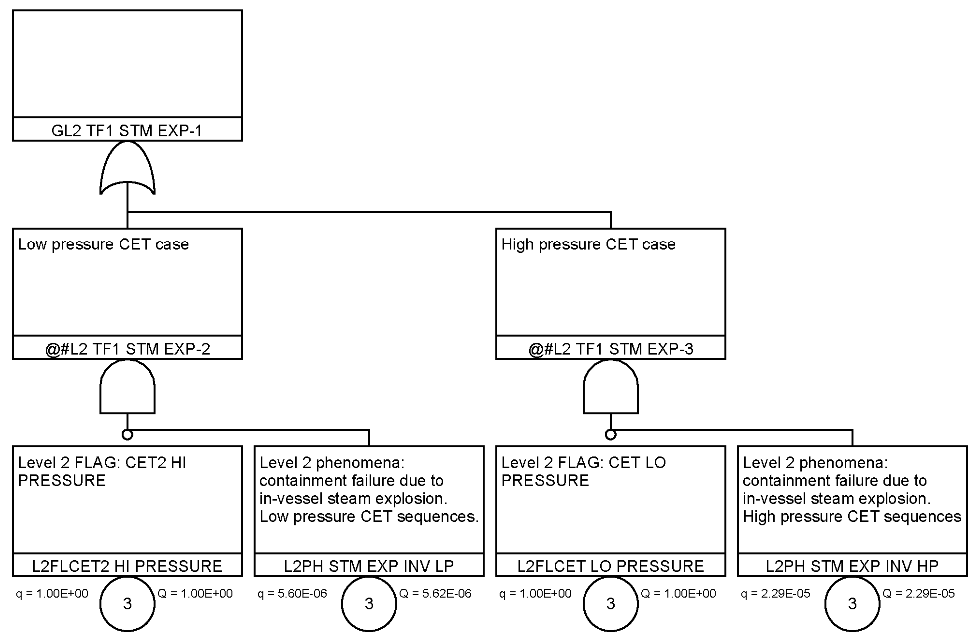


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 7 of 19)

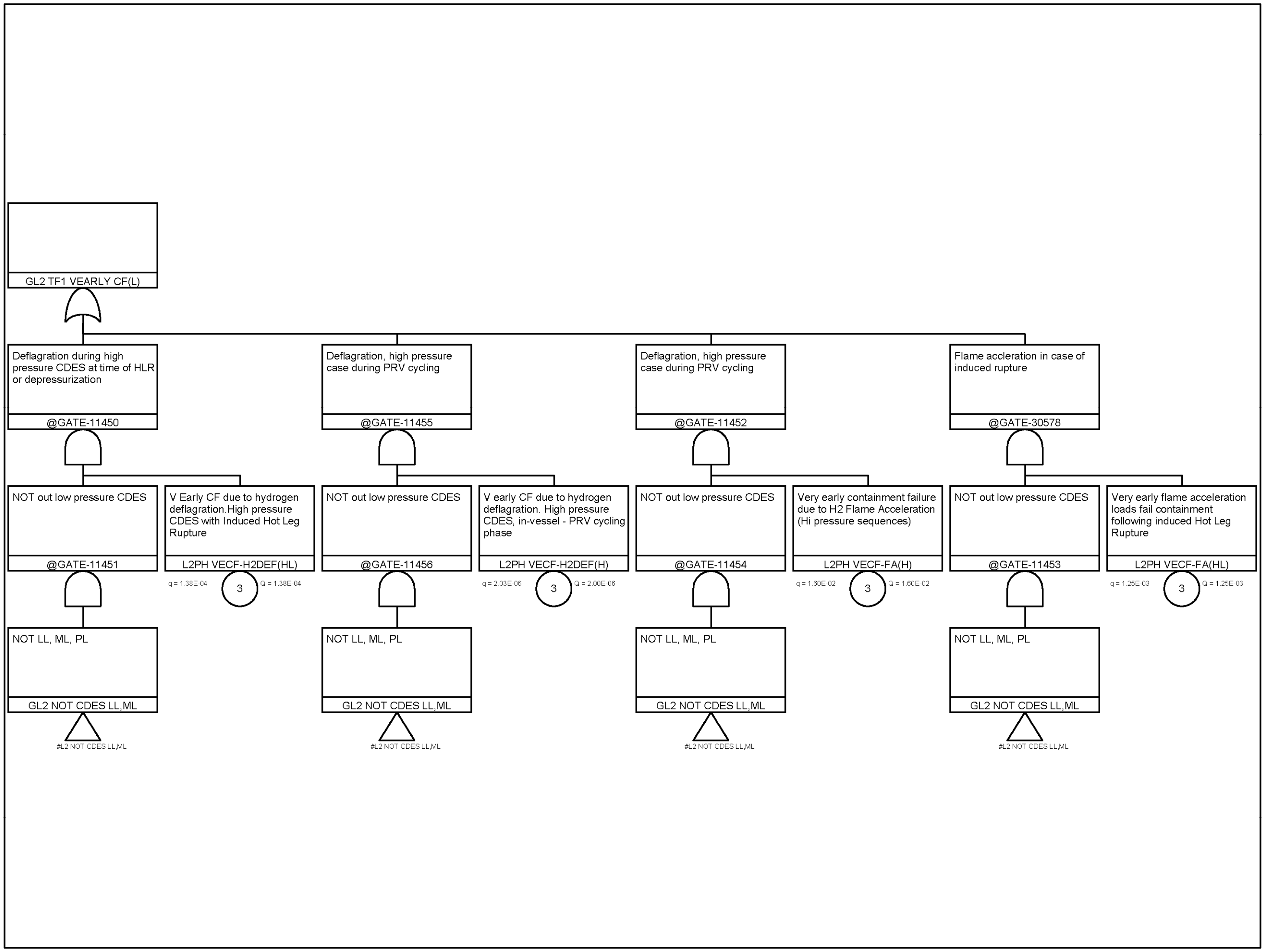


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 8 of 19)

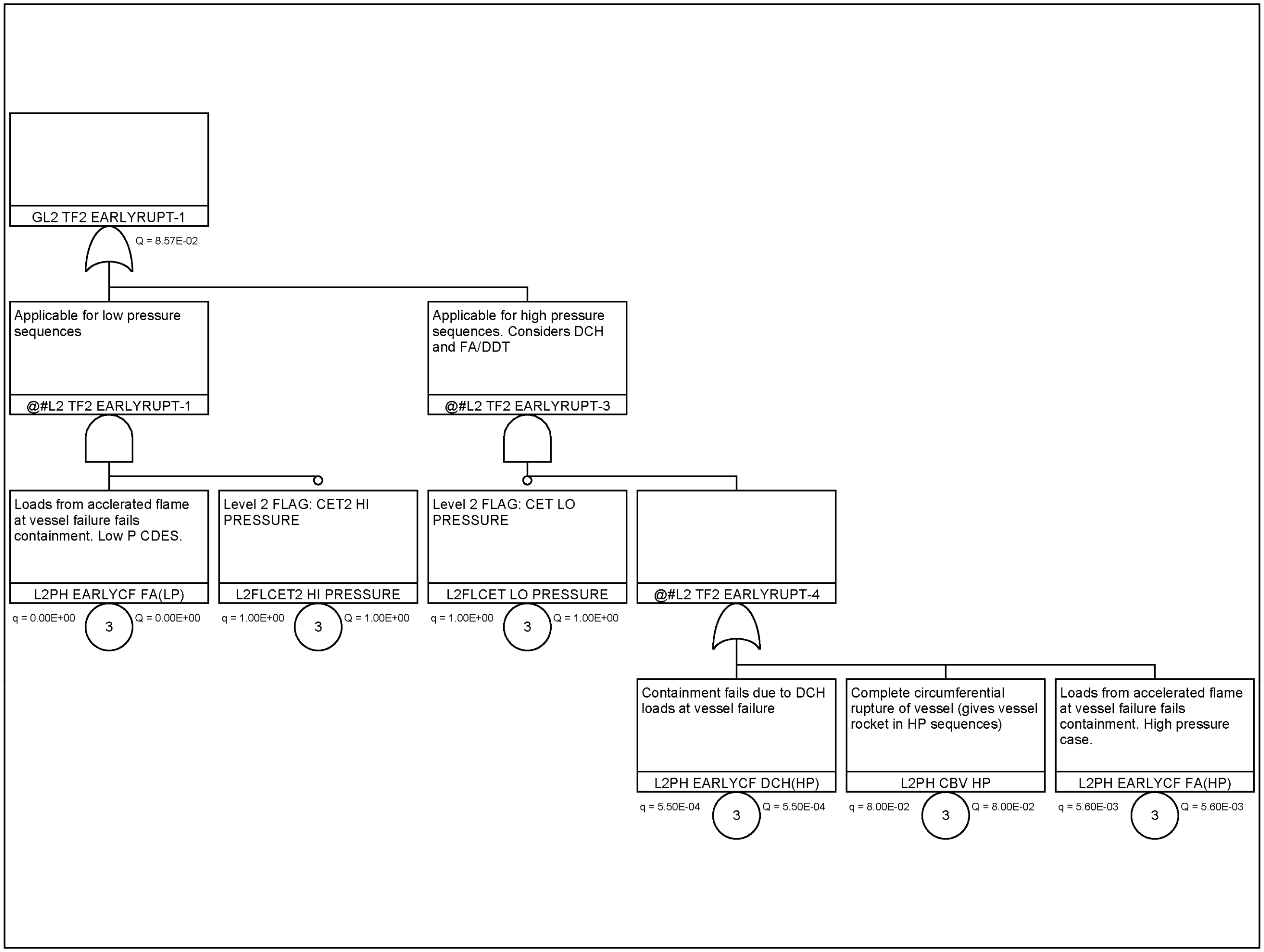


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 9 of 19)

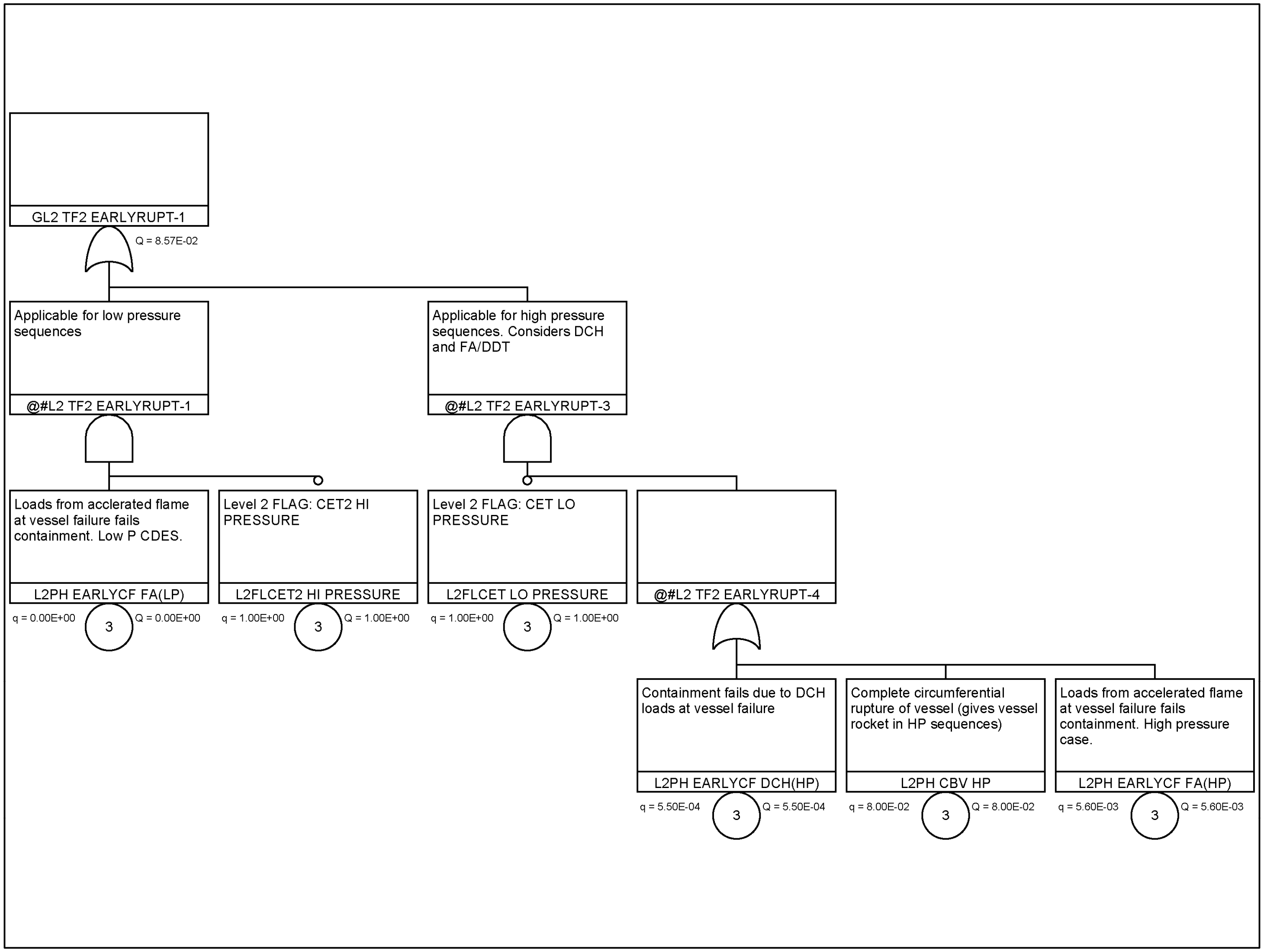


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 10 of 19)

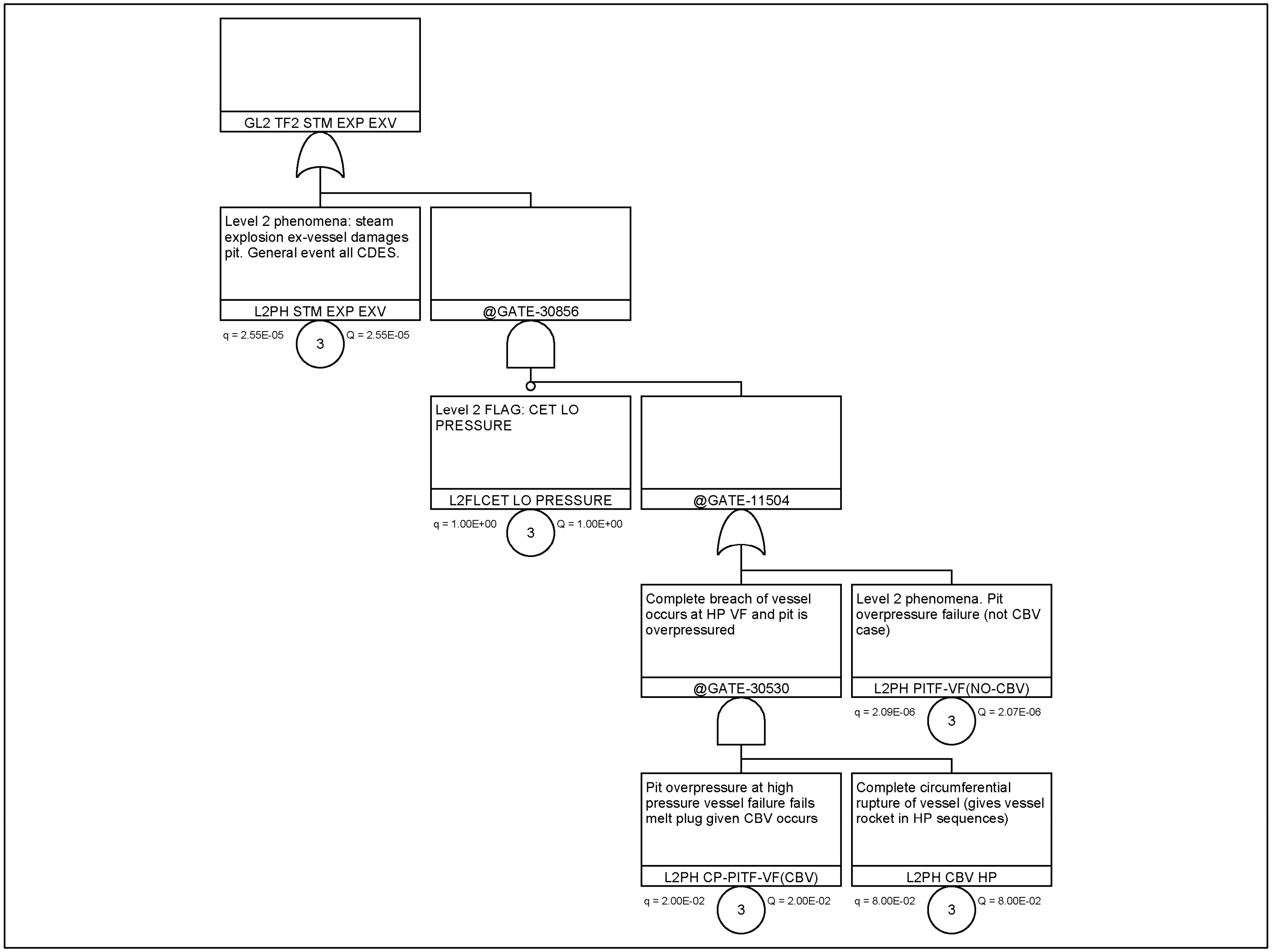


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 11 of 19)

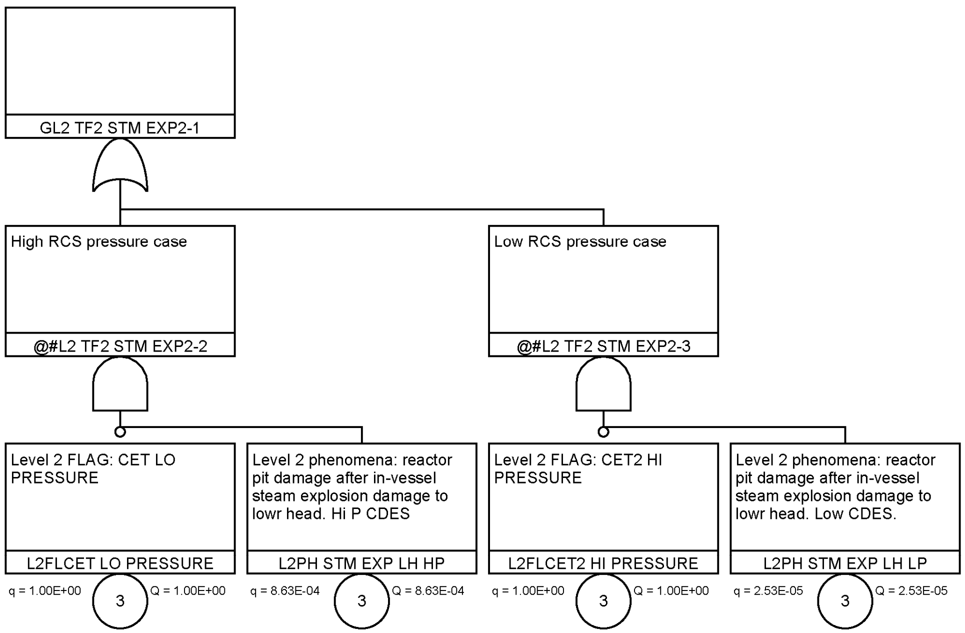


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 12 of 19)

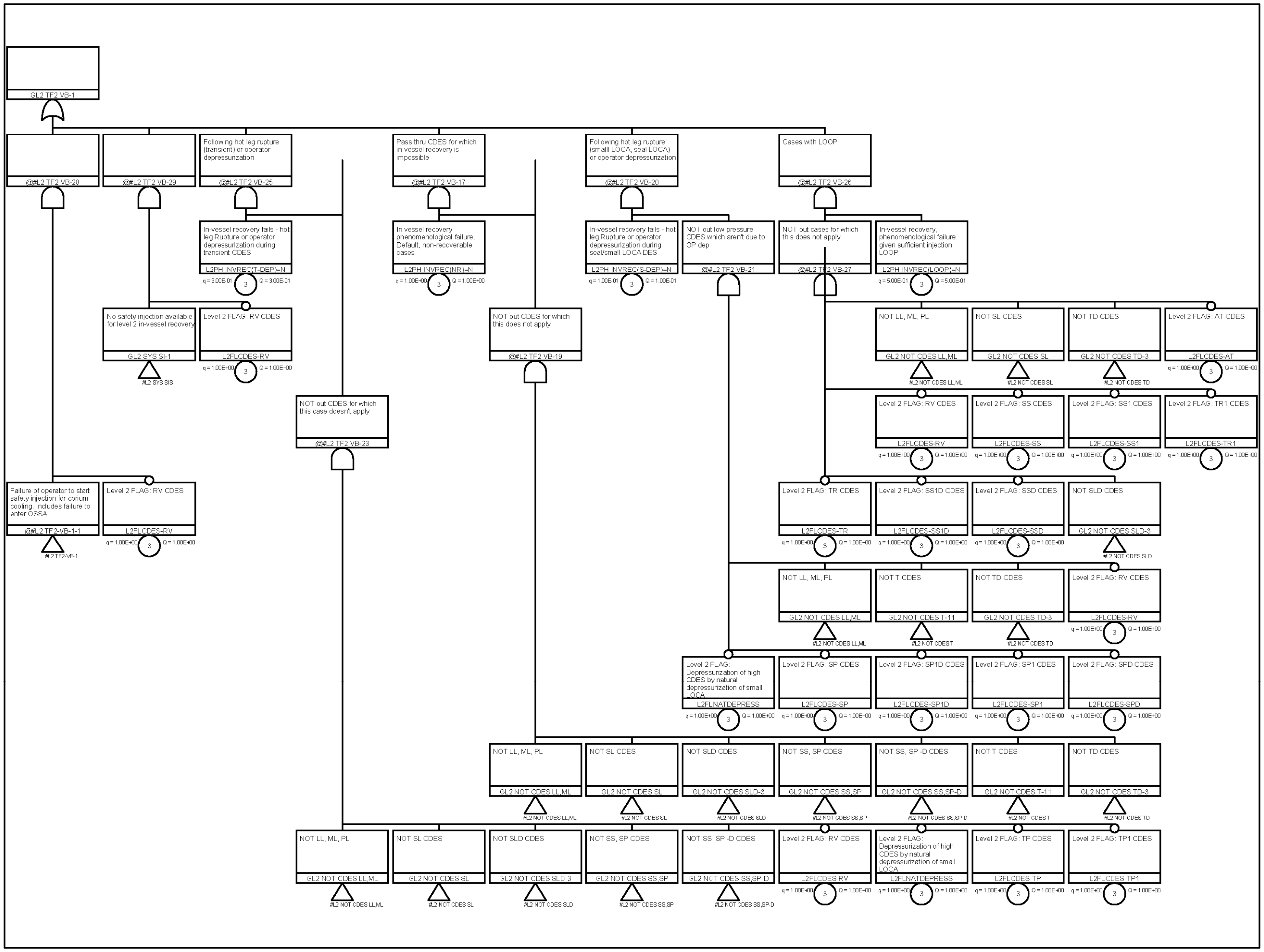


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 13 of 19)

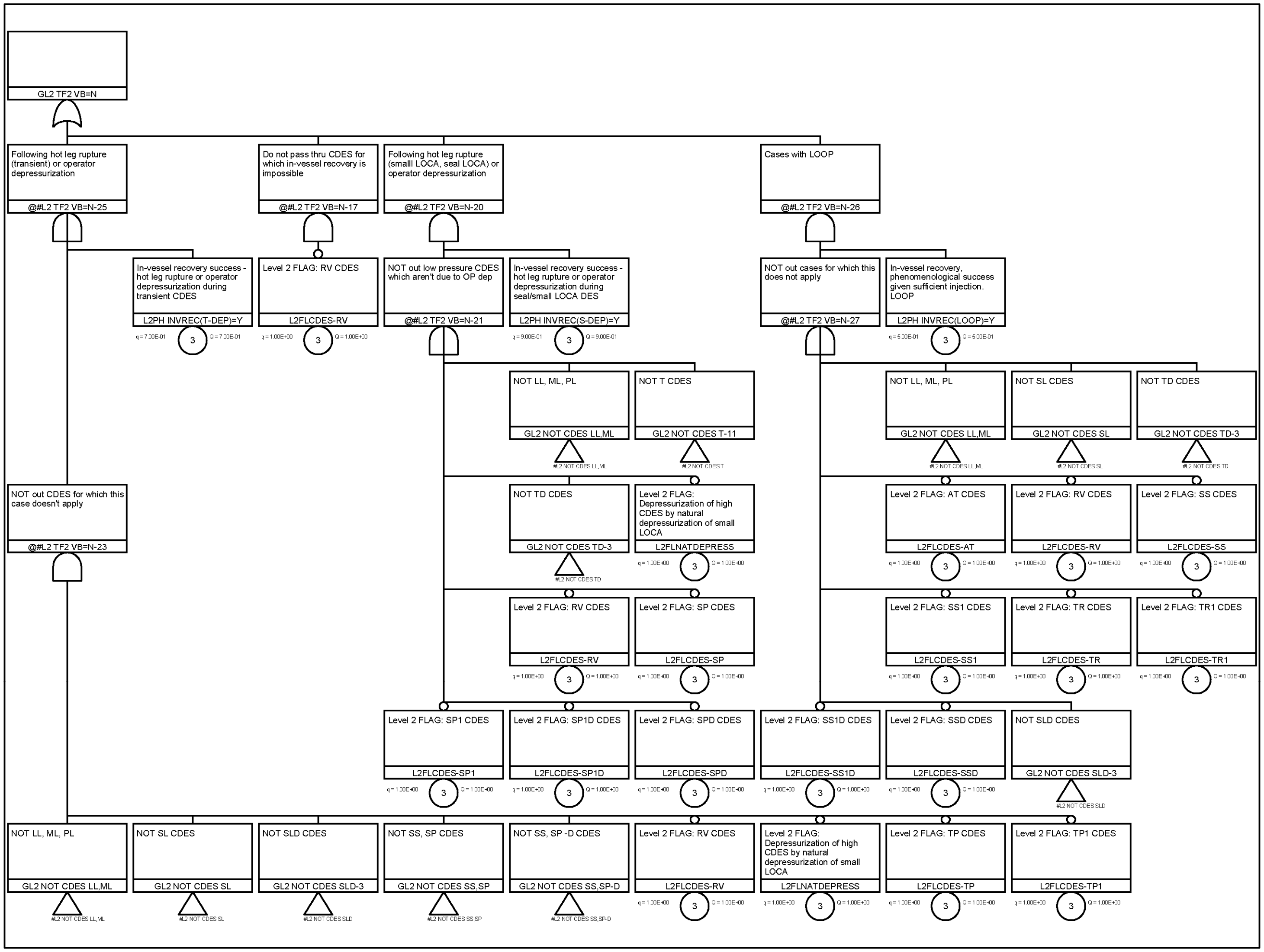


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 14 of 19)

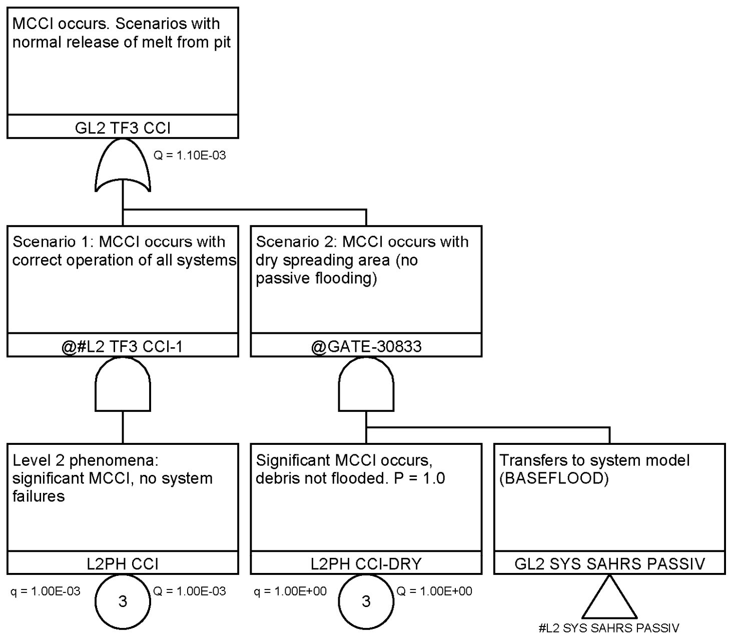


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 15 of 19)

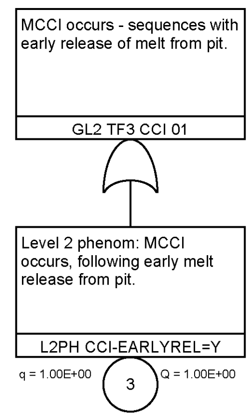


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 16 of 19)

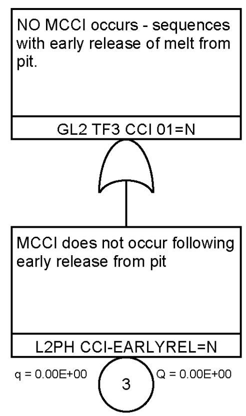


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 17 of 19)

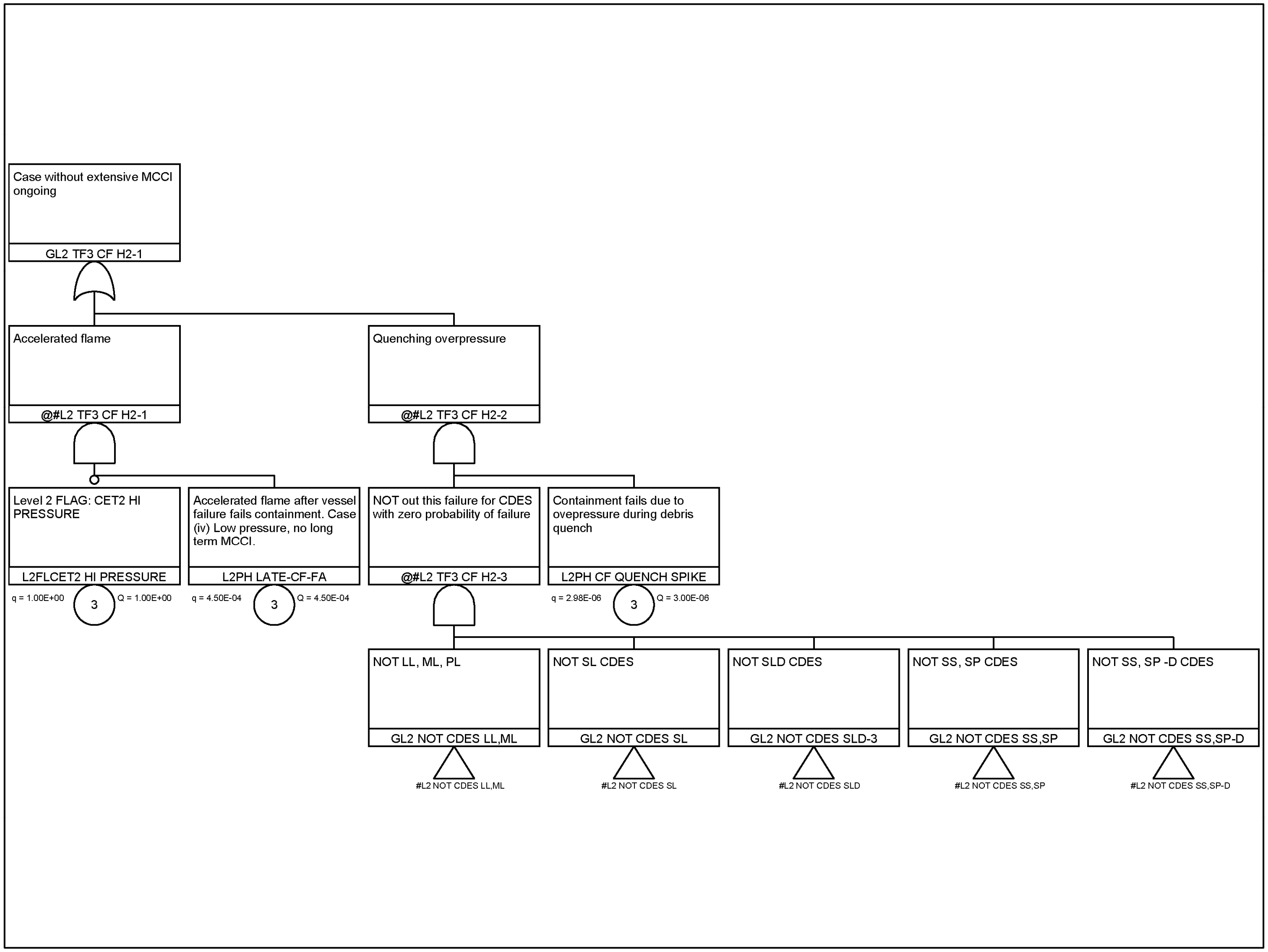


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 18 of 19)

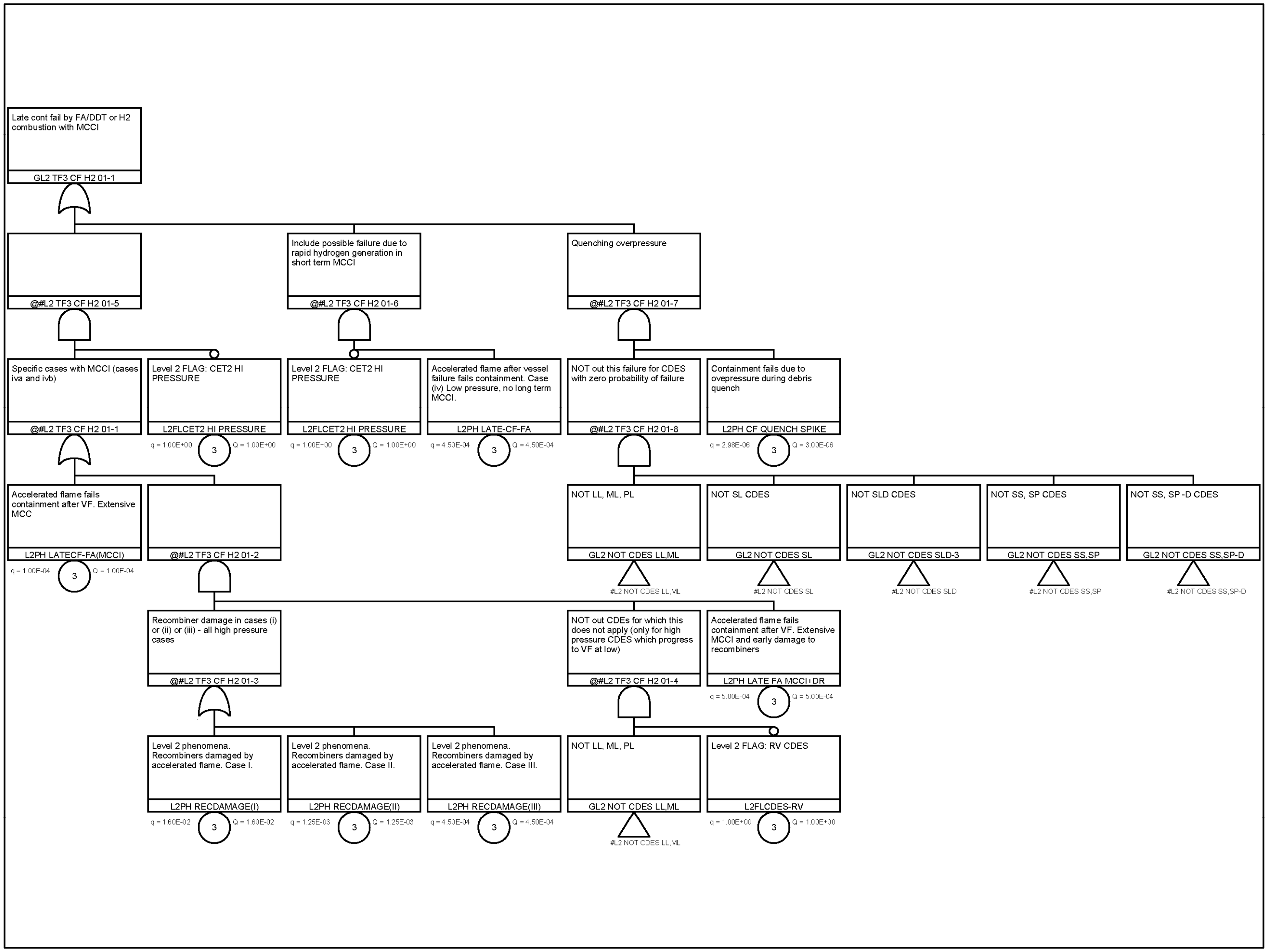
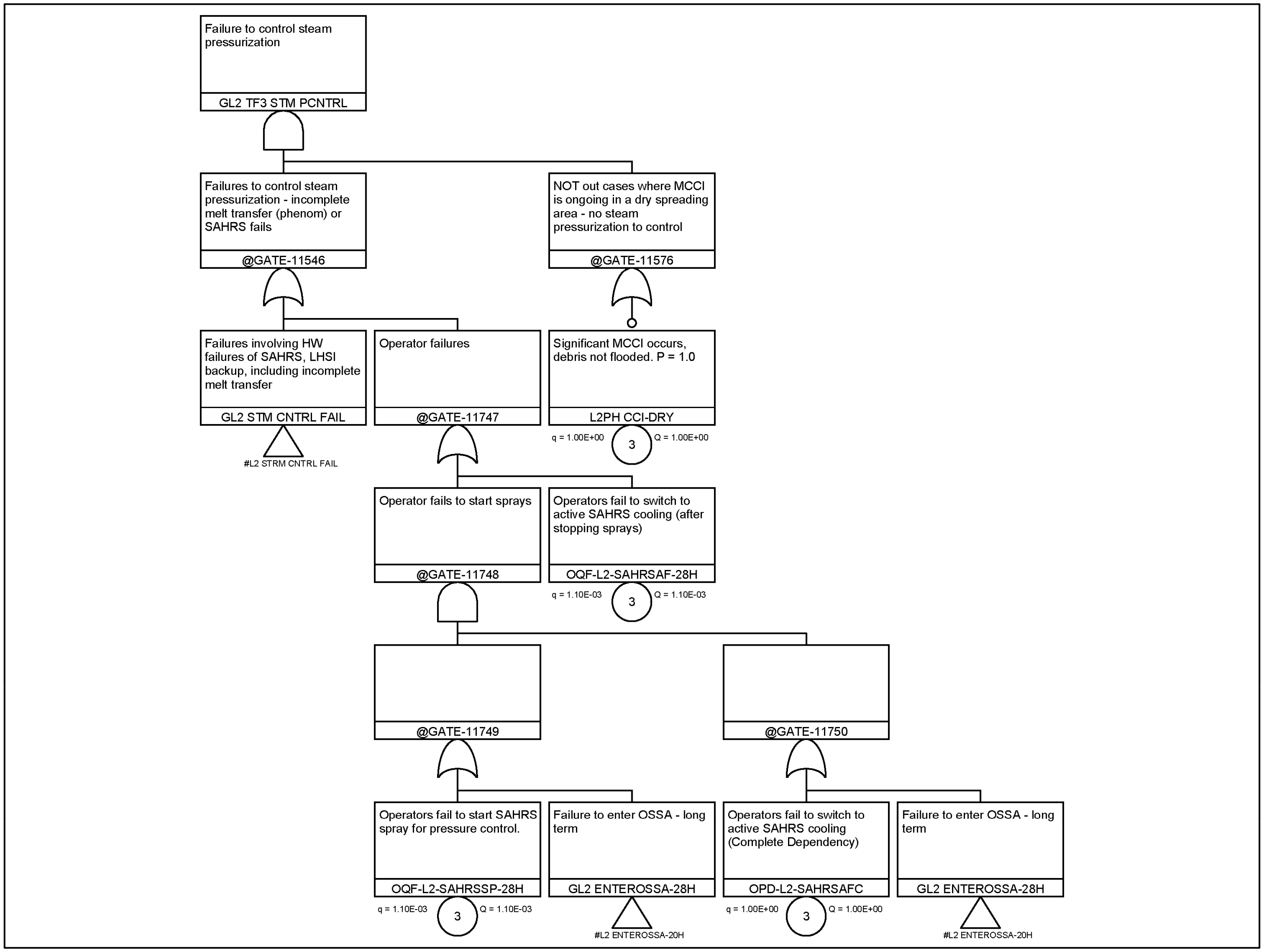


Figure 19-81-1—Containment Event Tree Supporting Fault Trees (Sheet 19 of 19)



Question 19-82:

Please define the MAAP cases run to support the CET quantification and phenomenological evaluations in the Level 2 PRA. For each initiator, indicate the status of safety injection, feedwater, secondary depressurization, primary depressurization, SAHRS, PARs/combustion, and other requirements. Explain the principal reason for including the case and identify the containment failure mode, release category, and sensitivity case. Include other relevant scenario details.

Response to Question 19-82:

Tables 19-82-1 and 19-82-2 provide the requested information.

The MAAP analysis was performed in two sets during the Level 2 analysis.

The first set of MAAP analyses were done to support the phenomenological evaluations, and were targeted to investigate particular aspects of the phenomena examined. The characteristics of each MAAP run are shown in Table 19-82-1. Because of the focused nature of these MAAP runs, the attributes of containment failure mode and release category did not necessarily apply, and are not noted in Table 19-82-1.

The second set of MAAP analyses were done to support the source term analysis, and include the attributes of containment failure mode, release category and sensitivity case if applicable. The characteristics of this second set of MAAP runs are shown in Table 19-82-2.

FSAR Impact:

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

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1] Passive flood	SAHRS[note 1] Spray	SAHRS[note 1] Active cool	PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
1.1	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	No simulation of induced rupture No seal LOCA	3: CDES TP Base case station blackout sequence with high pressure vessel failure. Used in induced rupture PE (and others)	Base case station blackout. Better to use 1.1F, which has FCRDR=0.3 to avoid slow addition of core debris to pit.
1.1A	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	No simulation of induced rupture No seal LOCA FFRICX=10. FFRICR=100.	3: Variant of 1.1 with reduced primary system gas natural circulation – sensitivity case for induced rupture	Both cases lead to NO creep rupture of hot leg
1.1AA	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	FFRICX=2.5 FFRICR=1.	3: Variant of 1.1 with reduced primary system gas natural circulation – sensitivity case for induced rupture	
1.1B	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	No simulation of induced rupture No seal LOCA XTSG= 50% of current value	3: (induced rupture, sensitivity to tube wall thickness)	
1.1C	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	No simulation of induced rupture No seal LOCA ICRPHL=2	3: (induced rupture, sensitivity to nozzle material – carbon steel)	
1.1D	Station blackout	none	none	All atmospheric SG relief valves open at time of core uncovery	none	yes	no	no	100% PARS available Combustion disabled	No simulation of induced rupture No seal LOCA	3: (induced rupture, sensitivity to depressurized SGs)	Induced tube rupture probabilities are significantly larger with depressurized SGs. Due to natural circulation of steam in secondary side when pressurized. See induced rupture PE

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
1.1E	Loss of main and emergency feedwater	none	none	none	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled	No simulation of induced rupture No seal LOCA	2: (check for differences between loss of power and loss of feed initiators)	Significant timing difference (this one is faster)
1.1F	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	FCRDR=0.3 No simulation of induced rupture No seal LOCA	Revised base case for station blackout with all core entering spreading area.	
1.1G	Loss of main and emergency feedwater	none	none	none	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled	FCRDR=0.3 No simulation of induced rupture No seal LOCA	Revised base case for station blackout with sprays, with all core entering spreading area.	
1.1H	Station blackout with 0.6 inch (total) seal LOCA	none	none	All atmospheric SG relief valves open at time of core uncovery	none	yes	no	no	100% PARS available Combustion disabled	FCRDR=0.3	3: Induced rupture CDES SS with SGs depressurized	
1.1I	Station blackout with 2 inch cold leg LOCA	none	none	All atmospheric SG relief valves open at time of core uncovery	none	yes	no	no	100% PARS available Combustion disabled	FCRDR=0.3	3: Induced rupture CDES SL with SGs depressurized	Severe! Tube rupture before hot leg.
1.1j	Station blackout with 2 inch cold leg LOCA	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	FCRDR=0.3	3: Induced rupture CDES SL with SGs pressurized	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
1.2	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Induced hot leg rupture simulated. Break flow to reactor pit (nozzle rupture).	3: (induced rupture – effects of hot leg rupture)	
1.3	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 12 hours	no	100% PARS available Combustion disabled		1: (PL) 3: Base case station blackout with ultimate depressurization	
1.3A	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	SAHRS spray recovered 5 hours after vessel failure.	no	100% PARS available Combustion disabled		1: PL, without it being a bypass 3: Effect of spray on hydrogen 5: Containment under-pressure	None of the cases analyzed led to containment under-pressure. This series shows the sensitivity of hydrogen concentration to sprays. In most cases, PARs ensure little or no increase in concentration when sprays actuated.
1.3B	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	SAHRS sprays on when Tcorout =650°C.	no	100% PARS available Combustion disabled		5: Containment under-pressure 3: Effect of spray on hydrogen	
1.3C	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	SAHRS sprays on when Tcorout =1050°C.	no	100% PARS available Combustion disabled		5: Containment under-pressure 3: Effect of spray on hydrogen	
1.3D	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled	Safety injection (LHSI) recovered 30 minutes before vessel failure	3: In-vessel recovery	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
1.3E	Station blackout	MHSI: 0/4 LHSI: 1/4	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	no	no	100% PARS available Combustion disabled	LHSI allowed to actuate and inject during the depressurization	4: Check on sequences which do not meet strict L1 success criteria. for feed and bleed	
1.4	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=1050°C	yes	Spray on at 12 hours	no	100% PARS available Combustion disabled		1: (TP) LOOP with depressurization at latest entry point to OSSA. Sensitivity with 1.3.	
1.5	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Loop seal and downcomer clearance in the broken loop only, at time of core uncovery	3: Induced ruptures with full loop nat circ Attempting to establish full loop circulation in the bk loop. Only limited success.	
1.6	Station blackout with 0.6 inch (total) seal LOCA	none	none	none	none	yes	no	no	100% PARS available Combustion disabled		3: Induced rupture with seal leak	
1.7	Station blackout with 2 inch (total) seal LOCA	none	none	none	none	yes	no	no	100% PARS available Combustion disabled		3: Induced rupture with seal leak	
1.7A	Reactor/turbine trip with 2 inch (total) seal LOCA (simulate LOCCW)	none	EFW available	Fast cooldown (to 20 bar) at 30 minutes	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	no	no	100% PARS available Combustion disabled		1: (SS)	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
1.7B	Reactor/turbine trip with 2 inch (total) seal LOCA (simulate LOCCW)	none	EFW available	Fast cooldown (to 20 bar) at 30 minutes	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 12 hours	no	100% PARS available Combustion disabled		2: Effect of spray	
1.7C	Reactor/turbine trip with 2 inch (total) seal LOCA (simulate LOCCW)	MHSI: 0/4 LHSI: 4/4	EFW available	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled		1: (SS)	
1.7D	Reactor/turbine trip with 2 inch (total) seal LOCA (simulate LOCCW)	none	EFW available	Fast cooldown (to 20 bar) at 30 minutes	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled		3: 1 (SS, LOCCW-43)	
1.8	SLOCA, 2 inch, hot leg	none	EFW available	Auto partial cooldown to 60 bar	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled		3: 1 (ML)	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
1.8A	SLOCA, 2 inch, hot leg	MHSI: 4/4 LHSI: 0/4	EFW available	Fast cooldown to 20 bar	none	yes	no	no	100% PARS available Combustion disabled		4: 1 (SL)	No core damage (not even a little bit…) Observation: the IRWST level is falling quite quickly. Some of this is steam (containment pressurizing and no SAHRS) but also several containment compartments are filling up with water (see ZWRB.XLS). This is unexpected since normally drainage back to RWST should be occurring. Suggests a check on junction elevations may be needed. Run released for use as not considered a major effect.
1.8B	SLOCA, 2 inch, hot leg	MHSI: 4/4 LHSI: 4/4	none	none	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled		4: 1 (SL)	
1.8C	S/MLOCA, 3 inch, hot leg	MHSI: 4/4 LHSI: 4/4	EFW available	none	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled		4: Success criterion check (partial cooldown needed or not).	No core damage, with margin.
1.8D	S/MLOCA, 3 inch, cold leg	MHSI: 4/4 LHSI: 4/4	EFW available	none	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled		4: Success criterion check (partial cooldown needed or not). Cold leg break location.	Cold leg typically limiting for core cooling. May be MAAP limitations with loop seal clearance?? No core damage, with margin. Thus implies partial cooldown not required for 3 inch and up cases with SI available. Probably also applies to cases w/o feedwater, since there would still be SG inventory at the time that the system depressurizes.

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
1.9	SLOCA, 2 inch, hot leg	none	EFW available	Auto partial cooldown to 60 bar	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled			
1.10	SGTR one tube double ended	MHSI: 4/4 LHSI: 4/4	EFW available	Auto partial cooldown to 60 bar	none	no (not relevant)			100% PARS available Combustion disabled	Ruptured SG isolated and ruptured SG GCT setpoint raised – as designed.	4: 1 (SG)	No core damage. Note – appears in L1.
1.11	SGTR one tube double ended	none	EFW available	Auto partial cooldown to 60 bar	none	no (not relevant)			100% PARS available Combustion disabled	Ruptured SG isolated and ruptured SG GCT setpoint raised – as designed.	4	No core damage.
1.12	SGTR one tube double ended	none	EFW available	Auto partial cooldown to 60 bar	none	no (not relevant)			100% PARS available Combustion disabled	Ruptured SG isolated but GCT setpoint not changed.	4: Check for time to core damage.	
1.13	SGTR one tube double ended	MHSI: 0/4 LHSI: 4/4	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled	Ruptured SG isolated and ruptured SG GCT setpoint raised – as designed.	1: (PL) 4	Very limited core damage
1.14	Steamline break outside containment upstream of MSIVs.	MHSI: 4/4 LHSI: 4/4	EFW available	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	no	no	100% PARS available Combustion disabled	Break area: 2x the area of one steamline (or flow limiters if existing), split equally amongst the four steamlines, upstream of MSIVs but outside containment JNSLB set to "28" the environment.	1: (TR) 4: (core damage?)	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1] Passive flood	SAHRS Spray	SAHRS Active cool	PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
1.15	MLOCA, 6 inch, hot leg	none	EFW available	none	none	yes	no	no	100% PARS available Combustion disabled		1 (ML) 3	
1.15A	MLOCA, 6 inch, hot leg	none	EFW available	none	none	yes	SAHRS sprays on when Tcorout =650°C.	no	100% PARS available Combustion disabled		5: Effect of SAHRS on hydrogen and containment pressure	
1.15B	MLOCA, 6 inch, hot leg	none	EFW available	none	none	yes	SAHRS sprays on when Tcorout =1050°C.	no	100% PARS available Combustion disabled		5: Effect of SAHRS on hydrogen and containment pressure	
1.15C	MLOCA, 6 inch, hot leg	none	EFW available	none	none	yes	SAHRS sprays on 5 hr after vessel failure.	no	100% PARS available Combustion disabled		5: Effect of SAHRS on hydrogen and containment pressure	
1.16	LLOCA Double ended loop pipe rupture on hot leg All accumulators available.	MHSI: 1/4 LHSI: 0/4	EFW available	none	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled		1: (LL, LLOCA-2) 3 4	Caution: model limitations

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
2.0A	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Vessel failure size = 0.1m diameter No simulation of induced rupture	3: Assessment of pit pressurization and rocketing	
2.0B	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Vessel failure size = 0.2m diameter No simulation of induced rupture	3: Assessment of pit pressurization and rocketing	
2.0C	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Vessel failure size = 1.5m diameter No simulation of induced rupture	3: Assessment of pit pressurization and rocketing	
2.0D	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Vessel failure size = 4.87m diameter No simulation of induced rupture	3: Assessment of pit pressurization and rocketing	
2.0E	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Vessel failure size = 0.5m diameter No simulation of induced rupture	3: Assessment of pit pressurization and rocketing	
2.1	Station blackout with 2 inch (total) seal LOCA	none	none	none	none	yes	Spray on at 5 bar (but not during quench)	no	75% PARs available.		Sensitivity to degraded PAR performance Check on SAHRS model and system depressurization capability.	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
2.1a	Station blackout with 2 inch (total) seal LOCA	none	none	none	none	yes	Spray on at 5 bar (but not during quench)	no	50% PARs available.		Sensitivity to degraded PAR performance Check on SAHRS model and system depressurization capability.	
2.2	Station blackout with 2 inch (total) seal LOCA	none	none	none	none	yes	Spray on at 5 bar (but not during quench)	no	100% PARS available Combustion disabled	Mixing damper failure to open. Containment flowpaths forced to remain closed throughout the run: 4, 5, 6, 7, 9, 10, 11, 12.	Sensitivity to failed rupture disks/foils	
2.3	Station blackout with 2 inch (total) seal LOCA	none	none	none	none	yes	Spray on at 5 bar (but not during quench)	no	50% PARs available.	Mixing damper failure to open. Containment flowpaths forced to remain closed throughout the run: 4, 5, 6, 7, 9, 10, 11, 12.	Sensitivity to failed rupture disks/foils and failed PARs	
2.4	Station blackout	none	none	none	none	yes	no	Active cooling on at 5 bar	100% PARS available Combustion disabled		Effectiveness of direct active cooling	
2.5	LLOCA	MHSI: 0/4 LHSI: 0/4 No accumu-lators	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at Pcontainment = 11 bar, with failure area 0.1m2.	Containment fragility - containment leak/break size sensitivity	
2.6	LLOCA	MHSI: 0/4 LHSI: 0/4 No accumu-lators	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at Pcontainment = 11 bar, with failure area 0.3m2.		

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
2.7	LLOCA	MHSI: 0/4 LHSI: 0/4 No accumul-ators	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at Pcontainment = 11 bar, with failure area 0.19m2.	Containment fragility - containment leak/break size sensitivity	
2.8	LLOCA	MHSI: 0/4 LHSI: 0/4 No accumul-ators	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at Pcontainment = 11 bar, with failure area 1m2.		
2.9	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at vessel failure, with failure area 0.1m2.	Containment fragility - containment leak/break size sensitivity	
2.10	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at vessel failure, with failure area 0.3m2.		
2.11	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at vessel failure, with failure area 0.19m2.		
2.12	Station blackout	none	none	none	none	yes	no	no	100% PARS available Combustion disabled	Containment failure at vessel failure, with failure area 1m2.		

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
2.13	LLOCA	MHSI: 0/4 LHSI: 0/4 No accumulators	none	none	none	yes	no	no	100% PARS available Combustion disabled	Case 2_5 with FCHF=1.0	Long term challenges - ex-vessel quench sensitivity	
2.13a	LLOCA	MHSI: 0/4 LHSI: 0/4 No accumulators	none	none	none	yes	no	no	100% PARS available Combustion disabled	Case 2_5 with FCHF=0.1		
2.14	LLOCA	MHSI: 0/4 LHSI: 0/4 No accumulators	none	none	none	yes	no	no	100% PARS available Combustion disabled	Case 2_5 with FCHF=0.0036		
2.15	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 12 hours	no	100% PARS available Combustion disabled	Recovery of one low head SI pump when max core outlet temperature reaches 1000°C .	Recovery in-vessel	
2.16	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 12 hours	no	100% PARS available Combustion disabled	Recovery of one low head SI pump when relocation begins (event code 2 TRUE)		
2.17	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 12 hours	no	100% PARS available Combustion disabled	Recovery of one low head SI pump when lower head dries out		

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
2.18	Station blackout	none	none	none	One dedicated valve (900 t/h) open at Tcorout=650°C	yes	Spray on at 12 hours	no	100% PARS available Combustion disabled	Recovery of one low head SI pump 600 seconds before vessel failure.	Recovery in-vessel	
2.19	LLOCA	MHSI: 1/4 LHSI: 0/4 No accumulators	none	none	none	yes	no	no	100% PARS available Combustion disabled		Success criterion check	Severe core damage results. MHSI alone (without accumulators or LHSI does refill the vessel, but it takes nearly one hour, and severe damage has occurred, and is not arrested by reflood. MHSI flow is about 50kg/s. Good example of an “inadequate injection flow” case.
2.20	LLOCA	MHSI: 0/4 LHSI: 1/4 No accumulators	none	none	none	yes	no	no	100% PARS available Combustion disabled		Success criterion check	Severe damage results. LHSI refloods more quickly, but with 1 pump, still not quick enough to avoid melting of around 25 tons, which subsequently re-solidifies but is non-coolable. These two runs indicate that accumulators are needed in large LOCA (or more LHSI pumps??)

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
3.1A	Station blackout	none	none	none	none	no	no	no	100% PARS available Combustion disabled	Fail passive flooding (dry spreading area) No SAHRS No hydrogen combustion. No hot leg creep rupture Spreading area 100% (iACMPLB(2)=177.5 ("nominal value) - ACMPLB(2)=177.5 - remove the FCHF include file - JFRB(185)=0 - JFRB(33)=0 - FCRDR=0.3 - XTHSRB(148)=5.0 - KHSF(5)=2.0	3: Long term challenges – MCCI and hydrogen "Base case" with dry spreading area.	
3.1C	Station blackout	none	none	none	none	no	no	no	100% PARS available Combustion disabled	Confine debris to 50% of spreading area. Fail passive flooding (dry spreading area) No SAHRS No hydrogen combustion. No hot leg creep rupture ACMPLB(2)=88.75 remove the FCHF include file JFRB(185)=0 JFRB(33)=0 FCRDR=0.3 XTHSRB(148)=5.0 XRBLK(2,1) = 5.20.	Long term MCCI for hydrogen and long term challenges. Effect of incomplete spreading.	
3.1D	Station blackout	none	none	none	none	no	no	no	100% PARS available Combustion disabled	Variant of 3.1A with high CaO concrete composition	Effect of high CaO concrete in MCCI case	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
3.2	Station blackout	none	none	none	none	no	no	no	100% PARS available Combustion disabled	Fail passive flooding (dry spreading area) No SAHRS No hydrogen combustion. No hot leg creep rupture Recover 1 LHSI 5 hours after vessel failure		
3.4	Station blackout	none	none	none	none	no	Spray on at vessel failure.	no	100% PARS available Combustion disabled	Model hot leg rupture	Long term MCCI for hydrogen and long term challenges	
3.6	0.6 inch diameter seal LOCA (split among the four intermediate legs)	none	none	none	none	no	Spray on at vessel failure.	no	100% PARS available Combustion disabled	No hot leg rupture		
3.7	0.36 inch diameter seal LOCA (split among the four intermediate legs)	none	none	none	none	yes	no	Active cooling on at 5 bar	100% PARS available Combustion disabled	No hot leg rupture	Effectiveness of active cooling w/o spray for seal LOCA case	
3.8	2 inch diameter seal LOCA (split among the four intermediate legs)	none	none	none	none	no	Spray on at vessel failure.	no	100% PARS available Combustion disabled	No hot leg rupture	Long term MCCI for hydrogen and long term challenges	

Table 19-82-1—U.S. EPR PRA Level 2 – MAAP Analysis in Support of CET Quantification and Phenomenological Evaluation – Summary Table

Case	Initiator	SI	Feedwater[note2]	Secondary Depressurization	Primary Depressurization	SAHRS[note 1]			PARS/ Combustion	Other Requirements	Principal reason for inclusion 1: Representative CDES sequence 2: Sensitivity calculation 3: Support to phenomenological evaluation 4: Check for core damage	Notes/Comments/ Conclusions
						Passive flood	Spray	Active cool				
3.11	Station blackout	none	none	none	none	no	Spray on at vessel failure.	no	25% PARS unavailable	Model hot leg rupture	Long term MCCI for hydrogen and long term challenges	
3.13	1.5 inch diameter seal LOCA (split among the four intermediate legs)	none	none	none	none	no	Spray on at vessel failure.	no	25% PARS unavailable	No hot leg rupture	Long term MCCI for hydrogen and long term challenges	
3.16	1.5 inch diameter seal LOCA (split among the four intermediate legs)	none	none	none	none	no	Spray on at vessel failure.	no	100% PARS available Combustion disabled	Hydrogen combustion re-enabled at vessel failure. Disabled again 15 minutes later	Long term MCCI for hydrogen and long term challenges	There is some combustion at vf, but not all hydrogen generated in-vessel is burned since steam fraction is very high. (However, there is not much left by this time due to recombination) Run can be compared directly with 3.13 for impact of recombiners. Compare with 3.8 to see important effect of break size 2 inch vs. 1.5 inch – especially on the time for the primary to depressurize below the SGs and the resultant effect on accumulator injection behavior and conditions at vessel failure.

Notes:

Note 1: Enable the SAHRS heat exchanger for all cases with SAHRS in operation.
Note 2: For all cases with feedwater available, DO NOT allow the CST to empty. Recommended local parameter change ALL RUNS: MWCST0=1e10, ACST=232.2*1e10/1.68e6

Table 19-82-2—U.S. EPR PRA Level 2 – Source Term MAAP4.0.7 Analysis Summary Table

Case	Initiator	SI	Feed-water[note2]	Sec Depress.	Primary Depress.	Hot Leg Rupture	SAHRS[note 1]			PARS / combustion	Other Scenario Details	Containment Failure Mode	Release Category Or Sensitivity Case
							Passive flood	Sprays (SPA)	Active cooling (SPC)				
st1_1	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	On at TCOROUT =1050°C	no	PARS OK (100%) Combustion permitted (model enabled)	Seal failure – 0.6 inch diameter break split equally between all four loops. Elevation of pump seals.	No failure	S: Intact containment source term with sprays (early initiation)
St1_1_5bar	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	On at 5 bar (never reached – so used as sensitivity case w/o sprays)	no	PARS OK (100%) Combustion permitted (model enabled)	Seal failure – 0.6 inch diameter break split equally between all four loops. Elevation of pump seals.	No failure	S: Intact containment source term without sprays
st1_2	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	On 2hours after corium enters spreading area	PARS OK (100%) Combustion permitted (model enabled)	Seal failure – 0.6 inch diameter break split equally between all four loops. Elevation of pump seals.	No failure	S: Importance of sprays in intact containment (active cooling instead of sprays)
st1_2_jfrb2	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	On 2hours after corium enters spreading area	PARS OK (100%) Combustion permitted (model enabled)	JFRB(2)=0 1 hour after corium enters spreading area	No failure	S: Diagnostic run – case with dry reactor pit to investigate importance of residual corium in the pit (Compare with st1_2)
St1.4	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	1. Seal failure (same as st1.1) 2. Pre-existing (ie from time zero) opening between the containment and the environment, at a room with an outer containment wall and elevation around the elevation of the primary loops. Diameter of opening: 0.5 inch.	Pre-existing opening (isolation failure)	S: Sensitivity to isolation failure diameter
St1.5	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	1. Seal failure (same as st1.1) 2. Pre-existing (ie from time zero) opening between the containment and the environment, at a room with an outer containment wall and elevation around the elevation of the primary loops. Diameter of opening: 1 inch.	Pre-existing opening (isolation failure)	S: Sensitivity to isolation failure diameter

Table 19-82-2—U.S. EPR PRA Level 2 – Source Term MAAP4.0.7 Analysis Summary Table

Case	Initiator	SI	Feed-water[note2]	Sec Depress.	Primary Depress.	Hot Leg Rupture	SAHRS[note 1]			PARS / combustion	Other Scenario Details	Containment Failure Mode	Release Category Or Sensitivity Case
							Passive flood	Sprays (SPA)	Active cooling (SPC)				
St1.6	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	1. Seal failure (same as st1.1) 2. Pre-existing (ie from time zero) opening between the containment and the environment, at a room with an outer containment wall and elevation around the elevation of the primary loops. Diameter of opening: 6.3 inch. (10 times the area of a 2" break)	Pre-existing opening (isolation failure)	S: Sensitivity to isolation failure diameter
St1.7	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	NO seal failure 1 inch pre-existing opening	Pre-existing opening (isolation failure)	S: Sensitivity of CIF source term to seal leakage
St1.8	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	No	Prevented if predicted	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	1 square meter opening	Pre-existing opening (isolation failure)	205
st1.8a	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	No	Prevented if predicted	yes	On at TCOROUT =1050°C	no	PARS OK (100%) Combustion permitted (model enabled)	1 square meter opening	Pre-existing opening (isolation failure)	204
st1.8b	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	No	Prevented if predicted	no	no	no	PARS OK (100%) Combustion permitted (model enabled)	1 square meter opening, as st1.8 MCCI case: • JFRB(185)=0 (no flood) • FCHF include file removed • XTHSRB(148)=5.0 • KHSF(5)=2.0 • CaCO3 in concrete note 3 below	Pre-existing opening (isolation failure)	203
st1.8c	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	No	Prevented if predicted	no	On at TCOROUT =1050°C	no	PARS OK (100%) Combustion permitted (model enabled)	1 square meter opening, as st1.8 MCCI case: • JFRB(185)=0 (no flood) • FCHF include file removed • XTHSRB(148)=5.0 • KHSF(5)=2.0 • CaCO3 in concrete note 3 below	Pre-existing opening (isolation failure)	202
st1.8d	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see "other requirements")	none	none	none	No	Prevented if predicted	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	2 inch opening	Pre-existing opening (isolation failure)	S (base case for sensitivity to depressurization and hot leg rupture in following runs)

Table 19-82-2—U.S. EPR PRA Level 2 – Source Term MAAP4.0.7 Analysis Summary Table

Case	Initiator	SI	Feed-water[note2]	Sec Depress.	Primary Depress.	Hot Leg Rupture	SAHRS[note 1]			PARS / combustion	Other Scenario Details	Containment Failure Mode	Release Category Or Sensitivity Case
							Passive flood	Sprays (SPA)	Active cooling (SPC)				
st1.8e	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see “other requirements”)	none	none	none	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	2 inch opening	Pre-existing opening (isolation failure)	S (cf 1.8d for effect of depressurizqtion)
St1.8f	Station blackout (LOOP, no diesels) with 0.6 inch diam seal LOCA (see “other requirements”)	none	none	none	no	yes	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	2 inch opening	Pre-existing opening (isolation failure)	206 S (cf 1.8d for effect of hot leg rupture)
st1.10	Station blackout (LOOP). No seal LOCA	none	none	none	none	yes	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	Hot leg rupture predicted and modeled. Break flow to node 8 (pit).	Overpressure rupture: • Containment failure at 11.64 bar absolute internal pressure • 1m2 opening at elevation 128ft. • Discharge direct to environment (no retention in annulus)	504 101
st1.10a	Station blackout (LOOP). No seal LOCA	none	none	none	none	yes	no	no	no	PARS OK (100%) Combustion permitted (model enabled)	Hot leg rupture predicted and modeled. Break flow to node 8 (pit). MCCI case: • JFRB(185)=0 (no flood) • FCHF include file removed • XTHSRB(148)=5.0 • KHSF(5)=2.0 • CaCO3 in concrete note 3 below	Overpressure rupture at 60 hours	502
st1.10b	Station blackout (LOOP). No seal LOCA	none	none	none	none	yes	no	no	no	PARS OK (100%) Combustion permitted (model enabled)	Hot leg rupture predicted and modeled. Break flow to node 8 (pit). MCCI case: • JFRB(185)=0 (no flood) • FCHF include file removed • XTHSRB(148)=5.0 • KHSF(5)=2.0 • CaCO3 in concrete note 3 below	Overpressure rupture at time of vessel failure	402

Table 19-82-2—U.S. EPR PRA Level 2 – Source Term MAAP4.0.7 Analysis Summary Table

Case	Initiator	SI	Feed-water[note2]	Sec Depress.	Primary Depress.	Hot Leg Rupture	SAHRS [note 1]			PARS / combustion	Other Scenario Details	Containment Failure Mode	Release Category Or Sensitivity Case
							Passive flood	Sprays (SPA)	Active cooling (SPC)				
st1.10c	Station blackout (LOOP). No seal LOCA	none	none	none	none	yes	no	On at time of vessel/containment failure	no	PARS OK (100%) Combustion permitted (model enabled)	Please ensure hot leg rupture is predicted and modeled. Break flow to node 8 (pit) (variables JNBB and JNUB set to 8) PLUS: this is a MCCI case, following changes needed: • JFRB(185)=0 (no flood) • Remove the FCHF include file • XTHSRB(148)=5.0 • KHSF(5)=2.0 • CaCO3 in concrete note 3 below	Overpressure rupture at time of vessel failure	401
st1.11	LOOP with diesels (same as station blackout but with SI available) with 0.6 inch diam seal LOCA (see "other requirements")	MHSI: 4/4 LHSI: 4/4	none	none	none	yes	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	1 square meter pre-existing opening in containment (isolation failure) 0.6" seal LOCA as st1.1 Hot leg rupture break flow to pit	Pre-existing opening (isolation failure)	201
St2.3	Station blackout with 2 inch cold leg LOCA	None available	No EFW or SSS	All atmospheric SG relief valves open at time of core uncovery	No	Induced SGTR modeled Hot leg rupture prevented	Yes	No	No	PARS OK (100%) Combustion permitted (model enabled)		Bypass (SGTR)	702
st3.1	10 inch diameter hot leg LOCA outside containment (ISLOCA)	3/4 MHSI 3/4 LHSI	No EFW or SSS	none	no	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	Release to atmosphere Ground level release height	Bypass (ISLOCA)	802
St3.1a	10 inch diameter hot leg LOCA outside containment (ISLOCA)	3/4 MHSI 3/4 LHSI	No EFW or SSS	none	no	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	Release to building Ground level release height	Bypass (ISLOCA)	802
st3_2	3 inch diameter hot leg LOCA outside containment (ISLOCA)	3/4 MHSI 3/4 LHSI	EFW available	Partial secondary cooldown OK	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	Release to atmosphere Ground level release height	Bypass (ISLOCA)	802
St3_2a	3 inch diameter hot leg LOCA outside containment (ISLOCA)	3/4 MHSI 3/4 LHSI	EFW available	Partial secondary cooldown OK	Open 1 dedicated valve (900 t/hr at Tcorout max =650°C	no	yes	no	no	PARS OK (100%) Combustion permitted (model enabled)	Release to building Ground level release height	Bypass (ISLOCA)	802

Notes:

Note 1: Enable the SAHRS heat exchanger for all cases with SAHRS in operation.
Note 2: For all cases with feedwater available, DO NOT allow the CST to empty. Recommended local parameter change ALL RUNS: MWCST0=1e10, ACST=232.2*1e10/1.68e6.

Question 19-83:

Please provide a source term grouping diagram that includes the various attributes of the accident sequences that have been considered in defining and describing the release categories. In addition, please provide a mapping of sequences simulated by MAAP 4.0.7 runs to the release categories.

Response to Question 19-83:

Figure 19-83-1 provides the source term grouping diagram that includes the various attributes of the accident sequences that have been considered in defining and describing the release categories.

Table 19-83-1 provides a mapping of sequences simulated by MAAP 4.0.7 runs to the release categories.

FSAR Impact:

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

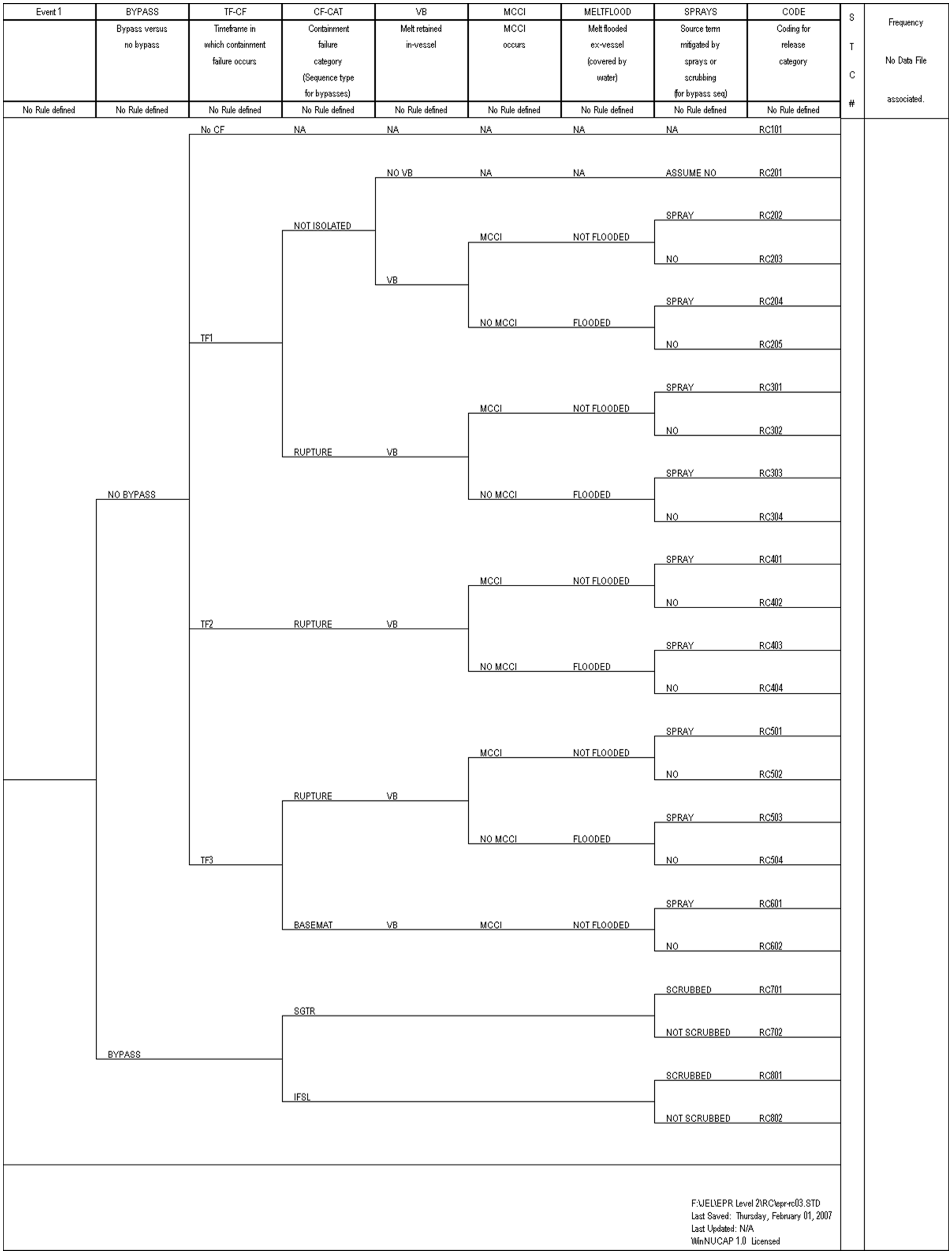
Table 19-83-1—Release Category MAAP Runs[5]

Release Category	MAAP Run
RC101	ST1.10[1]
RC201	ST1.11
RC202	ST1.8c
RC203	ST1.8b
RC204	ST1.8a
RC205	ST1.8
RC206	ST1.8f
RC301	ST1.8c
RC302	ST1.8b
RC303	ST1.8a
RC304	ST1.8
RC401	ST1.10c
RC402	ST1.10b
RC403	ST1.10c
RC404	ST1.10b
RC501	ST1.10a[3]
RC502	ST1.10a
RC503	ST1.10[3]
RC504	ST1.10
RC602	ST1.10a[4]
RC701	ST2.3[2]
RC702	ST2.3
RC802	ST3.2a

Notes for Table 19-83-1:

1. Used ST1.10 to a shorter time than RC504 since there is no containment damage is RC101.
2. Used ST2.3, but applied a decontamination factor of 20 to the release fractions; thus the release fractions are the only difference between the MAAP4.0.7 characterization of RC701 and RC702.
3. Used MAAP4.0.7 runs without sprays as a bounding approximation since there were no MAAP4.0.7 runs available with sprays functional.
4. Used ST1.10a as a surrogate for RC602, since there were no MAAP4.0.7 runs available that modeled basemat melt-through.
5. Table 19-82-1 in the Response to Question 19-82 provides the characteristics for each of the MAAP runs listed in Table 19-83-1.

Figure 19-83-1—Source Term Grouping Diagram



Question 19-84:

The tabulated release fractions in the FSAR show peculiarities as related to reactor coolant system retention, and overall release magnitudes. In order to confirm the consistency of the tabulated data, please provide the following source term information for the Level2/Level 3 interface:

- The technical bases for various release fractions listed in Table 19.1-20 (i.e., release evolution of various groups from fuel, retention in the reactor coolant system, the steam generator secondary side (in case of SGTR), the reactor containment, and the annulus and/or other buildings, as applicable). Please limit the information to risk- and consequence-dominant scenarios.
- Show how the twelve MAAP radiological groups (defined in pages 19.1-91 and 19.1-92) were regrouped into the nine radiological groups in Table 19.1.-20 of the FSAR.
- For each of the Release Categories listed in Table 19.1-20, provide the release characteristics (e.g., time of alarm, delay time, number of plumes, plume duration, plume energy, etc.) necessary for ex-plant consequence analysis.
- It is recognized that an assessment of uncertainties in source terms was not performed. Instead, a number of MAAP parametric sensitivity calculations were performed. Please provide a list of sensitivity cases, including the MAAP parameters, the associated ranges, the basis for their selection, and the resulting impact on the calculated fission product release and transport results as applicable to U.S. EPR.
- Provide the assumptions related to the number of steam generator tubes that are considered in the MAAP analyses that result in the release quantities listed for Release Category 702 of Table 19.1-20 (i.e., steam generator tube rupture with no scrubbing). In addition, provide the technical justification that supports the bounding nature of the selected scenario for this release category with regards to the multiple tube ruptures in the steam generators under this accident conditions.
- Please provide the technical bases for apparent high retention associated with the release magnitude for Release Category 802 (i.e., Interfacing Systems LOCA with no scrubbing). In addition, please discuss the rationale that the selected scenario for this release category envelops other Interfacing Systems LOCA scenarios.
- The data listed in various tables are not self-explanatory, since not all abbreviations and/or acronyms have been fully defined. For instance, Table 19.1-26 lists fifteen different acronyms for the core damage end states (CDES), which are not described. Please provide a complete list of abbreviations and acronyms applicable to FSAR Sections 19.1 and 19.2. In addition, with respect to this table, please discuss the differences between the two TP1 entries, and provide the contributions of each CDES to the applicable release category.

Response to Question 19-84:

A response to this question will be provided by August 8, 2008.

Question 19-85:

Please provide the analysis of the scrubbing of releases in the steam generator for release categories 701 and 702.

Response to Question 19-85:

The Level 2 PRA has two release categories for SGTRs. The SGTRs are not processed through the containment event tree since they are bypass sequences, and the only question posed is “are the releases scrubbed in the SG?”, and then categorized into RC701 or RC702 (scrubbed and unscrubbed respectively). Induced and initiator SGTRs are treated in the same way.

The MAAP4.0.7 analyses for the U.S. EPR included analysis of an induced SGTR source term. This is an unscrubbed case. However, no SGTR sequences with feedwater available have been analyzed. The purpose of this analysis is to investigate likely decontamination factors (DFs) in the SGs.

Important MAAP4.0.7 models used in the SGTR source term analysis are shown in Table 19-85-1. Full descriptions of the modeling can be found in the descriptions of the associated subroutines in Volume 2 of Electric Power Research Institute, MAAP 4 Computer Code Manual, May 1994.

Pool scrubbing is modeled in MAAP4.0.7 via a series of lookup tables with data originating from analysis with the SUPRA® pool scrubbing code. DFs are interpolated for each aerosol particle size range and for various thermal hydraulic conditions, as shown in Table 19-85-2.

Table 19-85-1—MAAP4.0.7 Models of Pool Scrubbing

Name	Type	Description
POOLD	Blockdata	Contains all the look-up data based on SUPRA® analysis of pool scrubbing
POOLDF	Subroutine	Calculates overall pool DF
AERODF	Subroutine	Calculates overall pool DF
AMDEF	Subroutine	Sets up mass / size distributions for aerosols

Table 19-85-2—Independent Parameters and their Respective Values for the DF Data Tables in MAAP4.0.7

Mode of Gas Injection[1]	Parameter	Range of Values Used to Create DF Tables
Sparger	Aerosol Particle Radius Steam Mass Fraction Pool Height Pressure Subcooling of Pool Gas Composition	Ten Values[2] 0, 0.5, 0.9, 0.99 0.5, 1, 2, 4, 6 m 1, 2, 5 atm 0, 1, 1.15, 1.5, 10.15, 10.5, 30, 30.15, 30.5 K Hydrogen
Downcomer	Aerosol Particle Radius Steam Mass Fraction Pool Height Pressure Subcooling of Pool Gas Composition	Ten Values[2] 0, 0.5, 0.9, 0.99 0.5, 1, 3 m 1, 5 atm 0, 1, 1.15, 2.15. 2.5, 10.15, 10.5, 30.15, 30.5 K Air, Hydrogen
Side Vent	Aerosol Particle Radius Steam Mass Fraction Pool Height Pressure Subcooling of Pool Gas Composition	Ten Values[2] 0, 0.5, 0.9, 0.99 0.5, 0.8, 1.8 m 1 atm 0, 1.15, 2.15, 10.15, 30.15K Air, Hydrogen

Notes for Table 19-85-2:

1. The nominal geometry used in the SUPRA® calculation for each type of injector is as follows. The DFs are a strong function of the mode of injection, but are not very dependent on the injector diameter.
   - Sparger – 1500 orifices per sparger 0.5 in diameter orifices.
   - Downcomer – 2 foot diameter vertical pipe.
   - Side Vent – 2.3 foot diameter opening in wall of pool.
2. The ten particle radii are 0.01, 0.035, 0.05, 0.08, 0.1, 0.2, 0.4, 0.6, 0.8, and 1.0 microns.

Estimation of SG DFs:

Interpolations were done within a spreadsheet with data chosen to be typical of a SGTR severe accident, as shown in Table 19-85-3.

Figure 19-85-1 shows the results for “downcomer” injection mode, and various of the other parameters. If the 3m deep pool with zero subcooling is used, a wide range of DF is apparent, with all particle sizes larger than 0.3 – 0.4 micron radius experiencing DF>20. In order to have an idea of the aerosol size distribution, Figure 19-85-4 was used (distribution with aerosol source taken since it has more small particles which have lower DFs). To read this plot, the non-dimensional particle size parameter VR has to be converted back to a radius in meters (refer to Table 19-85-4). This was done using the data shown in Table 19-85-3 with the result shown in Table 19-85-5. It can be seen that at about a 0.5 micron radius, the value of VR is 1.0. Comparing now with the size distribution shown in Figure 19-85-4, it is clear that a large fraction of the aerosol mass will be in the form of particles with DF>20. On this basis, for the conditions assumed, the value of 20 was selected.

Table 19-85-3—Values Used in the DF Interpolations

Parameter	Value Used	Units	Reference / description
gamma	2.5	-	Parameter file, variable GSHAPE
g	9.81	ms^{-2}	MAAP source, S/R AMDIST
k	1.38E-23	J/K·molecule	Boltzmann constant
Assumed gas conditions:			
T_{gas}	900	K	Temperature Avg of hottest tube gas temp and saturation
P	1.00E+05	Pa	Pressure - depressurized SGs
rho	1000	kg/m^3	Density of aerosol particles, S/R FPTRAN
mu	3.37E-05	Pa·s	Viscosity of gas

Injection Mode:

There is a sensitivity to the parameter “injection mode”. For the steam generator, MAAP4.0.7 does not use downcomer injection, but rather a side vent injection mode. Repeating the interpolations using the side vent yields the data results in Figures 19-85-2 and 19-85-3. Particles above 1 micron are likely to be scrubbed with DF>20 provided the pool is at least approximately 1.8 m deep.

Conclusion:

For the characterization of the scrubbed SGTR source term (RC701), a DF of 20 has been applied to the unscrubbed SGTR source term (RC702) from MAAP4.0.7. It is noted that scrubbed SGTRs can only occur as a result of a tube rupture initiator, since induced ruptures require loss of feedwater (and hence dry SGs).

References for Question 19-85:

1. Electric Power Research Institute, MAAP 4 Computer Code Manual, May 1994

FSAR Impact:

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

Table 19-85-4—Non-dimensional Scalings for the Macroscopic Properties of a Sedimentation Aerosol (Reference 1)

Nondimensional Mass, M	Nondimensional Particle Volume, $\overline{V}$	Nondimensional Source, M_p
$\left(\dfrac{g\,h^4\,\gamma^9}{\rho^3\,\mu\,k_o\,\alpha^3}\right)^{1/4} m$	$\left(\dfrac{\gamma\,g\,\rho}{\alpha^{1/3}\,\mu\,k_o}\right)^{3/4} v$	$\left(\dfrac{\gamma^{11}\,\chi^4\,\mu\,h^8}{\alpha^5\,g\,k_o^3\,\rho^5}\right)^{1/4} \dot{m}_p$

with

$$k_o = \frac{4\,k\,T_{gas}}{3\,\mu}$$

where
- h - equivalent length for deposition
- γ - agglomeration enhancement factor
- α - particle porosity factor
- χ - Stoke's law correction factor
- v - particle volume
- m - aerosol concentration (kg m/m^3)
- $\dot{m}_p$ - aerosol source concentration (kg m/m^3 sec)
- k - Boltzmann's constant
- T_{gas} - gas temperature
- μ - gas viscosity

Table 19-85-5—Relation Between Non-dimensional Particle Size, VR and Particle Radius in m

VR	r (m)
1.00E-02	1.21E-07
1.00E-01	1.21E-06
1.00E+00	5.62E-07
1.00E+01	1.21E-06
1.00E+02	2.61E-06
1.00E+03	5.62E-06
1.00E+04	1.21E-05
1.00E+05	2.61E-05
1.00E+06	5.62E-05
1.00E+07	0.000121

Figure 19-85-1—Pool Scrubbing Data for Downcomer Venting

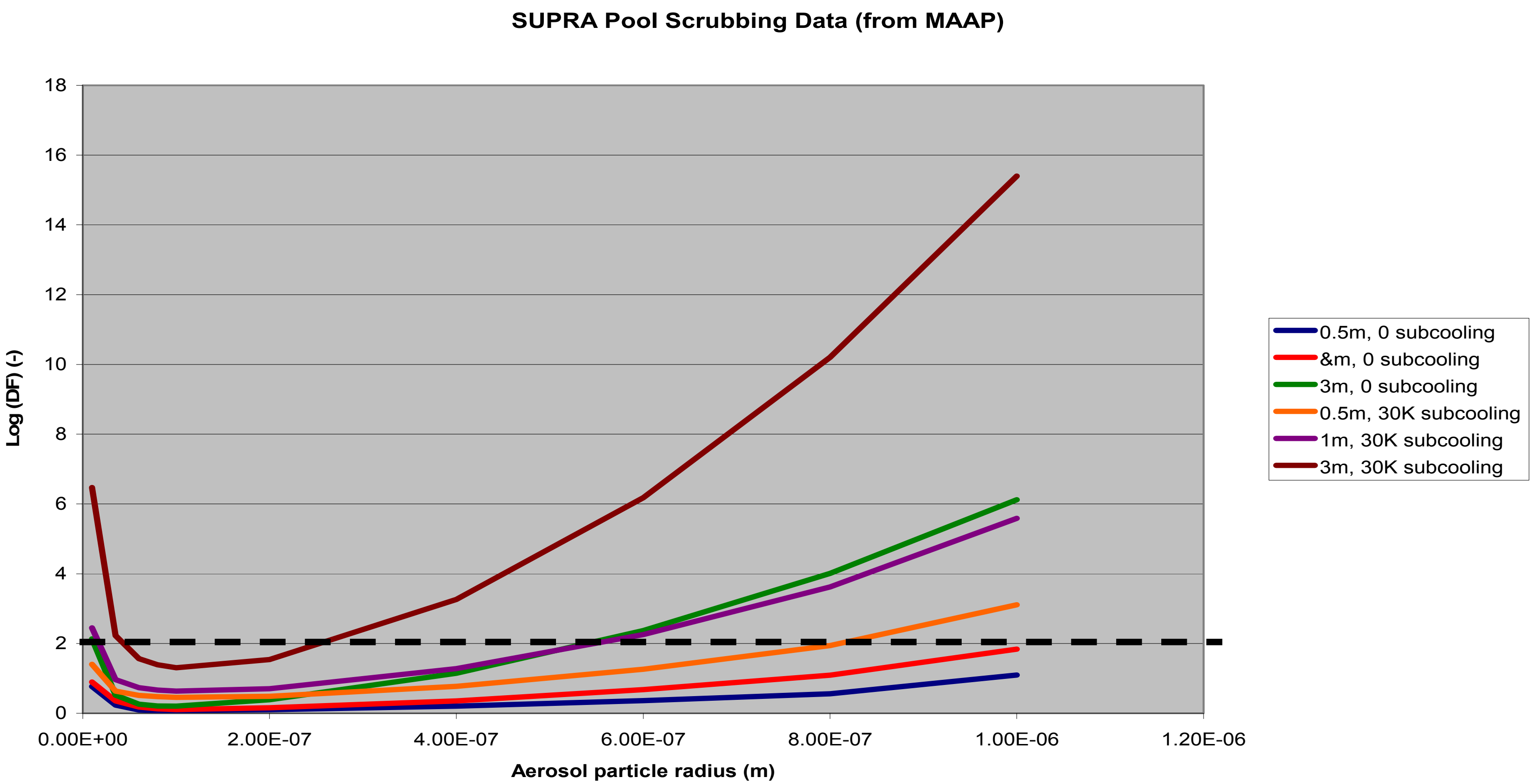


Figure 19-85-2—Pool Scrubbing Data for Side Venting

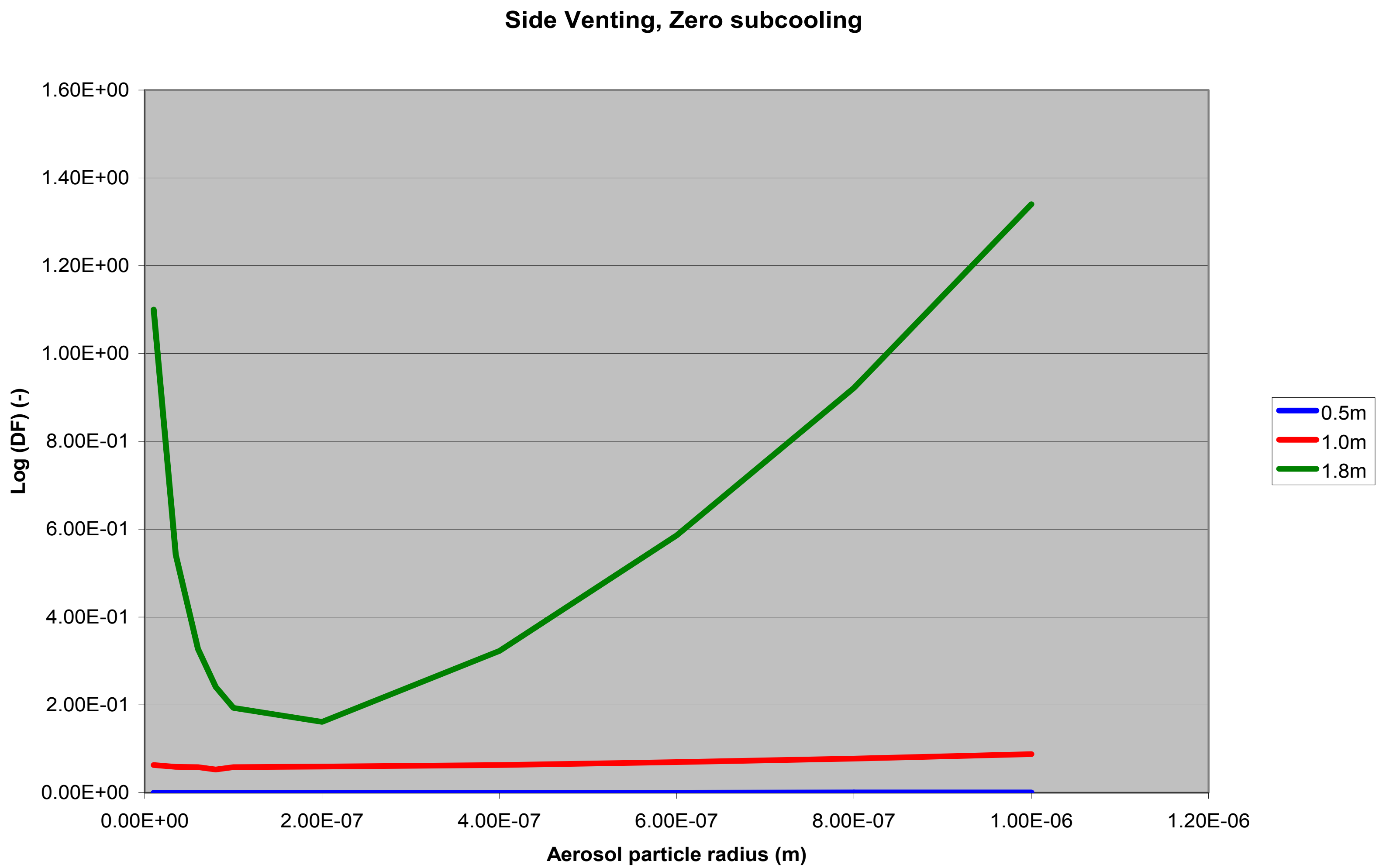


Figure 19-85-3—Log DF vs. Pool Depth – 1 micron particle – Side Venting

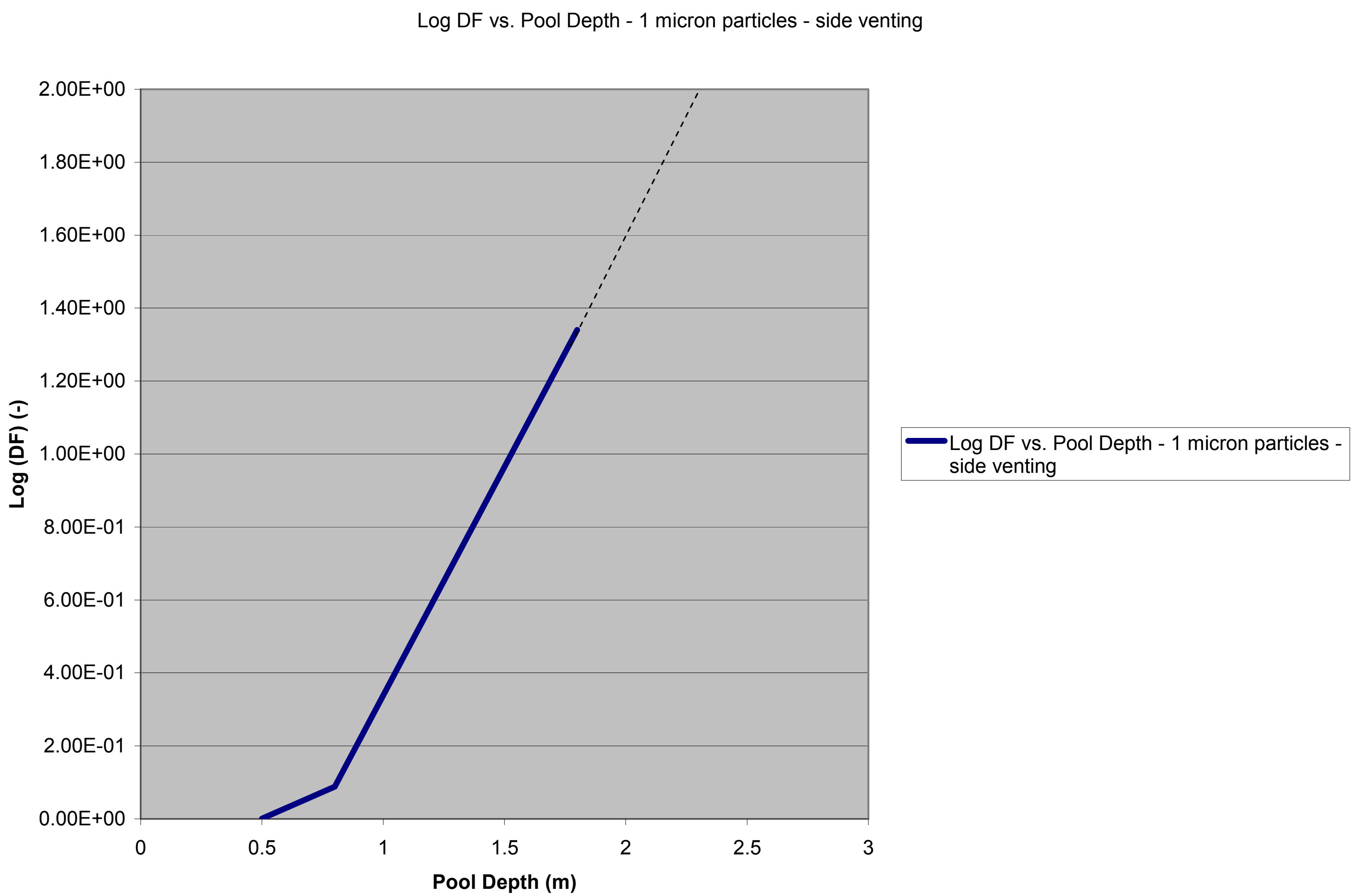
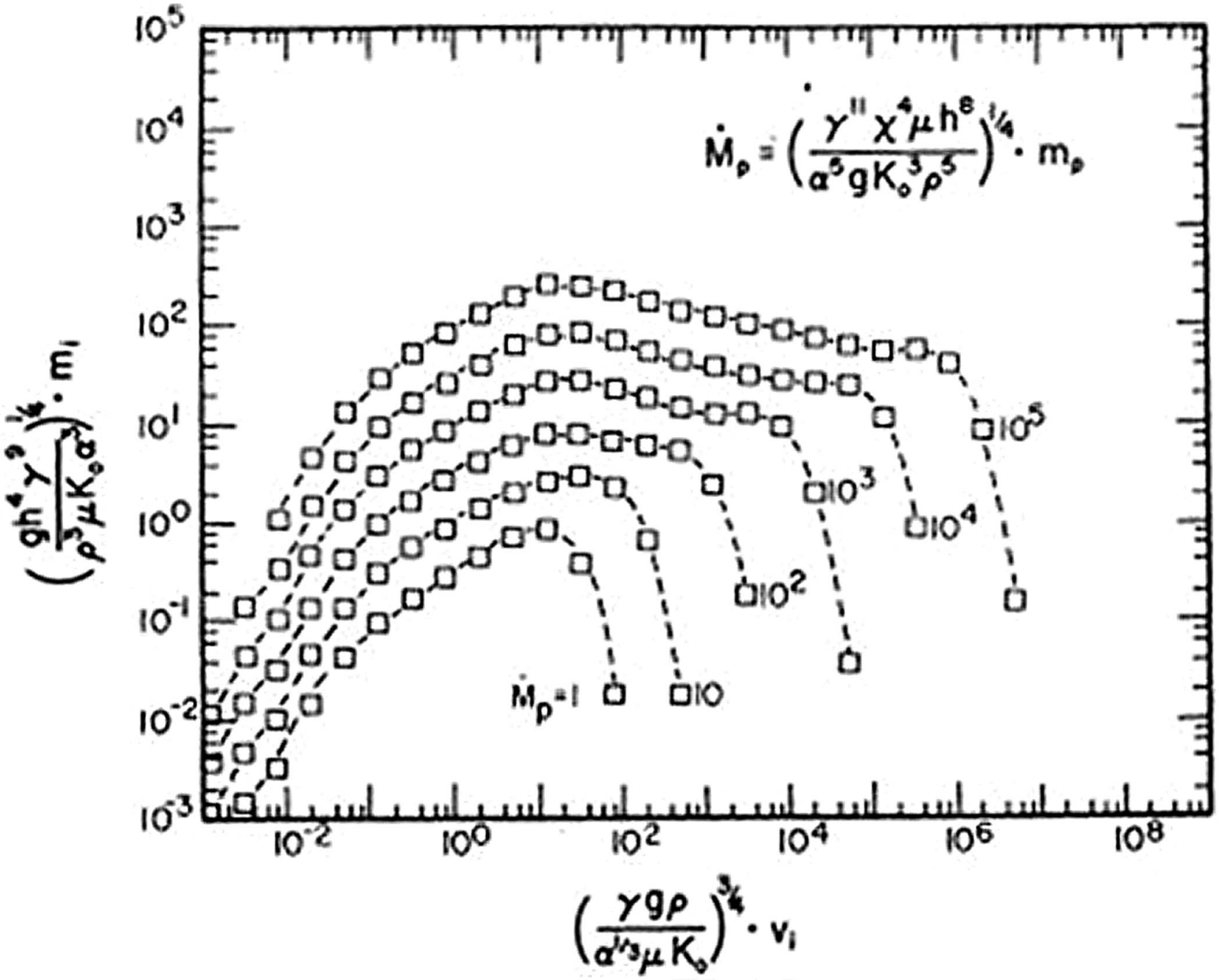


Figure 19-85-4—Steady State Aerosol Distribution for Different Non-Dimensional Source Rates (From Ref. 3)



Question 19-86:

A. Please provide the analysis of the Interfacing System LOCA release category 802 results reported in Table 19.1-20 of the FSAR.

B. Include the MAAP analysis results, and show the details of the fission product deposition in the fuel building and in the safeguards building. Include the analysis showing flooding levels for the break scenarios considered.

Response to Question 19-86:

Response to Question 19-86a - ISLOCA MAAP Results

A response to this question will be provided by August 8, 2008.

Response to Question 19-86b - Flooding Levels for Break Scenarios:

For the U.S. EPR, no credit is taken for scrubbing of fission products for any of the ISLOCA sequences.

Scrubbing of fission products is not credited when the analysis shows that the flooding from an ISLOCA is insufficient to fill the compartment where the break occurs, and there is insufficient water in the compartment to cause significant scrubbing of fission products from the release.

The ISLOCA analysis was examined to determine the location of the breaks for each of the ISLOCA scenarios. The location of the breaks was stated in terms of components within the system, and this information was correlated with the plant arrangement drawings to determine the building and elevation where releases from the break would occur.

To have effective scrubbing, the elevation of the break needs to be lower, by at least 6 feet, than the expected water level in the compartment. The amount of water released by the ISLOCA was obtained from the ISLOCA analysis, and the expected height of the water in the building was calculated, using the “gallons per foot” values for flooding levels in each building.

When this data is combined, there were no cases with significant submergence of the break location under the water level could be shown. The analysis is summarized in Table 19-86-1.

FSAR Impact:

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

Table 19-86-1—Analysis of Flooding Levels for ISLOCA Scrubbing

Pene-tration	IE Designator	System	Break Scenario	Size	CDF	Leakage Release Point: Building	Leakage Release Point: Room	Leakage Release Point: Elevation	Water Volume from ISLOCA	Gallons per foot in building	Height of Water from ISLOCA	References
002 and '004	ISL-CVCS-INJ	CVCS charging	CVCS charging flow enters the RCS at the cold leg of Loops 2 and 4 RCP seal injection	3-in 2-in	6.25E-12	Fuel Building		-11	677 867	29 032	-8.15	No. Less than 6 feet of water over break.
003	ISL-CVCS-REDS	CVCS letdown from Loop 1	Spurious operation (full-open) of the pressure reducing station can also over pressurize letdown line. (a break in the letdown line in fuel Bldg results in a LOCA outside of containment)	3-in	6.25E-12	Fuel Building		-11	677 867	29 032	-8.15	No. Less than 6 feet of water over break.
108	ISL-CVCS HPTR	Break in CVCS High Pressure Cooler	A break in the cooling tubes of the HP cooler results in high pressure RCS fluid entering the lower pressure Component Cooling Water System.	0.4" tubes	1.28E-10	CCWS Surge tank		+69 ft level in Safeguards building 1			Evaluation stops at 0 ft 0 inches	Non scrubbed release. No scrubbing credited at higher than 0 ft. 0 in.-
117/118	ISL-CCW RCP TB	RCP Thermal Barrier Cooling Coil	A break in the cooling tubes of the Thermal Barrier cooler results in high pressure RCS fluid entering the lower pressure Component Cooling Water System.	Two .51 inch internal diameter per tube	8.97E-13	CCWS Surge tank		+69 ft level in Safeguards building 1			Evaluation stops at 0 ft 0 inches	Non scrubbed release. No scrubbing credited at higher than 0 ft. 0 in
101/201/301/401	ISL_SIS LHSI	JNG – LHSI	LHSI connects to the RCS via its associated RCS cold leg. The MHSI and LHSI lines for a division merge to one injection line within containment prior to entering the RCS.	10-in at cold leg	3.45E-11	Safeguards Building 31/34 UJH	1/4UJH05 004	-16	677 867	37840	-13.59	Non scrubbed release. Less than 6 feet of water over break.
				10-in at cold leg	3.45E-11	Safeguards Building 32/33 UJH	2/3UJH05 004	-16	677 867	36850	-13.1	Non scrubbed release. Less than 6 feet of water over break.
102/202/302/402	ISL_SIS MHSI	JND – MHSI	MHSI connects to the RCS via its associated RCS cold leg. The MHSI and LHSI lines for a division merge to one injection line within containment prior to entering the RCS.	10-in at cold leg	3.45E-11	Safeguards Building	1/2/3/4UJH05 004	-16	677 867	37840	-13.59	Non scrubbed release. Less than 6 feet of water over break
103/203/303/403	ISL_SIS RHR	JNA – LHSI/RHR	RHR suction line: each SIS division has one suction line from its associated RCS hot leg.	10-in	7.87E-12	Safeguards Building	1/2/3/4UJH05 004	-16	677 867	37840	-13.59	Non scrubbed release. Less than 6 feet of water over break

Assumptions for water level calculations above:
SAB 1&4 are 37,840 gallons per foot, and SAB 2&3 are 36, 850 gallons per foot
Conversion factor for Gallons per cubic foot = 7.4805 gal/cubic foot
Conversion factor for fuel building– 900,000 gallons below grade = 29,032 gallons per foot

Question 19-87:

Please provide the assessment of the potential for in-vessel retention of core debris for the various core damage end states. Include a discussion of the potential for recovery during the following phases: core heatup to the onset of core melt; the onset of core melt to relocation of core debris into the lower head of the vessel; and relocation into the lower head of the vessel until vessel failure. Please relate the probability of successful quenching in each phase to the time when the primary system is depressurized. Provide the rates of hydrogen production from debris quenching during each phase. Please include results from applicable MAAP runs.

Response to Question 19-87:

A response to this question will be provided by August 8, 2008.

Question 19-88:

One of the uncertainties that can potentially impact the quantification of a number of severe accident phenomenological issues in the containment event progression is the expected mode of in-vessel crucible crust failure and melt relocation to the lower plenum. Please provide the technical bases for the quantification of this uncertainty in the U.S. EPR PRA. Furthermore, please discuss the details of the quantification process of accounting for the presence of the heavy reflector near the core boundary (a significant point of difference with conventional U.S. PWR designs).

Response to Question 19-88:

The examination of the mechanism of melt relocation was performed in the U.S. EPR as part of the analysis of the phenomena leading to containment loads at vessel failure.

This phenomena is discussed in the Response to Question 19-91.

FSAR Impact:

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

Question 19-89:

In the probabilistic analysis of vessel rocketing at the time of vessel melt-through, please provide the technical basis for quantification (or exclusion from the analysis, if not quantified) of the following forces:

- Jet forces of corium and RCS gases;
- Pressure differential between the reactor pit and the upper containment, including the effect on reactor pit pressure of blowdown, hydrogen combustion, and/or direct heat transfer from ejecting debris and pit atmosphere; and
- Vessel restraining forces, including the effect of uncertainties on pipe temperature at the RPV nozzles and its impact on yield strength, if applicable.

Response to Question 19-89:

A response to this question will be provided by August 8, 2008.

Question 19-90:

The U. S. EPR Final Safety Analysis Report, Pages 19.1-67 through 19.1-70 discuss the probabilistic evaluation of the potential impact of in-vessel and ex-vessel steam explosions. The estimated probability of containment failure due to an in-vessel steam explosion for high-pressure scenarios is by a factor of ~4 greater than the corresponding probability for low-pressure scenarios. This is appears to be counter- intuitive because the likelihood of triggering a steam explosion increases as the pressure is reduced. Please provide and discuss:

- The approach and the quantification process that support the estimated probabilities of containment and lower head failure.
- The range and uncertainties associated with the pour mass, composition, temperature, and location (relative to the reactor cavity wall), for assessment of the probability of containment failure due to ex-vessel steam explosions.
- The range and uncertainties associated with the water pool depth and temperature for the assessment of the probability of containment failure due to ex-vessel steam explosions.
- The potential impact of ex-vessel steam explosions on containment integrity (e.g., vibration of the reactor coolant system, the steam generators and the associated containment penetrations) under the condition that the lower vessel head may be submerged in water.
- The probabilities and uncertainties that quantify the potential for a significant water presence in the cavity at the time of vessel failure. This should also include a detailed description of the design features that limit this probability.
- The presence, if any, of drainage paths to and from the cavity, inadvertent system operation, diversion of outflows from various piping breaks and relief valve operations, and the provisions for detection and removal of water from the cavity during operation, with implications on ex-vessel steam explosions.

Please support the discussions of the probabilistic quantification process in terms of available analytic and experimental data.

Response to Question 19-90:

A response to this question will be provided by August 8, 2008.

Question 19-91:

Provide the basis for how the issue of over-pressurization of the reactor pit around the time of vessel breach was quantified in the Level-2 PRA. If the issue was decomposed into multiple sub-issues for purposes of quantification, also provide information the decomposition event tree and details of the technical basis for the quantification of the sub-issues.

Response to Question 19-91:

A response to this question will be provided by August 8, 2008.

Question 19-92:

In the assessment of HPME-induced DCH, please discuss the technical basis for the values assigned to the input parameters of subcompartment retention fraction and the cavity dispersion fraction. In addition, please discuss in the context of the U.S. EPR and the TCE model what is viewed as the “subcompartment” and what is the likely path or paths for transport of dispersed debris to the upper compartment.

Response to Question 19-92:

A response to this question will be provided by August 8, 2008.

Question 19-93:

Please discuss the maximum duration that has been considered in the assessment of long-term containment challenges for the Level-2 PRA. Please provide the technical basis for the selection of this duration and exclusion from consideration of possible later failure of the containment.

Response to Question 19-93:

A response to this question will be provided by August 8, 2008.

Question 19-94:

The containment event trees appear to show that, in the event of a large (3 in. or larger) containment isolation failure, or of containment failure resulting from in-vessel steam explosion, most of the remaining top events are not considered, including ex-vessel melt stabilization. For purposes of source term calculation, please clarify if MCCI is assumed to take place or not. If not, please justify how fission product release due to MCCI can be neglected under these conditions.

Response to Question 19-94:

Figures 19C-4 and 19C-8 of the U.S. EPR FSAR show the containment event trees for low pressure (and depressurized) and high pressure core damage end states, respectively.

In both CETs, the failure of containment isolation is represented by the top event #T1 CI "containment isolated". This event is resolved into three branches in both of the CETs.

The topmost branch, labeled with the number 3, is the branch where containment isolation is successful. The middle branch, labeled with the number 1, is the branch for sequences with a large containment isolation failure. The bottom branch, labeled with the number 2, is the branch for sequences with a small containment isolation failure.

As can be seen in both Figures, branch 1, the branch for sequences with a large containment isolation failure evaluated, results in sequences classified as Release Categories RC 202 and RC 203, where MCCI is taking place. These occur where top event #T3 MSXV "Melt Stabilization Ex-vessel" is assumed to have failed.

Therefore, MCCI is assumed to take place for large containment isolation failures.

FSAR Impact:

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

Question 19-95:

Please provide the assessment of the potential for containment failure in the U.S. EPR due to combustion of hydrogen and/or carbon monoxide.

a) Please discuss if and how the presence of carbon monoxide was also treated (for those sequences that may involve protracted MCCI).

b) Consider both deflagration and detonation loads, and the effectiveness of the PARs in limiting the concentrations of combustible gases in the containment. Provide the results from supporting MAAP calculations.

c) It is stated in Section 19.1.4.2.1.2 of the FSAR that, in evaluating hydrogen deflagrations for the Level-2 PRA, reference was made to MAAP results in order to determine the amount of hydrogen and oxygen consumed by PARs. Given that the amount of recombined hydrogen is subject to uncertainty (e.g., from uncertainties in timing of release to containment, distribution of gases in the containment, etc.), please discuss the extent by which the results of the deflagration analysis could be impacted by a reduced degree of PAR effectiveness.

d) Section 19.1.4.2.1.2 of the FSAR presents conditional probabilities of containment failure due to flame acceleration loads for a number of assessed cases. Please provide details on the formulation of the analysis by which these results were arrived at and the basis for the values of any input parameters used in performing this analysis.

e) Please provide the assumed value (or range of values) assigned to the fraction of in-vessel zirconium oxidation, and the technical basis for this (these) value (s).

f) In discounting potential containment loads from late hydrogen deflagrations in the Level-2 PRA, the statement is made that the containment would be steam-inerted in this time frame. Please provide the basis for this assumption.

g) Furthermore, a scenario can be envisioned within the containment event progression whereby SAHRS sprays are actuated in the late time frame, which could possibly result in a sudden de-inerting scenario with hydrogen present in the containment. (For example, the CET in Figure 19C-8 includes sequences with initial failure of steam suppression but with later credit to sprays for aerosol removal.) Please explain whether and how such a scenario is treated within the current Level-2 PRA.

h) For each PAR unit in the U.S. EPR containment, please provide its model or type and its location (with both room number and room description). Also, for each model or type of PAR present, please provide its nominal hydrogen recombination rate and the influence on recombination rate of pressure, hydrogen concentration, and any other factors. If any equivalent data on recombination rates of carbon monoxide are available, please provide these as well. These data are needed for the planned NRC confirmatory analyses.

Response to Question 19-95a:

Carbon monoxide is conservatively treated as being equivalent to hydrogen.

In the flame acceleration loads analysis, for accident sequences where carbon monoxide is produced due to molten corium-to-concrete interaction (MCCI), the concentration of CO is quantitatively addressed by adding this to the concentration of H_2 to obtain the overall flammable gas concentration, which is compared to the flame acceleration limit on a node by node basis for the whole transient calculation.

In the deflagration analysis, which is performed on a global basis, CO is not explicitly treated since a review of calculations showed that the global oxygen concentration would be below 5% before the production of significant amounts of CO occurred.

The consideration of carbon monoxide is only important in a small minority of hypothetical accident sequences in the U.S. EPR. Except in the very unlikely case of failure of the ex-vessel melt stabilization strategy, only small amounts of carbon monoxide production are expected. Refer to, for example, Figures A148 to A150 of Appendix A to this response, compared to Figures A178 to A174 of the same Appendix. The former set of figures show a transient in which the expected operation of the melt stabilization system occurs. This can be contrasted with the behavior observed in the latter set, where passive basemat flooding fails and the impact of ex-vessel hydrogen and carbon monoxide generation can be seen. The plots show total carbon monoxide + hydrogen concentration and the hydrogen concentration individually.

Response to Question 19-95b:

A summary of the overall hydrogen phenomenological evaluation is presented, including a summary of the results. The specific analysis of deflagration loads is also presented. The analysis of flame acceleration / detonation loads is presented as a Response to Question 19-95d.

FSAR Section 19.1.4.2.1.2 describes the different phenomena that may occur due to hydrogen generation during a severe accident: deflagration, deflagration to detonation transition (DDT) and flame acceleration. The effect of these phenomena on containment integrity is evaluated probabilistically: containment failure probabilities are derived for different cases and input to the containment event trees (CET).

Reference 1 shows that for flame acceleration and DDT, the effective pressure (i.e., the static pressure that would give a load equivalent to the dynamic load) is in the region of 1.5 to 2 times the pressure that would arise from a deflagration of the same amount of hydrogen. Direct initiation of a detonation is generally agreed to be extremely unlikely in a reactor containment. Detonation is therefore only postulated as a potential occurrence due to an accelerated flame and subsequent DDT. Since the loads caused by the two phenomena are similar, and flame acceleration is a precondition for detonation, DDT is not evaluated separately. Instead, only an evaluation of containment challenges due to flame acceleration is performed. The question of whether or not DDT occurs after the initiation of an accelerated flame is considered irrelevant, as the occurrence or otherwise of DDT is not expected to have a significant impact on the total probability of containment failure if the containment already experiences loading from a fast flame. The approach taken means that no credit is taken to reduce the probability of containment failure on the basis of criteria of sufficient geometrical size (e.g., seven lambda criteria).

To support this phenomenological evaluation, a series of MAAP cases were run to simulate a wide range of scenarios. The MAAP runs are described in Table 19-82-1 in the Response to Question 19-82; the results plots are shown in Appendix A.

The deflagration load analysis is performed as follows:

- A bounding quantity of hydrogen generated is estimated based on a review of MAAP analyses
- The load resulting from the deflagration is conservatively evaluated using the adiabatic isochoric complete combustion (AICC) model.
- The probability of containment failure is then obtained by comparing the load to the containment fragility curve. The resulting containment failure probabilities (CFP) are summarized in Table 19-95b-1.
- The detailed analysis of deflagration loads is presented below in the response to this question.

The assessment of flame acceleration loads is performed as follows:

For each scenario, the chemical composition (H_2, CO, steam) of each of the containment 23 nodes at each time step is obtained from the MAAP cases. Nodes that are susceptible to experience accelerated hydrogen flame are identified.

For susceptible nodes, the probability of flame acceleration is calculated. Given occurrence of an accelerated flame, the resulting load is calculated using the adiabatic isochoric complete combustion (AICC) method and a multiplicative factor of 2 as explained above. If flame acceleration does not result in containment failure, local effects (loss of recombiners) are also considered.

For each scenario the probability of containment failure is assessed by comparing the load to the containment fragility curve. The resulting containment failure probabilities are summarized in Table 19-95b-1.

The detailed analysis of flame-acceleration loads is presented in the Response to Question 19-95d.

Table 19-95b-1 shows the summary of the containment failure probabilities due to hydrogen combustion for the different scenarios considered. For incorporation into the CETs, hydrogen loads are separated by time frame as follows:

- Timeframe 1 (TF1): between the onset of core damage and the time of vessel failure.
- Timeframe 2 (TF2): close to the time of vessel failure.
- Timeframe 3 (TF3): long term, after the time of vessel failure.

Table 19-95b-1—Summary of Containment Failure Probabilities due to Hydrogen Phenomena for Various Scenarios

Scenario / time frame	CFP from deflagration loads	CFP from flame acceleration loads
Transients at high pressure, discharge via pressurizer during in-vessel phase (TF1)	2.0E-06	1.6E-02
Transients at high pressure, discharge via hot leg rupture during in-vessel phase (TF1)	1.38E-04	1.25E-03
Transients at high pressure, at the time of vessel failure (TF2)	N/A	5.6E-03
Hydrogen generation due to initial concrete ablation post-vessel failure (TF3)	N/A	4.5E-04
Hydrogen generation due to MCCI post-vessel failure, with recombiners functioning at 75% efficiency (or damaged by a previous event) (TF3)	N/A	1.0E-04 5.0E-04 if recombiner damage has occurred

Response to Question 19-95b - Part 2—Analysis of Deflagration Loads

The deflagration assessment is performed on a global basis using a bounding assessment of expected hydrogen masses (uncertainty analysis on hydrogen production is therefore not performed). The AICC method is used to evaluate the global pressure resulting from the deflagration. Details of hydrogen distribution within containment during the transients considered therefore are not of interest. The main parameters considered in the global deflagration assessment are as follows:

- In-vessel hydrogen production.
- Ex-vessel hydrogen production.
- Steam concentration.

Consumption of hydrogen and oxygen by recombiners is considered by reference to the MAAP analyses performed. Consumption of hydrogen by random hydrogen burns at lower concentrations is conservatively ignored.

In-vessel hydrogen production is obtained from the in-vessel zirconium oxidation predicted in a series of MAAP runs, as shown in the Response to Question 19-95e.

During the pre-vessel failure period, MAAP calculations predict some short-term peaks in the baseline containment pressure. A review of the pre-vessel failure period in cases c1_1, c1_1a, c1_1d, c1_6, c1_7 and c1_8 indicate that the highest steam concentration during that period is 0.55, occurring at around 8250 seconds in case c1_1a. In case c1_7 this peak occurs slightly earlier in the transient. The corresponding peak baseline pressure is 36 psia. This peak steam concentration is insufficient to inert the containment for deflagration. This peak of containment pressure and steam concentration occurs a few hundred seconds before the onset of rapid hydrogen generation, implying that hydrogen masses in containment at this time would be well below the values previously discussed. Referring to Appendix A, Figure A1 shows the development of the total mass of hydrogen in containment for 2 inch seal loss of coolant accident (LOCA) cases with varying degrees of recombiner efficiency. The maximum peak observed is for the case of 50% recombiner effectiveness, this peak being slightly less than 600 kg of hydrogen. It is noted that the seal LOCA cases only show 57% zircaloy oxidation in-vessel.

A conservative evaluation of the peak hydrogen burn pressure can be made using the AICC method and the conditions previously discussed. It is assumed that the peak steam concentration / baseline pressure conditions in containment occur in conjunction with the maximum estimated zircaloy oxidation of 82%, and, conservatively 50% recombiner efficiency is assumed. As the maximum peak of hydrogen mass seen in the seal LOCA cases above is 600 kg but this corresponds to zircaloy oxidation of only 57%. An adjustment is therefore made to represent 82% oxidation conditions. The hydrogen mass corresponding to the difference between 57% oxidation and 82% oxidation is 15% of 1520 kg = 228 kg. This results in an upper bound maximum hydrogen mass in containment of 828 kg.

An AICC evaluation assuming 55% steam in containment and 828 kg of hydrogen gives a peak containment pressure of 94 psia. Evaluation of this pressure as compared to the U.S. EPR containment fragility curve results in a probability of containment failure of 2E-06.

A review of MAAP cases c1_2 and c3_4, which model an induced hot leg rupture, shows that in these cases, although hydrogen production is not increased, the mass of hydrogen present in containment does increase due to the rapid release of hydrogen from the reactor coolant system (RCS) at the time of hot leg rupture. This is shown in Appendix A, Figure A2, compared against case c1_1e. It can be seen that the peak hydrogen mass is about 300 kg higher in case c3_4. An AICC evaluation assuming 1128 kg of hydrogen in containment was therefore performed to represent the loading in this case. The resulting peak pressure was 111.7 psia, which corresponds to a containment failure probability of 1.38E-04. It is noted that the hydrogen peak mass is a short term phenomena; the recombiners quickly act to reduce this mass as the transient progresses.

The two probabilities derived above show the risk of containment failure due to hydrogen deflagration before vessel failure is small and it is noted that some bounding choices were made in their evaluation. The risk posed by hydrogen deflagration at later time frames (at the time of vessel failure or after vessel failure) is shown in the Response to Question 19-95f and Response to Question 19-95g to be negligible and is therefore discounted.

Response to Question 19-95c:

The uncertainty in the efficiency of the passive autocatalytic recombiners (PAR) is considered in the analysis. The MAAP runs performed to support analysis of deflagration (see Response to Question 19-95b) and flame acceleration (see Response to Question 19-95d) used different assumptions regarding the efficiency of the PARs (100%, 75%, 50%). Table 19-95d-1 (in the Response to Question 19-95d) shows a list of the MAAP runs used in this analysis, including the assumed PAR effectiveness for each run.

The possibility that one or several PARs are failed by local consequences of flame acceleration is discussed in the Response to Question 19-95d.

Response to Question 19-95d:

The assessment of flame acceleration was performed as follows:

1. A set of scenarios were chosen for evaluation. These scenarios are listed in Table 19-95d-1.

2. MAAP analyses corresponding to these scenarios were reviewed to identify periods and containment nodes for which mixture conditions were inside the limits where flame acceleration was possible.

3. For the containment nodes and times selected in step 2, an evaluation was performed to support the assignment of probability values for flame acceleration and consequent containment failure.

In step 2, all 27 nodes of each MAAP calculation were evaluated at each timestep to check whether or not the mixture conditions were susceptible or not to flame acceleration. The hydrogen concentration limit for flame acceleration was evaluated according to the recommendations and the detailed description of Reference 2 Appendix B. In Appendix A, Figure A5 to Figure A144 show the nodal histories of the steam, oxygen and resulting hydrogen concentration limit for flame acceleration for the MAAP cases listed in Table 19-95d-2. Note that the histories for nodes 1, 5, 10 and 23 are always shown in these figures to facilitate review and understanding of the behavior of the calculated limit. Histories for other nodes are presented if the node hydrogen concentration exceeded the flame acceleration limit at any time. Note that calculated conditions in node 27 exceeded the flame acceleration limit in several cases; however, this node is not presented or assessed because the concentrations seen in the node are fictitious and result from a dummy air space required by to allow the MAAP model to function correctly.

The behavior of node 5 is taken as an example for cases c1_1e, c3_4 and c3_13 (see Appendix A, Figure A21, Figure A113, and Figure A135).

In case c1_1e, the flame acceleration limit concentration goes to 1.0 periodically during the first 10,000 seconds. The value of 1.0 is used to show that the node is inert for flame acceleration; the observed behavior in this case during this period is due to period release of steam into containment via the pressurizer relief valves, causing high steam concentrations above 55% and hence steam inerting for flame acceleration. The corresponding peaks of the steam concentration can be clearly seen in the figure. This node also becomes steam inert in the longer term, after vessel failure; again the correspondence of the flame acceleration limit and the behavior of the plotted steam concentration can be clearly seen. In this case, it is noted that node 5 remains steam inert despite the operation of the severe accident heat removal system (SAHRS) spray.

In case c3_4, as in case c1_1e, the flame acceleration limit concentration goes to 1.0 periodically during the first 10,000 seconds due to steam inerting. This node also becomes steam inert in the longer term, after vessel failure. Again, the correspondence of the flame acceleration limit and the behavior of the plotted steam concentration can be clearly seen in Appendix A Figure A113. The operation of SAHRS sprays then reduces the steam concentration, leading to non-inert conditions. However, in the long term inerting occurs again due to low oxygen concentrations (the drop in oxygen concentration can be seen on Appendix A Figure A113).

In case c3_13, only a single steam inerting peak is seen, followed later by inerting due to oxygen depletion. This case is a 2 inch seal LOCA and steam inerting due to steam release from the primary circuit via the pressurizer relief valve (cycling at high pressure) is not seen.

The hydrogen concentration histories, plotted together with the calculated flame acceleration limit, are shown in Appendix A, Figure A145 to Figure A209. The sum of the hydrogen and carbon monoxide concentrations is also shown on these plots. For simplicity, carbon monoxide is conservatively treated as hydrogen in the evaluations performed.

As an example, node 5 is again chosen, for c1_1e, c3_4 and c3_13.

In case c1_1e, Appendix A, Figure A149, low hydrogen concentrations are generally seen throughout the transient. However, a short term spike is seen at around 10,000 seconds, and this spike just crosses the flame acceleration limit. The spike is due to the release of hydrogen into the compartment via the pressurizer relief valves, at a time just after a period of steam inertion in the node (due to steam also released via the pressurizer relief valve). According to Table 19-95d-2, this hydrogen spike lasts for less than 6 seconds. This sort of short term hydrogen spiking is seen in many high pressure transients calculated.

In case c3_4, Appendix A, Figure A196, hydrogen concentrations are well below the limit for flame acceleration for most of the transient. However, two points are worthy of comment. There is a short term hydrogen spike corresponding to release of hydrogen via the pressurizer relief valves. The discharge is into node 5, via node 3, as shown in Appendix A, Figure A195. As shown in Table 19-95d-2, this spike is seen at 13,252 seconds in node 3 (initial discharge) and propagates to node 5 at around 13,300 seconds. In the longer term it is noted that hydrogen and carbon monoxide concentrations increase, starting at around 75,000 seconds, but as discussed in the previous section on deflagrations at this time the containment is inert due to oxygen depletion and the probability of air replenishment is considered negligible.

In case c3_13, a hydrogen peak concentration occurs at the time of vessel failure (approximately 31,000 seconds). In this case, very little hydrogen has been released to containment during the in-vessel phase (the transient is a 2" seal LOCA, and core damage occurs at pressures below the system setpoint, reducing flows from the RCS into containment).

The peak at vessel failure is large, but short (just 1 timestep in MAAP). In the longer term, from around 5.5 hours after vessel failure, the hydrogen (and CO) concentration in this node increases due to combustible gas generation from MCCI. In this node (5), the flame acceleration limit is not exceeded (despite the calculation being performed with only 75% recombiner efficiency). However, generally in cases with a dry spreading area (no passive flooding) and MCCI, it is seen that in node 1 (the spreading area) the flame acceleration limit can be exceeded if the containment is not steam inert and before the time at which inerting occurs due to oxygen depletion.

In addition to the above specific examples, the more general behavior of the hydrogen concentration and flame acceleration susceptibility is as follows.

It is seen across all the calculations summarized in Table 19-95d-2 that there are four instances during which mixture conditions can indicate susceptibility to flame acceleration:

(i) During the in-vessel period for high pressure sequences with core damage occurring at the system setpoint, the discharge from the RCS via the pressurizer relief valves into nodes 3 and 5 can result in very short term peak hydrogen concentrations in these nodes. These peaks are not seen in seal LOCA cases, because the release of hydrogen is split between the pressurizer and the seal LOCA (which is modeled in all four loops) but node 3 is susceptible in a 2" small LOCA as the release is concentrated into a single LOCA location.

(ii) In the case of a hot leg rupture occurring nodes 6 and 10 are also susceptible. For example, see 1_2 at 13,712 seconds (HLR at 13,711s).

(iii) Various nodes are susceptible to flame acceleration at the time of vessel failure, these being node 8 (pit) as well as nodes 6, 7, 10 and 23. The latter nodes show a very short period of susceptibility, being just a few seconds. Furthermore the susceptibility of nodes 6, 7, 10 and 23 is not consistently seen in the calculations and this is believed to be due to the timestep chosen by MAAP (i.e., in many cases the timestep is too large to capture the very short period of susceptibility).

(iv) In the case of 75% recombiner efficiency, no steam inerting, ongoing MCCI (dry spreading area due to no passive flooding but with SAHRS sprays reducing the containment steam concentration), concentrations in node 1 (spreading area) can cross the flame acceleration limit. A very short spike of hydrogen in node 1 is also seen in cases c2_5 and c1_7d. For evaluation, this case will be separated into three cases - (iv)a which represents the short spike scenario as seen in cases c2_5 and c1_7d, (iv)b in which there is 100% recombiner efficiency but MCCI due to failed passive flooding, (iv)c which is as (iv)b except that it is applicable to sequences for which an outcome of an earlier CET node has lead to reduced recombiner efficiency (damage). In Appendix A, Figure A210 compares the ablation behavior of cases c1_1 (base case, high pressure), c2_5 (Large LOCA), c3_1a, and c3_13 (dry spreading area cases). It is noted that c2_5 presents more concrete erosion than c1_1 before melt stabilization and it is this short term behavior which leads to a spike of hydrogen concentration in node 1 for case c2_5.

The probability of flame acceleration and consequent containment failure or other damage is evaluated according to the following method:

The probability of flame acceleration occurring with only local effects (no containment failure) is P1 x P2 x P3 x (1 - P4 x P5), where:

- P1 is the probability of no continuous burning on release from RCS or from the ex-vessel corium pool. This probability depends on the presence of a proximate ignition source and the conditions of the release (e.g., turbulent jet or other).

- P2 is the probability of ignition during the period of high concentration given no continuous burn. (This probability value is reduced if the period of exposure to a high concentration is shorter.)

- P3 is the probability of flame acceleration given ignition and this value is dependent on the geometry of the node. For flame acceleration to occur, it is necessary for the geometry to be favorable. If the geometry is not favorable, then flame acceleration is generally assessed as unlikely, even if the mixture characteristics would support this.

P4 and P5 are defined below.

The total probability of containment failure is also evaluated by expanding the preceding expression, as follows:

Probability of containment failure = P1 x P2 x P3 x P4 x P5

Where P1 to P3 are as defined above and,

- P4 is the probability that an enhanced load is experienced at the containment boundary. This depends on the location of the initial node (proximate or not to the containment boundary).

- P5 is the probability of containment failure given the enhanced load. This probability is evaluated by calculation of the AICC burn pressure and the use of the multipliers taken from Reference 1 as discussed in the Response to Question 19-95b, Part 1. A multiplier of 2.0 is used in the evaluations and the resulting enhanced load is compared to the U.S. EPR fragility curve to generate the final value of P5.

Table 19-95d-2 presents the evaluation of the individual probabilities P1 to P5 and the overall results for the cases (i) to (iv) previously described.

Table 19-95d-1—Summary of Supporting MAAP Analyses for Analysis of Hydrogen Deflagration and Flame Acceleration

Case	Description / comments	PARS	In-vessel zircaloy oxidation	Onset of hydrogen productio n (sec)	Vessel failure (sec)	Mixture sensitive to flame acceleration (sec)
C1_1	High pressure CDES - SBO - no induced rupture or depressurization	100%	67%	11100	24106	Node 3. 13375.8 to 13396.7 Node 5. 13383.2 to 13397.9 Node 8. 24166.1 to 24166.1
C1_1a	High pressure CDES - SBO - no induced rupture or depressurization. Variant modified FFRIC factors.	100%	48%	10028	23575	No
C1_1a a	High pressure CDES - SBO - no induced rupture or depressurization. 2nd variant with modified FFRIC factors.	100%	66%	10050	23575	No
C1_1e	High pressure CDES - LOFW - no induced rupture or depressurization. Spray on at 5 bar.	100%	63%	6435	19007	Node 3. 8015.25 to 8721.86 Node 5. 8024.74 to 8032.3 Node 8. 19067.0 to 19067.0
C1_2	High pressure CDES - SBO - induced rupture occurs, no safety injection available.	100%	72%	11100	32242	Node 10. 13712.7 to 13712.7.* Node 3. 13375.8 to 13396.7 Node 5. 13383.2 to 13712.1 Node 6. 13712.1 to 13712.1*
C1_3	High pressure CDES, operator depressurization of RCS, no safety injection available. Spray on at 12 hours (= 43200 sec)	100%	46%	14911	28364	Node 1. 39346.0 to 39346.0 Node 8. 34784.9 to 35504.9
C1_6	Seal LOCA 0.6" with SBO. No active SAHRS/sprays.	100%	63%	10825	25324	Node 3. 12842.8 to 12858.3 Node 5. 12848.6 to 12857.7

Table 19-95d-1—Summary of Supporting MAAP Analyses for Analysis of Hydrogen Deflagration and Flame Acceleration

Case	Description / comments	PARS	In-vessel zircaloy oxidation	Onset of hydrogen production (sec)	Vessel failure (sec)	Mixture sensitive to flame acceleration (sec)
C1_7	Seal LOCA 2" with SBO. No SAHRS/sprays. Accumulator available.	100%	65%	5656	44818	Node 8. 50279.5 to 50939.5
C1_7d	Seal LOCA 2", LOCCW IE. EFW available, fast cooldown at 1800s.	100%	56%	21667	46734	Node 1. 60837.2 to 61197.2
C1_8	Small LOCA, 2", safety injection not available. EFW available. Accumulators available.	100%	59%	6972	33122	Node 3. 7611.39 to 7667.2
C2_1	Seal LOCA 2" - SBO IE. SAHRS spray at 5 bar.	75%	53%	5656	45244	Node 8. 50645.3 to 51305.3
C2_1a	Seal LOCA 2" - SBO IE. SAHRS spray at 5 bar.	50%	57%	5656	44486	Node 8. 49947.2 to 50727.2
C2_2	Seal LOCA 2" - SBO IE. SAHRS spray at 5 bar. Sensitivity to mixing dampers (fail to open).	100%	57%	5684	44819	Node 8. 50279.8 to 50939.8
C2_3	Seal LOCA 2" - SBO IE. SAHRS spray at 5 bar. Sensitivity to mixing dampers (fail to open) + degraded pars.	50%	57%	5684	44486	Node 8. 49947.2 to 50727.2
C2_5	Low pressure CDES - Large LOCA. No SI, no accumulators, no active SAHRS/Sprays.	100%	14%	99	6519	Node 1. 15374.4 to 16386.0 Node 8. 11749.0 to 14602.7

Table 19-95d-1—Summary of Supporting MAAP Analyses for Analysis of Hydrogen Deflagration and Flame Acceleration

Case	Description / comments	PARS	In-vessel zircaloy oxidation	Onset of hydrogen production (sec)	Vessel failure (sec)	Mixture sensitive to flame acceleration (sec)
C3_1a	High pressure CDES - SBO. No passive or actives SAHRS/sprays.	100%	66%	11717	24783	Node 3. 13253.1 to 24798.4 Node 5. 13314.3 to 24808.4 Node 6. 24808.4 to 24808.4 Node 7. 24798.4 to 24808.4 Node 8. 24808.4 to 24868.6 Node 10. 24808.4 to 24808.4 Node 23. 24788.4 to 24788.4
C3_1c	High pressure CDES - SBO. No passive or actives SAHRS/sprays. Debris confined to 50% of spreading area.	100%	66%	11721	25927	Node 3. 13448.0 to 25939.1 Node 5. 13458.4 to 25949.2 Node 6. 25949.2 to 25949.2 Node 7. 25939.1 to 25959.2 Node 8. 25959.2 to 25999.9 Node 10. 25959.2 to 25959.2
C3_1d	High pressure CDES - SBO. No passive or actives SAHRS/sprays. Note ablation behavior not clear. Variant of 3_1a with high CaO concrete.	100%	66%	11717	24783	Node 3. 13253.1 to 24798.4 Node 5. 13314.3 to 24808.4 Node 6. 24808.4 to 24808.4 Node 7. 24798.4 to 24808.4 Node 8. 24808.4 to 24868.6 Node 10. 24808.4 to 24808.4 Node 23. 24788.4 to 24788.4

Table 19-95d-1—Summary of Supporting MAAP Analyses for Analysis of Hydrogen Deflagration and Flame Acceleration

Case	Description / comments	PARS	In-vessel zircaloy oxidation	Onset of hydrogen production (sec)	Vessel failure (sec)	Mixture sensitive to flame acceleration (sec)
C3_2	High pressure CDES - SBO. No passive or actives SAHRS/sprays. LHSI recovered 5 hours after vessel failure.	100%	68%	11720	25405	Node 3. 13364.6 to 25423.2 Node 5. 13374.6 to 25423.2 Node 6. 25433.3 to 25433.3 Node 7. 25423.2 to 25433.3 Node 8. 25433.3 to 25483.6 Node 10. 25433.3 to 25433.3 Node 23. 25413.2 to 25413.2
C3_4	High pressure CDES - SBO. SAHRS sprays followed by active SAHRS at 5 bar. Induced rupture simulated at 13711 seconds.	100%	72%	11717	29893	Node 3. 13253.1 to 13324.3 Node 5. 13314.3 to 13324.3
C3_6	SBO with 0.6" seal leak. SAHRS spray at 5 bar, followed by switch to active reflood.	100%	68%	11376	25091	Node 3. 25094.1 to 25104.2 Node 5. 25094.1 to 25114.2 Node 6. 25114.2 to 25114.2 Node 7. 25094.1 to 25114.2 Node 8. 25114.2 to 25164.3 Node 10. 25114.2 to 25114.2
C3_7	SBO with 0.36" seal leak. No SAHRS spray, but active cooling at 5 bar.	100%	63%	11623	23844	No
C3_8	2" seal LOCA with SBO. SAHRS spray at 5 bar, switch to active reflood.	100%	58%	5685	44443	Node 8. 49559.1 to 51109.1

Table 19-95d-1—Summary of Supporting MAAP Analyses for Analysis of Hydrogen Deflagration and Flame Acceleration

Case	Description / comments	PARS	In-vessel zircaloy oxidation	Onset of hydrogen production (sec)	Vessel failure (sec)	Mixture sensitive to flame acceleration (sec)
C3_13	2" seal LOCA with SBO. SAHRS spray at 5 bar, switch to active reflood.	75%	59%	9036	31251	Node 1. 60098.7 to 81818.7 Node 3. 31254.9 to 31265.0 Node 5. 31265.0 to 31265.0 Node 7. 31265.0 to 31265.0 Node 8. 31275.0 to 31305.3
C3_16	2" seal LOCA with SBO. SAHRS spray at 5 bar, switch to active reflood. Combustion occurs at vessel failure.	100%	59%	9036	30864	Node 8. 30885.2 to 30905.2

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1 0.5	P2 0.25	P3 0.25	P4 0.5	P5 1.0	Results
(i)	Transients at high pressure, discharge via pressurizer during in-vessel phase. Susceptible nodes are 3 and 5. Node 3 represents the Level 1 Lower Equipment Rooms. Node 5 represents the Level 1 Middle Equipment Rooms.	It is discussed in the literature that when hydrogen is released as a turbulent jet this may represent an ignition source in which case continuous burning (rather than accumulation and subsequent burning) would be likely. However, the phenomena of turbulent jet ignition itself is difficult to quantify so a probability of 0.5 is assigned (uncertain behavior).	The period of exposure to ignition with a susceptible mixture for flame acceleration ranges from 10 seconds to up to 700 seconds in these cases, according to MAAP. At the shorter end of this time scale ignition during the critical period (given no continuous burns) would be very unlikely. However, at the higher end of the scale ignition is believed to be more likely, thus an overall probability of 0.25 is assigned for ignition occurring (no ignition is considered	These volumes are considered to have geometries that are susceptible to flame acceleration (presence of obstacles, etc). However, the peak concentrations seen only just exceed the flame acceleration limit which reduces the assigned probability value for P3 and hence it is judged that no flame acceleration is the more likely outcome. A	These nodes are not proximate to the containment boundary. It is therefore uncertain if the enhanced load from an accelerated flame or DDT would be seen at the containment boundary.	An AICC evaluation was performed at 12% hydrogen concentration (i.e., representative of the local node conditions). The resulting load was 1.08 MPa absolute or 0.98 MPa relative. According to the multipliers of the effective load from an accelerated flame or DDT may be up to two times the deflagration loading. The effective pressure is therefore 2x 0.98 +0.1 MPa = 2.06 MPa. An evaluation against the	

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1 0.5	P2 0.25	P3 0.25	P4 0.5	P5 1.0	Results
			overall the more	value of P3 =			
			likely outcome)	0.25 is therefore assigned.		U.S. EPR fragility curve indicates that this effective pressure corresponds to a containment failure probability of 1.0.	P1xP2xP3x(1-P4xP5) = 0.016 P1xP2xP3xP4xP5 = 0.016

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1 0.5	P2 0.25	P3 0.25	P4 0.5	P5 1.0	Results
(ii)	Transients at high pressure, discharge via hot leg rupture during in-vessel phase. Susceptible nodes are nodes 6 and 10 at the time of hot leg rupture. Node 3 and 5 are susceptible at an earlier time as in case (i). Node 6 represents the Level 2, 3, 4 Middle Equipment Rooms. Node 10 represents the Level 2, 3, 4 upper equipment rooms.	The evaluation of case (i) is applied. Although it is recognized that discharge via a hot leg rupture would result in different release characteristics to the case of release via the pressurizer, the level of uncertainty in both cases is considered equivalent, resulting in the same probability assignment.	Given no continuous ignition, ignition during the period of high concentration is considered very unlikely. The MAAP analysis for c1_2 shows susceptibility during a single timestep only. In case c3_4, for which induced rupture is also simulated, the flame acceleration limit is not exceeded in nodes 6 and 10.	The evaluation of case (i) is adjusted upwards by a factor 2 (P3 = 0.5) due to the higher hydrogen concentration s seen in nodes 6 and 10 at the time of hot leg rupture.	The evaluation of case (i) is applied.	The evaluation of case (i) is applied.	P1xP2xP3x(1-P4xP5) = 0.00125 P1xP2xP3xP4xP5 = 0.00125

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1 0.5	P2 0.25	P3 0.25	P4 0.5	P5 1.0	Results
(iii)	At the time of vessel failure for high pressure sequences a number of nodes have mixture conditions susceptible to flame acceleration - nodes 8 (pit), 6, 7 (Level 1 upper equipment rooms), 10, 23 (Staircase South) Note that in sequences with failure of the vessel at low pressure only the pit (node 8) exceeds the flame acceleration limit. Loads from accelerated flames in that volume are considered of little consequence and so flame acceleration at the time of a low pressure vessel failure is discounted as a threat for the global or local containment structures.	There is an ignition source present at the time of vessel failure and it is expected that the source would be co-entrained with the combustible gases at this time. Therefore, continuous burning of the gases is considered likely. Thus, P1 = 0.1 (unlikely).	Given no continuous ignition, ignition during the period of high concentration is nevertheless considered likely in this case because an ignition source (corium) would be present. Therefore P2 = 0.9 is assigned.	The assignment is made by comparison to case (i). This would lead to a value of 0.25, however, a review of the MAAP analyses for the high pressure vessel failure cases shows that high concentrations are not consistently predicted in nodes 6, 7, 10 and 23. The existence of high concentrations in these nodes is therefore subject to uncertainty and a probability of 0.5 (of the flame acceleration limit not being	The evaluation of case (i) is applied.	The evaluation of case (i) is applied.	P1 xP2 x P3 x (1 - P4 x P5) = 0.0056 P1 xP2 x P3 x P4 x P5 = 0.0056

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1 0.5	P2 0.25	P3 0.25	P4 0.5	P5 1.0	Results
(iv)a	With recombiners functioning at 100% efficiency MAAP occasionally predicts that some rapid initial concrete ablation may generate sufficient hydrogen quickly enough for Node 1 (which represents the spreading room and steam chimney) to experience a short period of mixture sensitivity to flame acceleration. This is the case even with successful passive flooding and but no long term active SAHRS/sprays.	The probability of continuous burning of released hydrogen (and carbon-monoxide) is assessed as higher than in case (iii). This is because the hydrogen is being released directly from the ignition source (debris) rather than being co-entrained with it. The less dynamic situation is judged to make continuous ignition easier.	The evaluation of case (iii) is applied.	Node 1 is characterized by an open geometry. A susceptible geometry is expected to be a necessary condition for flame acceleration to occur and hence the assigned probability P3 is reduced compared to the nodes assessed in cases (i), (ii), (iii). It is however noted that the hydrogen concentrations reach the range 15% to 20% in this node and for this reason a value of P3= 0.05 rather than a lower value is assigned.	Node 1 is proximate to the containment boundary, therefore is an accelerated flame occurs it is expected that enhanced loads would be experienced at the containment boundary.	The evaluation of case (i) is applied.	P1xP2xP3x(1-P4xP5) = 0.0 P1xP2xP3xP4xP5 = 0.00045

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1 0.5	P2 0.25	P3 0.25	P4 0.5	P5 1.0	Results
(iv)b	With recombiners functioning at 75% efficiency MAAP predicts that if MCCI occurs due to the failure of passive flooding (dry spreading area) concrete ablation may generate sufficient hydrogen such that Node 1 experiences an extended period of mixture sensitivity to flame acceleration. However, this effect is not seen if recombiners function at 100% efficiency in such scenarios.	The evaluation of case (iv)b is applied.	As the predicted period of mixture sensitivity (if it occurs) is expected to be long, the probability of ignition is increased to 1.0 (certain).	The assigned probability value is reduced compared to case (iv)a since best estimate MAAP analyses with 100% recombiners do not predict mixtures sensitive to flame acceleration. The results of this evaluation will be applied in the CET for sequences with no recombiner damage.	Node 1 is proximate to the containment boundary, therefore is an accelerated flame occurs it is expected that enhanced loads would be experienced at the containment boundary.	The evaluation of case (i) is applied.	P1xP2xP3x(1-P4xP5) = 0.0 P1xP2xP3xP4xP5 = 0.0001

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1	P2	P3	P4	P5	Results
		0.5	0.25	0.25	0.5	1.0	
(iv)c	As case (iv)b except for use in scenarios where recombiner damage has occurred. (See evaluation of P3).	The evaluation of case (iv)b is applied.	The evaluation of case (iv)b is applied.	The probability value from case (iv)a is used without reduction. The results of this evaluation will be applied in the CET for sequences with recombiner damage having occurred at previous CET nodes.	The evaluation of case (iv)b is applied.	The evaluation of case (i) is applied.	P1xP2xP3x(1-P4xP5) = 0.0 P1xP2xP3xP4xP5 = 0.0005

Table 19-95d-2—Calculation of Flame Acceleration Probability and Overall Flame Acceleration/DDT Containment Failure Probability

Case	Description	P1 0.01	P2 1.0	P3 0.05	P4 1.0	P5 1.0	Results
(iv)c	As case (iv)b except for use in scenarios where recombiner damage has occurred. (See evaluation of P3).	The evaluation of case (iv)b is applied.	The evaluation of case (iv)b is applied.	The probability value from case (iv)a is used without reduction. The results of this evaluation will be applied in the CET for sequences with recombiner damage having occurred at previous CET nodes.	The evaluation of case (iv)b is applied.	The evaluation of case (i) is applied.	P1xP2xP3x(1-P4xP5) = 0.0 P1xP2xP3xP4xP5 = 0.0005

Response to Question 19-95e:

The fraction of in-vessel zirconium oxidation is estimated based on the representative MAAP runs performed in support of this analysis. The fraction of zirconium oxidation is the basis for evaluating the mass of hydrogen generated in-vessel.

In-vessel zirconium oxidation can be affected by the characteristics of the accident sequence under consideration. There are also uncertainties related to the code calculation. Table 19-95e-1 summarizes the results of the MAAP analyses that were reviewed with respect to in-vessel zirconium oxidation. For each type of sequence, one or several MAAP cases are observed in order to obtain the range of oxidation fraction for that type of accident.

Table 19-95e-1—Summary of Key MAAP Analyses for Deflagration Loads

Sequence Characteristics	Relevant U.S. EPR MAAP Analyses	In Vessel Zirconium Oxidation Fraction
High-pressure CDES, no induced rupture or depressurization	C1_1, c1_1a, c1_1aa, c1_1e	67%, 48%, 66%, 63% Range: 48% to 67%
High-pressure CDES, induced rupture occurs, no safety injection available.	C1_2	72%
High-pressure CDES, operator depressurization of RCS, no safety injection available.	C1_3	46%
Seal LOCA CDES	C1_6, c1_7 (0.6" & 2" break) Degraded PARS: 2_1, 2_1a (2" break total)	65%, 63% 64%, 70% Range: 63% to 70%
2"LOCA, safety injection not available	C1_8	59%
Low pressure CDES (Large LOCA)	C2_5	14%

The following conclusions can be drawn from Table 19-95e-1:

For high pressure sequences which remain at high pressure, predicted in-vessel zircaloy oxidation is in the range 48% to 67%. The top of this range increases to 72% when case c1_2 (induced hot leg rupture) is also considered.

Zircaloy oxidation is reduced in cases where there is an early emergency depressurization (case c1_3).

In transients that progress to vessel failure more quickly (6 inch and greater LOCAs - cases c1_8 and c2_5), zircaloy oxidation is reduced. These cases present a maximum oxidized fraction of 59%.

Based on these conclusions, two groups of transients are considered sufficient to represent the variation in in-vessel hydrogen production:

- High pressure cases, including cases with induced hot leg rupture. The range to be assumed in the analysis for these cases will be 48% to 82% zircaloy oxidation. The rationale for this range is that the minimum value of 48% was generated in sensitivity study (case 1_1a) with parameter variations aimed at generating sensitivity results for in-vessel hydrogen production. An average value for the base cases c1_1 and c1_1e is 65% oxidation. If 65% were taken as the mid-point value with 48% as a minimum, this gives a symmetric upper bound of 82% (=65% + 17%). It is noted that this range of 48% to 82% zircaloy oxidation is very similar to the range presented in Figure 7 of Reference 1 (50% - 80%, read from figure).
- Low pressure cases with a 6 inch (or larger) break in the RCS at the onset of core damage. These are represented conservatively by assuming the MAAP value of 14% oxidation as a lower bound and taking the upper bound as 14% + 17 % = 33%.

Values for the mass of hydrogen present in the containment are inferred form the oxidation fractions previously discussed. The masses of zircaloy is calculated based on the mass of zircaloy present in the U.S. EPR core and the corresponding mass of hydrogen generated can then be obtained by considering the chemical reaction for zircaloy oxidation in steam. The chemical reaction for the production of hydrogen due to the oxidation of zircaloy by reaction with steam is:

$Zr + 2H_20 \text{ -> } Zr0_2 + 2\ H_2$

This reaction equation implies that 1 kg-mole of Zr will produce 2 kg-moles of H2. In terms of masses, this implies that 90 kg of Zr (1 kg-mole) would produce 4 kg (2 kg-moles) of H2. The mass of zircaloy in the U.S. EPR core is 34,200 kg. This implies that 4/90 x 34,200 kg = 1520 kg of H2 would be generated by 100% zircaloy oxidation. The ranges of zircaloy oxidation previously discussed therefore correspond to ranges of 730 kg to 1246 kg (48% to 82%) and 212 kg to 501 kg (14% to 33%).

Response to Questions 19-95f and 19-95g:

The following section supports discounting containment loads from late hydrogen deflagration. Depending on the scenario, one or several of the following factors support discounting long term containment loads due to hydrogen deflagration: steam inertion, hydrogen depletion due to the action of the recombiners and oxygen consumption. SAHRS spray actuation in particular is addressed. Results from the MAAP cases supporting this discussion are plotted in Appendix A.

The baseline pressure in containment just before vessel failure is after the short-term peaks previously discussed, by which time the baseline pressure is, according to various MAAP analyses, around 36 psia. It is noted that while many MAAP analyses show that at the time of vessel failure the mass of hydrogen in containment is greatly reduced due to the action of recombiners (e.g., case c1_7), others (e.g., case c1_1) show that significant amounts of hydrogen can be released at around this time, leading to similar masses being present as at the earlier peak which arises at the time of onset of rapid zircaloy oxidation. Evaluation of the AICC peak pressure corresponding to the time of vessel failure is therefore based on the same maximum amount of hydrogen identified in the cases above. Using the AICC spreadsheet calculation with this input mass of hydrogen and assuming 36 psia baseline pressure (45% steam concentration) gives an absolute peak pressure of 86 psia. Evaluation of this pressure against the U.S. EPR containment fragility curve gives zero probability of failure. On this basis failure of the containment due to hydrogen deflagration at the time of vessel failure is discounted. The reasons for the conclusion that containment failure is more likely from burns during the pre-vessel failure period are: (i) the baseline pressure is expected to be higher during that period; (ii) due to the action of the recombiners, the mass of hydrogen involved in a burn at the time of vessel failure will not be substantially higher than in the pre-vessel failure period.

In the short term after vessel failure (up to around 3 hours after vessel failure), without the operation of SAHRS sprays or active cooling (but with passive flooding of the debris), the baseline pressure in containment will begin to rise again. It can reach (and pass) conditions corresponding to 60% steam concentration. At the same time, the action of recombiners continues, mitigating the short term peak hydrogen global concentrations that arise at the time of vessel failure. As shown in Appendix A, Figure A3, the action of the recombiners reduces the mass of hydrogen much more quickly than the increase in steam concentration, meaning that the peak pressure from hydrogen burning in the first few hours after vessel failure is not expected to exceed the peak calculated in the previous paragraph and containment failure from hydrogen burning in this initial period is therefore also discounted. Furthermore, for accident sequences in which there are no sprays or active cooling, the containment atmosphere will be steam inert for all later times, precluding containment failure by hydrogen deflagration in these cases.

In cases where the containment is not steam inert, or deinerting occurs, the threat from a global deflagration can also be discounted. While the sprays or active reflooding maintains the baseline pressure low, it would only be possible for a global deflagration to generate loads greater than those already analyzed (and discounted) at vessel failure if there was an increase in hydrogen concentration. While hydrogen (and/or carbon monoxide) can theoretically be produced in a postulated core-concrete interaction scenario, this production of combustible gas can only happen on a long term timescale, by which time the containment atmosphere is inert due to lack of oxygen, even if steam de-inerting occurs. This behavior can be inferred from the nodal oxygen and steam concentrations seen in plots corresponding to cases:

- c1_1e (Appendix A, Figure A19 to Figure A24)
- c2_1 (Appendix A, Figure A55 to Figure A59)
- c2_1a (Appendix A, Figure A60 to Figure A64)
- c2_3 (Appendix A, Figure A70 to Figure A74)
- c3_4 (Appendix A, Figure A111 to Figure A115)
- c3_8 (Appendix A, Figure A128 to Figure A132)
- c3_13 (Appendix A, Figure A133 to Figure A139)

It can be seen in Appendix A, Figure A4, that for two cases (c3_8 and c3_13) with long term hydrogen generation, the long term hydrogen generation starts after around 100,000 seconds, by which time the nodal oxygen concentration histories show that the oxygen concentration has dropped below 5%, implying inert conditions for deflagration.

To complete the argument discounting long term deflagration threats in the U.S. EPR containment it is necessary to argue that oxygen leakage back into containment can be neglected for the following reasons:

- Direct leakage from the atmosphere back into the containment can be discounted because the containment pressure is higher than atmospheric pressure. Even with the sprays operating, a rapid depressurization of containment is not expected and this is reflected in the MAAP analyses performed For example, see cases c1_3, c3_16 (in which a burn occurs at vessel failure).
- The only identified potential source of air is the instrument air system. There are no sources of pure oxygen which could inject into containment. This system has two compressors, each capable of delivering 770 standard cubic feet per minute (21.8 m^3/minute) of air at 130 psig (equivalent to 0.896 MPa relative or 0.997 MPa absolute or 9.84 atmospheres). Air density at 1 atmosphere is 1.199 kg/m^3, therefore at 9.84 atmospheres it is 1.199 x 9.84 = 11.8 kg/m^3. The instrument air system could therefore pump 21.8 m^3/minute per compressor, which is equivalent to 257 kg per minute (11.8 kg/m^3 x 21.8 m^3/minute). The containment initially contains about 120,000 kg of air. Therefore it would require1200 kg of oxygen, or 6000 kg air to raise oxygen levels from by 1% (e.g., from 4% to 5%). The instrument air system operating at its maximum capacity with two compressors could do this in about 12 minutes, if the system leaked air into the containment. Such leakage is considered of negligible probability, as explained below.

It is noted that the oxygen depletion behavior is also important for flame acceleration/DDT. These phenomena are discussed in Response to Question 19-95d.

The compressed air system supplies air to the Reactor Building by two paths, the instrument air system and the service air system. The service air system is normally isolated from the Reactor Building by manually operated locked closed valves, and therefore cannot contribute oxygen to the containment atmosphere. The instrument air system will supply air to the instrumentation inside the containment. This air supply is automatically isolated by the protection system from the loads inside containment during accident conditions.

To be vulnerable to re-introduction of oxygen, this containment isolation line would have to fail to isolate, which is a low probability event. It is noted that containment isolation is modeled in the U.S. EPR CET, and it is expected that the instrument air isolation system would have a large degree of common signals and power supplies with this system. Thus, in sequences where these support functions are failed, it is expected that the containment would be "not isolated", leading to a release and making long term air injection an irrelevant event. This implies that isolation valve failure to close probabilities would dominate the reliability of the instrument air isolation function, vastly reducing any potential for dependent failure of the isolation function with the incoming CD sequences from Level 1. In addition, although the size and capacity of the instrument air line is not available, it is not expected that the normal leakage from the instrument air lines would provide compressed air at the rate necessary to deinert the containment. This implies that in addition to any failure of the instrument air system to isolate, there would have to be a coincident catastrophic failure of instrument air piping inside containment for sufficient leakage to occur. Given the basic principles of design engineering, it is judged highly unlikely that such a failure would occur simultaneously with a common cause failure of the containment isolation valves. Finally, it should be noted that all analyses of the recombiner capacity, including degraded recombiners, shows that the recombiners are able to cope with typical hydrogen generation rates (only rapid peak releases require significant time to mitigate), which strongly suggests that if sufficient oxygen leakage did occur into the containment, its consequence would be to cause recombination, rather than accumulation, of generated hydrogen. In conclusion, on the basis of the preceding discussion, long term hydrogen deflagration can be neglected.

Response to Question 19-95h:

Information concerning each passive autocatalytic recombiner (PAR) unit in the U.S. EPR containment, including its model or type and its location (with both room number and room description) is given in Table 19-95h-1. Also provided is nominal hydrogen recombination rate and the influence on recombination rate of pressure, hydrogen concentration (Table 19-95h-1, Note 1), along with efficiency at varying pressures of the large recombiner (Figure 19-95h-1) and small recombiner (Figure 19-95h-2).

Table 19-95h-1—U.S. EPR Combustible Gas Control System: Passive Autocatalytic Recombiner Location

Figure 19-95h-1—Efficiency of Large Passive Autocatalytic Recombiner (FT1-1500T)

Figure 19-95h-2—Efficiency of Small Passive Autocatalytic Recombiner (FT1-380T)

References for 19-95:

1. Breitung, W., Chan, C.K., Dorofeev, S.B., Eder, A., Gelfand, B.E., Heitsch, M., Klein, R., Malliakos, A., Shepherd, J.E., Studer, E., Thibault, P. “Flame Acceleration and Deflagration to Detonation Transition in Nuclear Safety,” (State-of-the-Art Report by a Group of Experts), OECD Nuclear Energy Agency, NEA/CSNI/R(2000)7, August 2000.
2. Gauntt R.O., “Uncertainty analyses using the MELCOR Severe Accident Analysis Code,” OECD/NEA Workshop Proceedings on Evaluation of Uncertainties in Relation to Severe Accidents and Level 2 probabilistic safety analysis. Aix-en-Provence, pages 398-419, 7-9 November 2005.

FSAR Impact:

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

Question 19-96:

Please provide the following data in support of the NRC severe accident confirmatory analyses:

1. The Annulus Ventilation System conditions that were used in the MAAP 4.07 analyses:
    - Accident operating pressure,
    - The total volumetric flow rate through both fans,
    - The Fuel and Safeguards Building ventilation exhaust rate, and
    - The elevation of the system with respect to the bottom of the reactor pressure vessel.
2. The Shield Building data that were used in the MAAP 4.07 analyses:
    - Volume and elevation entries,
    - Surface area of the Shield Building, and
    - The aerosol sedimentation area.
3. Diagrams showing the details of the core melt stabilization system (i.e., the reactor cavity, the cavity gate, and the melt discharge channel into the spreading area).
4. The six-group delayed neutron fractions, decay constants, and the neutron generation time applicable to analysis of steam line break inside containment. In addition:
    - The Doppler temperature coefficient (and the associate reference temperature)
    - The moderator temperature coefficient (and the associate reference temperature)
    - The boron coefficient
    - Separately, the total volume concentration of boron inside the chemical and volume control system and the Emergency Boron System (EBS) tanks.
    - The well-mixed boron concentration in the reactor coolant system during the cycle (prior to any accidents).
    - The actuation setpoint and rate of injection from the EBS tank.

Please state the fuel cycle period (e.g., EOL) corresponding to the aforementioned data as applicable.

Response to Question 19-96:

The annulus ventilation system conditions that were used in the MAAP 4.07 analyses are provided in Table 19-96-1.

Table 19-96-1—Annulus Ventilation System Conditions

Response to Question 19-96(2):

The Shield Building data that were used in the MAAP 4.07 analyses are provided in Table 19-62-2.

Table 19-62-2—Shield Building Data

	Volume (m^3)	**Height (m)**	**Bottom Elevation (m)**	**Surface Area (m^3)**	**Sedimentation Surface (m^2)**
Annulus	13,800	60.51	-4.3	13,700	2,776

Response to Question 19-96(3):

Diagrams showing the details of the core melt stabilization system (i.e., the reactor cavity, the cavity gate, and the melt discharge channel into the spreading area) are shown in Figures 19-96-1 and 19-96-2.

Figure 19-96-1—Overview Reactor Cavity, Cavity Gate, and Melt Discharge Channel

Figure 19-96-2—Reactor Cavity Gate and Cavity Support Frame

Response to Question 19-96(4):

The requested neutronics parameters were derived for the U.S. EPR design-basis main steam line break scenario. That analysis assumes the event occurs at end-of-cycle (EOC) conditions with highly negative moderator and Doppler temperature coefficients. The cooldown of the core may insert sufficient positive reactivity to overcome the shutdown margin and return the reactor to a critical state. Design-basis analyses also assume that the highest-worth rod cluster control assembly (RCCA) is (1) stuck in the fully withdrawn position (which minimizes the shutdown margin) and (2) located above the affected core sector (which weights the reactivity feedbacks of that sector more heavily). The particular data is:

- Six-group delayed neutron fractions taken from the 1973 ANS standard fission product data are used.
- The total nominal delayed neutron fraction at end of cycle (EOC) 0.005151. This value is used to convert reactivity to dollars.

Total delayed neutron fraction divided by prompt neutron lifetime at EOC nominal is 214.0834 1/s. Prompt neutron lifetime can be evaluated from this value and the stated value for delayed neutron fraction above.

Table 19-96-3 provides Doppler fuel temperature coefficient data.

Table 19-96-3—Doppler Fuel Temperature Coefficient Data

Reactor Power (%)	K-effective	Reactivity (pcm)	Reactivity ($)
0	0.904584	0	0.00
5	0.900684	-479	-0.93
10	0.898028	-807	-1.57
15	0.895949	-1065	-2.07
20	0.894307	-1270	-2.47
25	0.892969	-1438	-2.79
30	0.891807	-1584	-3.07

Table 19-96-4 provides moderator temperature coefficient data.

Table 19-96-4—Moderator Temperature Coefficient Data

Mod Temp (°F)	Mod Density (kg/m^3)	k-effective	Unbiased Defect (pcm)	Reactivity Bias (pcm)	Biased Defect (pcm)	Biased Defect ($)
600.0	691.475	0.902830	-215	-72	-287	-0.56
595.6	697.303	0.904584	0	0	0	0.00
578.0	719.688	0.911046	784	293	1077	2.09
500.0	797.159	0.930628	3094	1111	4205	8.16
400.0	870.018	0.945259	4757	1623	6380	12.39
300.0	926.199	0.954426	5773	1962	7735	15.02
200.0	969.983	0.960664	6453	1969	8423	16.35
100.0	999.710	0.964801	6900	1645	8544	16.59

Table 19-96-5 provides boron coefficient data.

Table 19-96-5—Boron Coefficient Data

Boron Concentration (ppm)	Boron Density (kg/m^3)	K-effective	Reactivity (pcm)	Reactivity ($)
0	0.00000	0.904584	0	0.00
400	0.27892	0.874498	-3803	-7.38
800	0.55784	0.846970	-7520	-14.60
1200	0.83676	0.821642	-11159	-21.66
1600	1.11568	0.798237	-14728	-28.59

Both the emergency boron system (EBS) tanks and the boric acid storage tanks have a minimum boron concentration of 7000 ppm. The boric acid storage tanks are part of the reactor boron and water make-up system (RBWMS) which feeds borated water to the RCS through the chemical and volume control system (CVCS).

At the end of a cycle, the boron concentration in the RCS will approach 0 ppm.

The EBS is a manually actuated system used to maintain the reactor sub-critical during accident conditions and has a minimum flowrate of 49 gpm and a maximum flowrate of 55.4 gpm (density = 61.68 lbm/ft^3).

FSAR Impact:

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

Question 19-97:

Please describe the analytical studies pertaining to the prevention of high-pressure severe accident scenarios, as well as the severe accident uncertainty analyses pertaining to primary circuit depressurization. Include discussions of accident management prior to RCS depressurization, assumptions on PDS valve discharge capacity, and operator response margins. Include the results of the DCH assessment with delayed depressurization, vessel rocketing, and induced RCS rupture, as well as the survivability of depressurization valves under harsh severe accident conditions.

Response to Question 19-97:

RPV failure at high internal pressure is of importance to severe accident risk due to the high pressure melt ejection (HPME) that can result in direct containment heating (DCH). Even though such a failure is physically unlikely, an objective of the U.S. EPR severe accident response strategy is to convert high pressure core melt sequences into low pressure sequences with high reliability so that the likelihood of a high pressure vessel breach is acceptably low. For the U.S. EPR, this is achieved through two dedicated severe accident depressurization valve trains that are part of the primary depressurization system (PDS). This design is described in greater detail in FSAR Tier 2 Section 19.2.3.3.4.1.

Severe accident management guidelines for the U.S. EPR will include a statement instructing an operator to initiate a primary-side depressurization by manually opening the PDS valves prior to reaching a high core exit temperature. This manual action has inherent uncertainty. The MAAP4-based uncertainty analysis presented in FSAR Tier 2 Section 19.2.4 explicitly addresses several event progression uncertainties including those associated with the prevention of high-pressure severe accident scenarios. The uncertainty in PDS valve actuation is addressed by sampling the core outlet temperature setpoint from 1200 – 1832°F (650 – 1000°C) associated with the manual action and an additional delay of up to 15 minutes. Uncertainty in PDS valve discharge (approximately 550 lb/s of saturated steam at design pressure) was not treated as an uncertainty parameter, as this is a functional requirement of the design.

The PDS plays a major role in the prevention of HPME, vessel rocketing, and induced RCS rupture. Probabilistic analyses on DCH, vessel rocketing, and induced RCS rupture appear in FSAR Section 19.1.4.2.1.2. The following pertain to the sensitivity of these analyses to PDS performance:

- DCH – RCS pressure at reactor pressure vessel failure was evaluated from the MAAP4-based uncertainty analysis in FSAR Tier 2 Section 19.2.4.4.3. The PDS efficiently reduces RCS pressure below the design load for the reactor cavity (290 psia). Similarly, the work of Tutu, Ginsberg, and Fintrok (Reference 12 in FSAR Tier 2 Section 19.2.7) was cited that identifies an RCS pressure of 145 psia at RPV failure as the threshold under which HPME (and therefore DCH) is unlikely to occur. While a few event samples in the MAAP4-based uncertainty analysis reported RCS pressure above 145 psia, all sampled events were bounded by an RCS pressure at RPV failure of 200 psia. This is still a rather low maximum RCS pressure at RPV failure and compliments the probabilistic result provided in FSAR Tier 2 Section 19.1.4.2.1.2.
- Vessel Rocketing - FSAR Tier 2 Section 19.1.4.2.1.2 also reports on a vessel rocketing assessment, concluding that the vessel is well contained by structural restraints for

bounding scenarios (pressure and mass ejected). As with the DCH analysis, the RCS is expected to be significantly depressurized by reactor vessel rupture; thus, eliminating the vessel rocketing potential.

- Induced RCS rupture (i.e., hot leg and steam generator tubes) – RCS integrity is sensitive to the temperature of the steam/hydrogen mixture expected to flow through the RCS. Fluid temperature in these scenarios is very sensitive to the timing of PDS valve actuation. As such, analyses considered a bias of about 540°F.

A functional requirement for the PDS is its survivability during the more likely severe accident scenarios, including its performance under the environmental conditions anticipated for the PDS. Component sensitivity to temperature is the principle concern for the PDS. Based on results from the uncertainty analysis and a follow-on performance analysis, the PDS is expected to be exposed to a steam/hydrogen mixture near 1200°F for on-time event response and below 1700°F in a situation in which PDS valve actuation is significantly delayed.

FSAR Impact:

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

Question 19-98:

Please describe the consequences of overpressurizing the reactor pit during blowdown following RPV failure. Include a discussion of potential impacts on severe accident melt stabilization.

Response to Question 19-98:

The reactor cavity of the U.S. EPR is designed to withstand structural loads up to 290 psia. Six walls located with 60° azimuthal separation aligned radially from near the melt plug to the reactor cavity walls are provided to protect the sacrificial concrete that conditions melt released from a failure of the reactor pressure vessel. The principle role of the reactor cavity is to temporarily retain molten corium for a period sufficient to accumulate the complete release. Failure of the temporary melt retention feature introduces event progression uncertainty related to the plant's designed response to a severe accident. It is this uncertainty that is to be avoided.

To undermine the role of the reactor cavity in the melt stabilization process, the damage would be realized as either an early failure of the melt plug and gate or a blockage preventing corium release into the spreading area. To address the uncertainty associated with potential reactor cavity damage, the functional requirement that the structure withstand 290 psia was established. As shown in the MAAP4-based uncertainty analysis, maximum RCS pressure at RPV failure is expected to be below 200 psia. As such, actual damage to the reactor cavity is unlikely and the reactor cavity's role in temporary melt retention is minimal.

Several tangential event progressions are conceivable, including various fuel-coolant interactions scenarios and delayed stabilization. Considering the reliability of the U.S. EPR's PDS coupled with the designed strength of the reactor pit at 290 psia, the potential for serious consequences remain a remote possibility. Even minor deviations to the design ex-vessel event progression would not result in a situation in which any failure of the temporary melt retention phase leads to a failure to ultimately cool the core melt and/or preserve containment integrity. Rather, the core debris coolability and severe accident melt stabilization safety issue is best resolved by demonstrating a reliable mechanism for rapid and efficient heat removal.

FSAR Impact:

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

Question 19-99:

Please describe the base cases and uncertainty analyses of relevant severe accident scenarios that document the calculation matrix of process studies that examine the performance of the U.S. EPR severe accident response features. Explain how the MAAP 4.0.7 code was used in the analyses, and which scenarios and their variations were modeled. Please include, as well, analyses of the main steam line break inside containment scenario, since its cutsets dominate the large release frequency.

Response to Question 19-99:

As identified in FSAR Tier 2 Section 19.2.4.2.2 – Relevant Scenarios, the base case scenarios are as follows:

- Loss of offsite power with Seal LOCA (LOOP SS).
- Loss of offsite power with a low pressure end state (LOOP PL).
- Loss of offsite power with a high pressure end state (LOOP TR).
- Loss of balance of plant (LBOP TR).
- Small LOCA (2 to 8.5-inch diameter) (SLOCA SL).

These events were derived from a scenario identification analysis in which the dominant severe accident sequences from the initiating events appearing in FSAR Table 19.1-103 were either explicitly selected to represent a class of event types or dispositioned. Steam line breaks were dispositioned for the purposes of the analyses appearing in FSAR Tier 2 Section 19.2, and are thus not explicitly analyzed as a base case for the relevant scenarios. The principle assumption that leads to the severe consequences associated with steam line breaks is a coincident severe ATWS. The inclusion of the ATWS condition is considered an additional failure that moves this event out from the suite of relevant scenarios. Without this assumption, steam line breaks leading to core damage are events in which core damage results from the loss of secondary heat sink. As such, steam line break contributions are assigned to LBOP TR category.

In general, the normal response to each of these events can be described as a “feed and bleed” approach. The threshold for whether a particular event results in core damage is the balance between the safety injection system (“feed”) and mass lost from the primary, either as break leakage or as pressurizer safety or severe accident depressurization valve relief (“bleed”). Based on the analyses performed for Level 1 PRA, the “feed and bleed” strategy may fail, leading to a severe accident, when:

- Main feedwater system, startup and shutdown system and emergency feedwater system fail, loss of heat sink leads to overpressurization of the primary system, and relief out the pressurizer safety or severe accident depressurization valves.
- Partial cooldown fails (i.e., main steam relief train fails to open), thus, the primary system does not depressurize to actuate safety injection.
- Low head safety injection and/or medium head safety injection are unavailable.
- Accumulators are unavailable.

With the exception of the last point, these failures are preconditions in the suite of selected relevant scenarios. In addition, other conservatisms, such as degraded instrument setpoints, component performance, and response times applied in design basis analysis were incorporated in the characterization of the relevant scenarios. Beyond those conditions, it is further assumed to assure core failure that all cases occur at a time when preventive maintenance activities disable ECCS and EFW from one loop. Therefore, all cases have a maximum, if available, of three of four operating SI and EFW pumps. Accumulators are further penalized such that only two of four accumulators are considered available. Using two accumulators has shown to further penalize hydrogen generation (relative to having either one or three accumulators available) in scoping studies examining the LBOP/LOMFW event. This is done to further establish compliance with 10 CFR 50.44 requirements. Other details characterizing the relevant scenarios are provided in Table 19-99-1.

Table 19-99-1—Characterization of Relevant Scenarios

Case	**loop ss**	**loop pl**	**loop tr**	**lbop tr**	**sloca sl**
SCRAM	Yes @t=0	Yes @t=0	Yes @t=0	No	No
Pzr Htr &Spray	Both off	Both off	Both off	Both off	Both off
RCPs	Trip @t=0	Trip @t=0	Trip @t=0	Always run	Trip @t=0
MSIVs	Normal	Normal	Normal	Normal	Normal
PCD	No	Yes	No	No	Yes
FCD	Yes 40 min after PCD signal	No	No	No	No
MSRV	Normal	Normal	Normal	Normal	Normal
MSSV	Normal	Normal	Normal	Normal	Normal
MFW	Trip @t=0	Trip @t=0	Trip @t=0	Trip @t=0	Trip @t=0
EFW	On (3 of 4)	Off	Off	Off	On (3 of 4)
MHSI	Off	Off	Off	Off	Off
ACC	On (2 of 4)	On (2 of 4)	On (2 of 4)	On (2 of 4)	On (2 of 4)
LHSI	Off	Off	Off	Off	Off
PSV	Normal	Stick open	Normal	Normal	Normal
DRVopens at:	650°C	650°C	650°C + 2hrs	650°C + 2hrs	650°C
SAHRS	On	On	On	On	On
Primary Loop Break	Yes 0.75”φ	No	No	No	Yes 3”φ

In addition, for these best-estimate scenarios, the following event conditions are assumed:

- Reliable scram.
- Efficient “partial cooldown” when actuated (no stuck PORVs).
- No operator initiated late reflood.
- PDS actuates when core exit temperature is greater than 650°C.
- Full melt gate failure area.
- SAHRS active containment spray occurs 12 hours after PDS valves open.

- All mixing dampers and convection/rupture foils open when containment pressure/temperature conditions reach their associated setpoints.
- Hydrogen recombiners start automatically when hydrogen appears in containment.

Considering only these relevant scenarios, their corresponding fraction of the total CDF is presented in Table 19-99-2. This event frequency data provides the probability of a particular event type given a random relevant severe accident.

Table 19-99-2—Relevant Core Damage Events and Frequency

Description	Freq.
LOOP with Seal LOCA (LOOP SS)	50%
LOOP with a low pressure end state (LOOP PL)	10%
LOOP with a high pressure end state (LOOP TR)	13%
LBOP/LOMFW (LBOP TR)	20%
All LOCAs	7%

Several parameters are considered in the uncertainty analysis. Table 19-99-3 presents a summary of sensitivity parameters and presents a quantification of associated uncertainty ranges. This list was originally presented in the form of an RAI response for Reference 1.

Table 19-99 3—Summary of Parameter Ranges for Uncertainty Analysis

Table 19-99 3—Summary of Parameter Ranges for Uncertainty Analysis

The following steps were involved in performing the uncertainty analysis:

1. Identification of input parameters whose uncertainty has the most influence on the results.
2. Quantification of parameter uncertainty ranges and distributions developed based on the results of step #1.
3. Determination of desired statistical quantification statement for principal output measure. A 95% coverage/95% confidence was assumed to be an estimate of all output measure tolerance limits.
4. Generation of a sufficient number of MAAP4.0.7 input files that differ only in regard to the uncertainty parameters that are each randomly sampled over their specified uncertainty range. Fifty-nine MAAP4.0.7 input files were created to correspond to the desired statistical statement (i.e., 95/95). Each input file was distinctive by a unique sampling of the sensitivity parameters and by the sampling of the initiating event.
5. Execution of the 59 MAAP4.0.7 variation calculations.
6. Distill the results based on the values generated for the output metric(s) of interest.
7. Perform supplemental analyses, as needed.
8. Quantify importance of individual uncertainty parameters through the evaluation of correlations between the uncertainty parameter sampling set and the output metric set.

References for 19-99:

1. AREVA NP Document ANP-10268(P)(A), Revision 0, "U.S. EPR Severe Accident Evaluation," cited NRC document ADAMS Accession No. ML080650390, February 2008.

FSAR Impact:

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

Question 19-100:

Deleted.

Question 19-101:

Please provide a detailed description of the severe accident heat removal system (SAHRS), and a detailed assessment of its performance during representative and bounding severe accident scenarios.

Response to Question 19-101:

A response to this question will be provided by August 8, 2008.

Question 19-102:

Deleted.

Question 19-103:

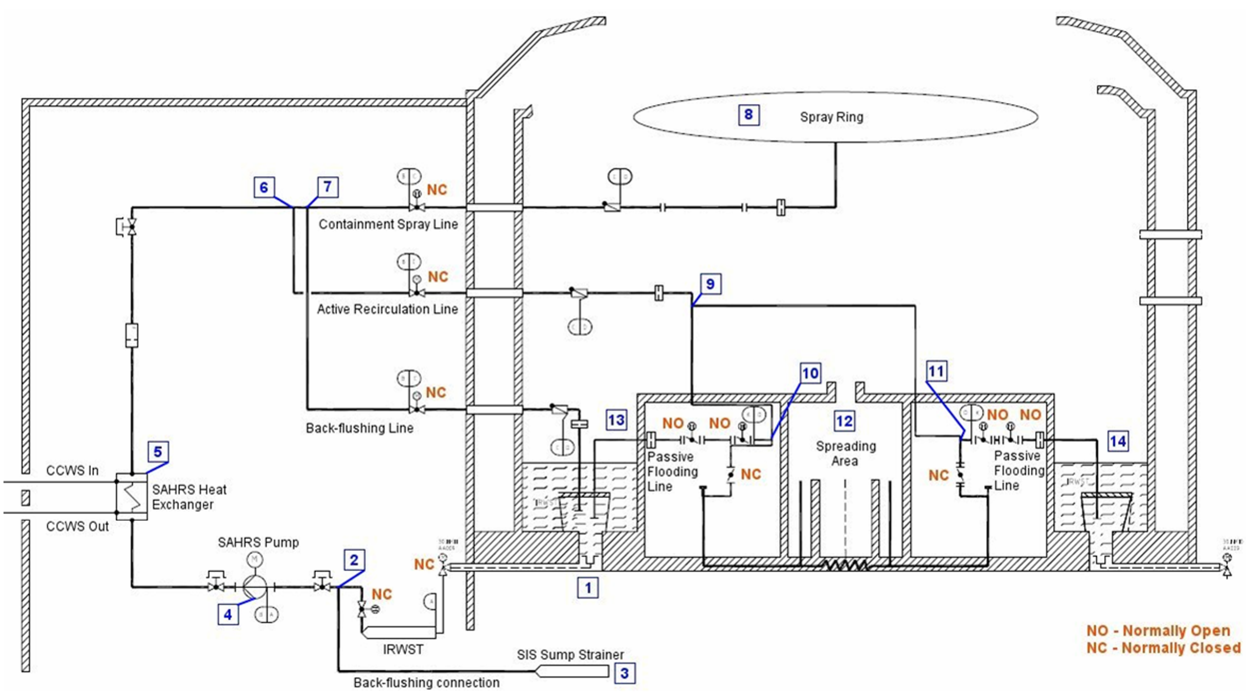
Provide a description and a simplified flow diagram (including valves and orifice plate) for the various in-containment flow paths of the SAHRS in its four modes of operation. Show the location of the piping connections to the flow paths for the active recirculation of water through the core melt spreading area and cooling structure. In addition, please provide the length associated with each pipe section and the design loss coefficients for all active and/or passive valves for various flow SAHRS paths. Furthermore, please provide the characteristic curve (head vs. flow) for the recirculation pump, the design head and the design pump flow rate.

Response to Question 19-103:

To transport the residual heat out of the containment in the long-term, the U.S. EPR is equipped with a dedicated severe accident heat removal system (SAHRS). As schematically depicted in Figure 19-103-1, it draws water from the in-containment refueling water storage tank (IRWST), feeds it through an external heat exchanger and re-injects it into the containment, thus avoiding outside dispersal of fission products.

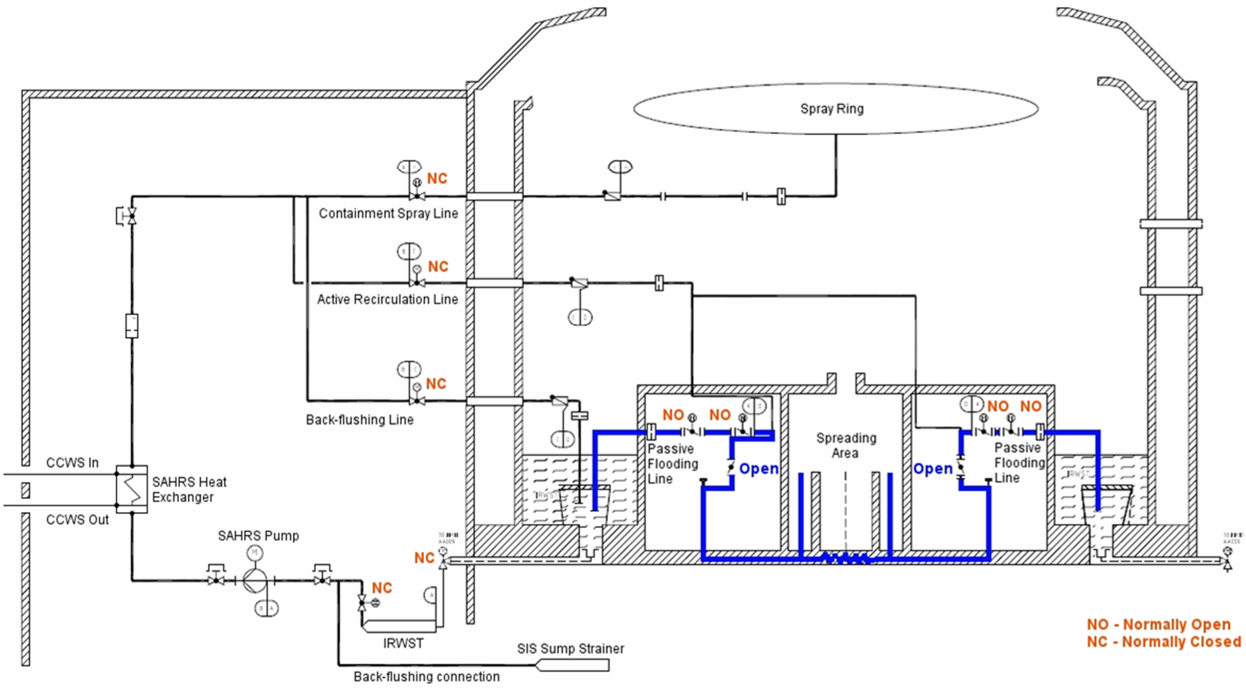
Figure 19-103-1 presents a simplified flow diagram for the SAHRS in stand-by mode. The containment isolation valves are normally closed (NC) as well as the passive flooding valves. The only valves normally open (NO) are the isolation valves on the passive flooding lines connecting the IRWST to the spreading area. The containment isolation valves and the isolation valves in the passive flooding lines are motor operated, while the passive flooding valves are passively operated and do not require electrical power to actuate.

Figure 19-103-1—SAHRS in Stand-By Mode



After initiation and progression of a severe accident, the molten corium relocates from the RPV to the spreading area. The arriving corium destroys the devices keeping the passive flooding valves closed and initiates the gravity driven flow of water from the IRWST to the spreading area via the cooling channels below and on the sides of the cooling structure as shown in Figure 19-103-2. No other valves will have to actuate during this phase of the corium stabilization and is therefore considered to be passive. The resulting steaming from the influx of water on top of the corium results in an increase in containment pressure and temperature.

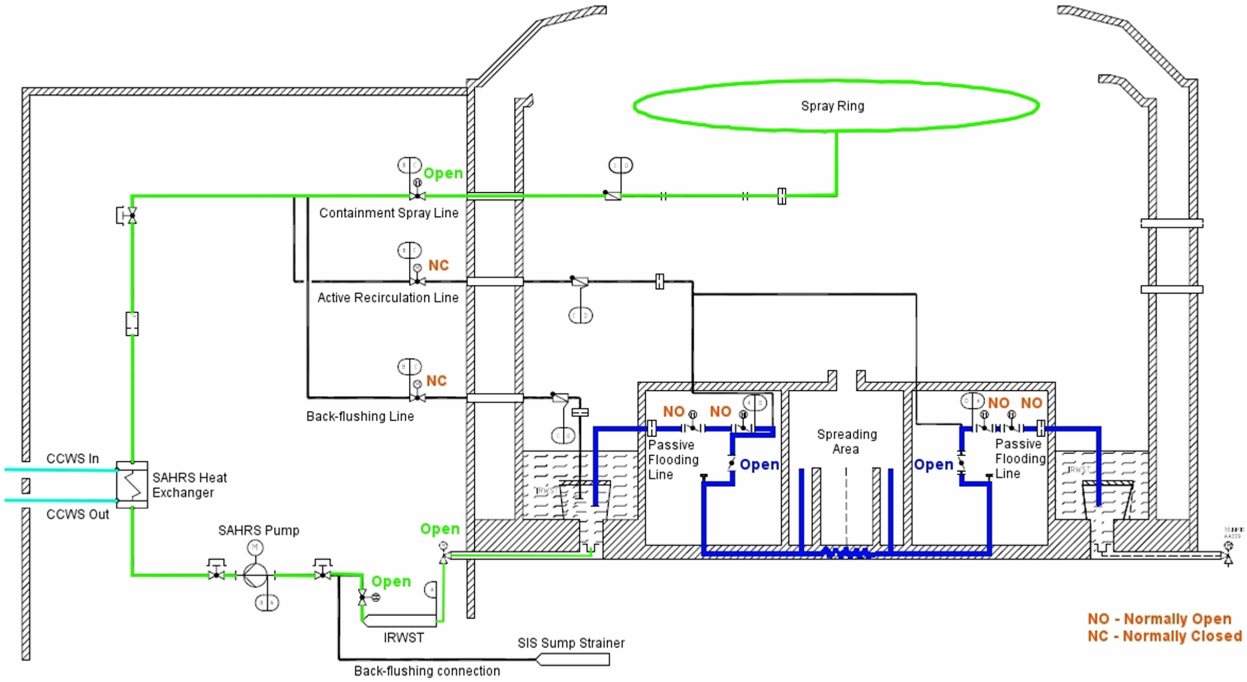
Figure 19-103-2—SAHRS in Passive Flooding Mode



In response to the increase in containment pressure, the SAHRS can be used to spray water drawn from the IRWST into the containment atmosphere. The condensate drains back into the IRWST from which the SAHRS pump takes suction. The water is pumped and cooled in the SAHRS heat exchanger before being re-injected into the containment. The main objective of the SAHRS in this mode is to reduce the containment pressure and to wash-out air-borne fission products. The operator manually starts the SAHRS based on a pressure criterion.

Figure 19-103-3 provides a simplified flow diagram for the SAHRS in spray mode. The containment isolation valves isolating the IRWST have to be opened, along with the containment isolation valve in the containment spray line. The SAHRS pump will not start unless valves downstream and upstream of the pump are open. No other valves have to be opened or closed to operate the SAHRS in this mode.

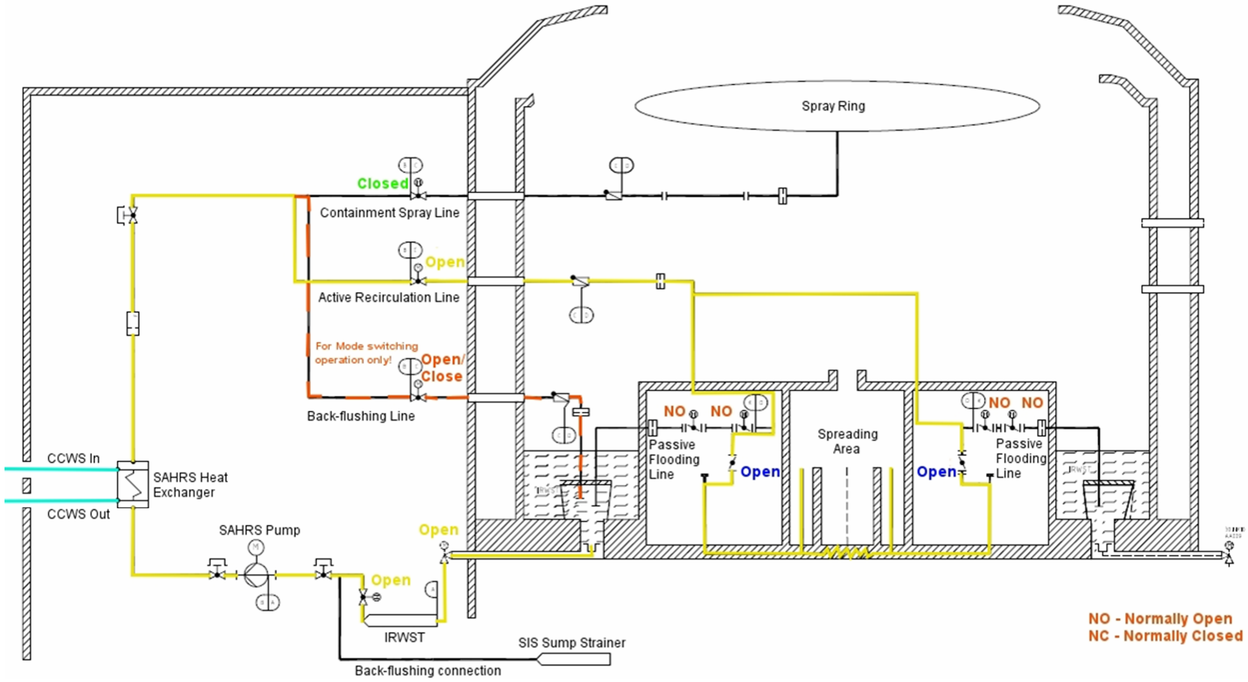
Figure 19-103-3—SAHRS in Spray Mode



In the long-term, the system can be used to feed re-circulated water directly into the spreading area. As a result, the water in the cooling channels and atop the melt becomes sub-cooled. Decay heat will now no longer be removed from the spread melt by evaporation into the containment atmosphere, but instead by single phase flow. In this active mode of corium cooling, the water level in the spreading room rises to the top of the steam chimney outlet. From there, the water overflows onto the heavy floor and drains back into the IRWST, thus closing the loop for the SAHRS. As the spreading room and the reactor cavity are connected via the transfer channel and the now open gate, water will also enter the reactor pit and submerge the RPV up to the level of the loop-lines. This establishes long-term cooling also for any debris that potentially remained in the cavity or the RPV itself. Based on this mode of operation, an ambient pressure level can be achieved in the containment long-term, which terminates further release of fission products through potential leaks.

To put the SAHRS into the active recirculation mode shown in Figure 19-103-4, the containment isolation valve in the back-flushing line is temporarily opened to provide a minimum flow line for the SAHRS pump. Next, the containment isolation valve for the spray line can be closed and the isolation valve for the active recirculation line is opened. When the flow to the cooling structure has been established, the back-flushing isolation valve is closed.

Figure 19-103-4—SAHRS in Active Recirculation Mode



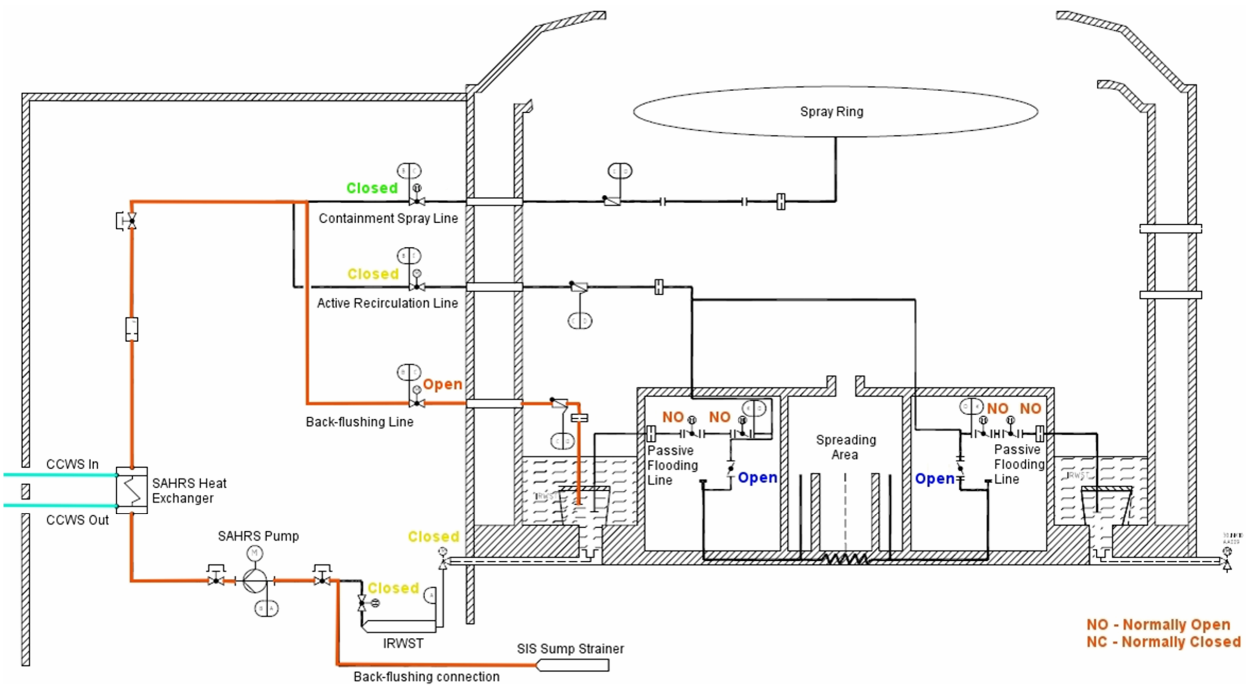
The back-flushing line can be used to flush the sump screens when too much debris has accumulated on the screens. Due to the additional pressure losses, the clogging effect decreases the suction head and thus the available NPSH for the SAHRS pump. When too much debris has accumulated on the screens, the operator switches the SAHRS train in the back-flushing configuration to protect the pump from cavitation, based on a differential pressure limit across the SAHRS strainer.

The back flushing is achieved by injecting water from the SAHRS pump into the clogged strainer through the dedicated back-flushing line. For effective back flushing, it is necessary to avoid pumping the water from the strainer that is being back flushed. Therefore, a connection to the safety injection system (SIS) allows the SAHRS pump to take suction from the neighboring SIS strainer. The back flushing is expected to last a few minutes.

The following steps are taken to achieve proper back-flushing configuration and are presented in the simple flow diagram in Figure 19-103-5:

- IRWST isolation valves are closed.
- SIS valves are open (not shown on the simplified flow diagram).
- Containment isolation valves on the spray and active recirculation lines are closed.
- The containment isolation valve on the back-flushing line is open.
- The pump is still in operation and takes suction in the SIS strainer.

Figure 19-103-5—SAHRS in Back Flushing Mode



The SAHRS piping and associated valves are represented in design calculations as shown in Tables 19-103-1 through 19-103-15. The numbers identifying the start point and end point of each section of pipe correspond to those in Figure 19-103-1. Some of the modeling has involved an embedded valve, or valves that are always open during SAHRS operation and only contribute a form loss. These valves are assumed to be globe valves. Their quantity and associated loss factors are shown next to the pipe in which they are embedded along with the internal diameter of the pipe. Other valves, such as the containment isolation valves, switch between the open and closed position. These valves are explicitly modeled. These valves are listed in the “Additional Valves” section, along with their form loss (K) and the internal diameter of the upstream pipe.

Figure 19-103-6 provides head, power and efficiency curves for the SAHRS pump.

Table 19-103-1

Table 19-103-2

Table 19-103-3

Table 19-103-4

Table 19-103-5

Table 19-103-6

Table 19-103-7

Table 19-103-8

Table 19-103-9

Table 19-103-10

Table 19-103-11

Table 19-103-12

Table 19-103-13

Table 19-103-14

Table 19-103-15

Figure 19-103-6—Characteristic Curve of the SAHRS (Recirculation) Pump

FSAR Impact:

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

Question 19-104:

Please state the time after initiation of a severe accident within which the SAHRS would have to become available, and relate the consequences were the SAHRS not to be available at this time. Also, please indicate whether there are other possibilities that might be available through SAMG operations to alleviate containment overpressure buildup.

Response to Question 19-104:

Review of the 59 cases from the uncertainty analysis shows that only 2 have containment pressures that exceed the design pressure of 5 bar (72 psia), and then only for a few minutes until SAHRS sprays initiate and bring down the pressure. Nevertheless, for the earlier case, this occurs at 13.24 hours from the start of the scenario. This is the latest time the SAHRS would need to be available to reduce containment pressure.

The Response to Question 19-109 illustrates the consequences of containment pressure if the SAHRS is unavailable. The upper curve in Figure 19.109-1 is a simulation of successful passive flooding, but a failure of the active portion of the SAHRS (no sprays or active cooling). The case on the bottom simulates a failure of both the active portion of the SAHRS and passive flooding. The containment pressure response is more severe when passive flooding is successful and the active portion has failed. The passive flooding water covers the melt and steams into the containment. While it carries heat away from the melt, pressure builds up in the containment.

The containment vent could possibly be used to alleviate containment overpressure conditions. The containment vent in the U.S. EPR is unfiltered, leaving the potential for a radioactive release. This release, however, would be controlled, as the containment would be closed again trapping any additional radioactive material. Venting with a controlled release is preferable during catastrophic containment failure, which would provide an uncontrolled release to the environment.

FSAR Impact:

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

Question 19-105:

Please provide the technical basis for debris/melt cooling on the spreading floor (a) in the early time due to gravity driven flow, and (b) over the long-term by pump recirculation. Please provide any experimental data that supports the conditions applicable to the U. S. EPR.

Response to Question 19-105:

A response to this question will be provided by August 8, 2008.

Question 19-106:

Please provide a detailed description of the core melt stabilization system (CMSS), and a detailed assessment of its performance during various stages of core melt progression, from vessel failure to long-term melt cooling and retention.

Response to Question 19-106:

A response to this question will be provided by August 8, 2008.

Question 19-107:

Please outline the assumptions regarding probabilities of, and containment loads due to, corium quenching following flooding in the spreading area for each distinct containment event tree branch as used for the Level 2 PRA. Include the assumptions and basis regarding base pressure at the time of this phenomenon.

Response to Question 19-107:

The analysis of containment failure due to containment loads during corium quenching involves the convolution of the loads expected during containment quench with the composite containment fragility function. The results of this convolution are distinct conditional containment failure probabilities that are specific to the Core Damage End State of the sequence being evaluated by the Containment Event Tree.

The following is an outline of the process by which this analysis is performed.

The peak containment pressure resulting from corium quench is determined by the formula:

Pcpeak = Pco + fq * P

With

- The base (initial) containment pressure at the time of debris flooding, Pco.
- The fraction of the core debris which is quenched, fq.
- The pressure increase in containment per fraction of debris quenched, P.

For the analysis of containment overpressure failure due to debris quench, the base pressure, fraction of debris quenched, and the pressure increase due to fraction of debris quenched are considered as key uncertainty parameters for the containment overpressure analysis.

The convolution of the containment loads with the containment fragility require the quantification of the peak containment pressure using distributions for the inputs.

For the initial containment pressure, Pco, the following values are chosen, with a uniform distribution taken between the two endpoints, for the base pressure in the Core Damage End States listed:

Table 19-107-1—Core Damage End State Containment Base Pressure

Core Damage End State	Base Pressure	Upper and Lower Bounds
TP/TR	45 psia / 30.5 psig	±7.3 psi
PL	33.4 psia /18.9 psig	±7.3 psi
SL / ML / SS / LL	27.6 psia / 13.1 psig	±7.3 psi

The base pressures are chosen based on the results of MAAP analyses performed specifically to examine the containment pressurization. These MAAP analyses were performed with conditions chosen to reflect the phenomenology of the Core Damage End State being examined.

For the fraction of core debris quenched, the MAAP 4.07 model uses a distribution describing the fraction of the debris quenched, assuming that heat transfer is limited by heat conduction through a solid crust. This distribution has a median at 10% and lower and upper bounds at 0 and 80%, respectively. This treatment assumes that crack formation and water ingression during quench is impossible. While it may be likely that a stable crust will form, at least initially, it is not considered impossible that crust cracking could occur during quenching.

A modified distribution function for the fraction of core debris quenched, fq, has been developed for the determination of peak containment pressure using the following hypotheses:

- A likely situation is that a stable crust will form and heat transfer will be conduction limited. In the distribution, a probability of 0.45 is assigned for quenching between 8 and 12%of the debris.
- Another likely configuration would be debris cracking and water ingress during debris quench, resulting in a critical heat flux limited heat transfer rate, which could allow quenching of close to 100% of the debris. In the distribution, a probability of 0.45 is assigned for quenching between 96 and 100% of the debris.
- All other physical situations of crust and water interaction are assumed to be equally likely. A uniform distribution, total probability of 0.1, is assigned to these.

For the probabilistic analysis of pressure increase during quench and in order to avoid potential non-conservatisms, the distribution for containment pressure rise per fraction of debris quenched is developed based on the MAAP results with fixed values of FCHF (the flat plate critical heat flux (CHF) Kutateladze number) for the LLOCA sequence. The basis for this distribution is:

- Most likely value (from FCHF=0.1 case): 53.7 psi pressure increase.
- Upper bound (from FCHF=1.0 case): 62.4 psi pressure increase.
- Distribution type: symmetric triangular. The triangular is chosen because FCHF=1.0 is seen as very extreme and this implies that care has been taken to choose a distribution that gives greater weight to the median value (i.e., some concentration of probability as the tail values are close to incredible).
- The same distribution is used for all CDES since this value is not expected to be dependent on the initiator.

For the calculation of conditional failure probability, the range peak containment pressure, Pcpeak, is evaluated by a Monte Carlo analysis in a spreadsheet with the Crystal Ball® software using the values and distributions listed above. The result of this calculation is a distribution of peak containment pressures, with a distribution dependent on the values of base pressure, fraction of core debris quenched, and pressure rise per fraction of debris quenched.

Pcpeak is compared directly with the composite containment fragility within the same spreadsheet, and a distribution of the results of the comparison – either containment fails, or containment remains intact, is evaluated.

The results of the Monte Carlo simulation using 1 million samples show a conditional probability of containment failure of 0.0 for CDES PL, SL, ML, SS, LL, and 3E-06 for CDES TP/TR.

FSAR Impact:

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

Question 19-108:

Deleted.

Question 19-109:

If available, please provide figures from MAAP calculations showing containment pressurization up to at least 80 hours for (a) a representative sequence with successful quenching and cooling of core debris ex-vessel resulting in continued long-term evaporation of water but without SAHRS sprays; (b) a representative sequence with MCCI but without SAHRS sprays; and (c) a representative sequence with MCCI and with operating SAHRS sprays.

Response to Question 19-109:

During the examination of long-term containment challenges, the response of the containment to a variety of scenarios was assessed.

The long-term containment pressure response was examined with the conditions discussed in (a), successful quenching and cooling of core debris ex-vessel resulting in continued long-term evaporation but without SAHRS sprays. The pressure response is shown in Figure 19-109-1 (the upper curve). Note that in this analysis, containment failure was postulated at 60 hours, since this MAAP sequence was also used to characterize the source term in the Level 2 analysis. The trend of containment pressurization is expected to continue until at least 80 hours and to exceed 12 bar (174 psi).

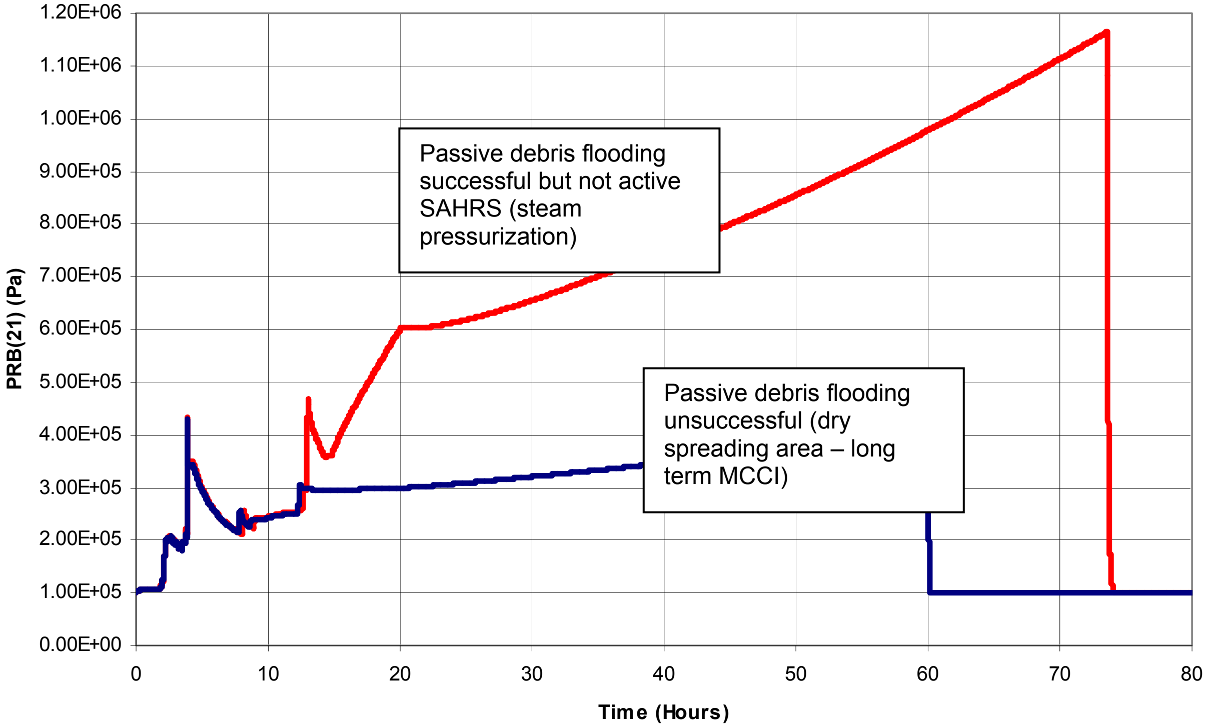
The corresponding containment pressure transient for (b) a representative sequence with MCCI but without SAHRS sprays is shown in Figure 19-109-1 (the lower curve), which indicates that by about 80 hours, the containment pressure reaches approximately 4 bar (58 psi). (Note that this sequence was also used to characterize source term, so there is a containment failure modeled at 60 hours).

A representative sequence with MCCI and with operating SAHRS sprays is not available. Without the significant steaming expected from passive flooding, containment pressure is expected to be very similar or bounded by the sequence with MCCI but without SAHRS sprays available (as a result of a preexisting high concentration of steam).

FSAR Impact:

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

Figure 19-109-1— Long Term Containment Pressure Transient for Cases with Flooding but No Heat Removal, and without Flooding



Question 19-110:

Please provide the basis for quantification in the Level 2 PRA of the probability that basemat melt-through will occur for branches in the CET involving protracted MCCI. Also, please provide the minimum thickness of the containment basemat assumed for U.S. EPR.

Response to Question 19-110:

The probability for basemat penetration is requested for sequences with significant MCCI where sprays are available, and where sprays are not available but overpressure failures do not occur. This split fraction is assigned a failure conditional probability of 0.99 (basemat melt-through occurs when significant MCCI is present), and a conditional success probability of 1E-02 (basemat melt-through does not occur when significant MCCI is present).

Theoretically, due to the large spreading area the possibility exists that even a dry core debris bed may cool sufficiently for MCCI to be arrested before the basemat is penetrated. Physically, this is possible if heat generated in the melt can be conducted away into the concrete with a ΔT below that required to sustain the concrete decomposition temperature.

Looking at the trends in Figure 19-110-1 for ablation rate in the spreading area [the pink line – XCNDB(2)], the rate of ablation in the spreading area is approx. 0.5 m in 30 hours (estimated from the value at 60 hrs and the estimated value at 90 hours), or 0.017 m/hr. The thickness of the basemat below the spreading area is 4.4 m. The time to penetrate the basemat is therefore, approximately:

$(4.4 – 1.5) / 0.017 + 60 = 230$ hr = 9.5 days.

It is assumed that overpressure failure does not occur for MCCI in a flooded spreading area. From the lower (blue) trend line in Figure 19-110-2, the containment pressurization rate during MCCI is approximately 14.5 psi (1 bar) in 40hr, or 0.363 psi/hr (0.025 bar/hr). At 60 hr, the pressure is approx. 58 psia (4 bar). Thus to reach the median failure pressure of 168.3 psig (11.6 bar), or 182.7 psia (12.6 bar), would take approximately:

$(12.6 – 4.0) / 0.025 + 60 = 404$ hours, or about 17 days.

These approximate calculations show that the first failure mode due to sustained MCCI is basemat penetration. Also, because of the debris temperatures during MCCI, and the ablation rate described above, there is no significant probability that MCCI will arrest before the basemat is penetrated. Therefore, the probability that the containment will fail due to basemat penetration is assigned a value of 0.99.

FSAR Impact:

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

Figure 19-110-1—Concrete Ablation in the Reactor Pit and Spreading Area – Failure of Passive Flooding

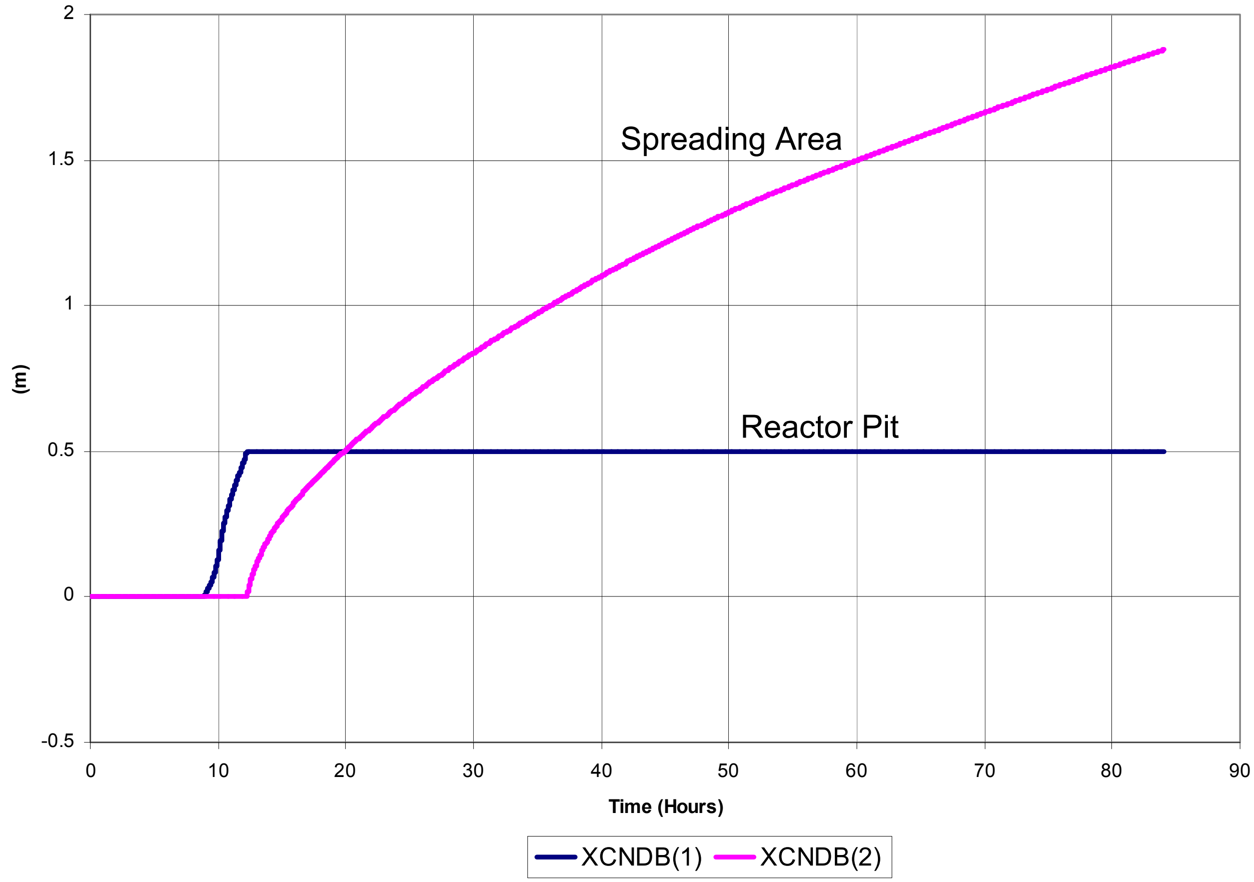
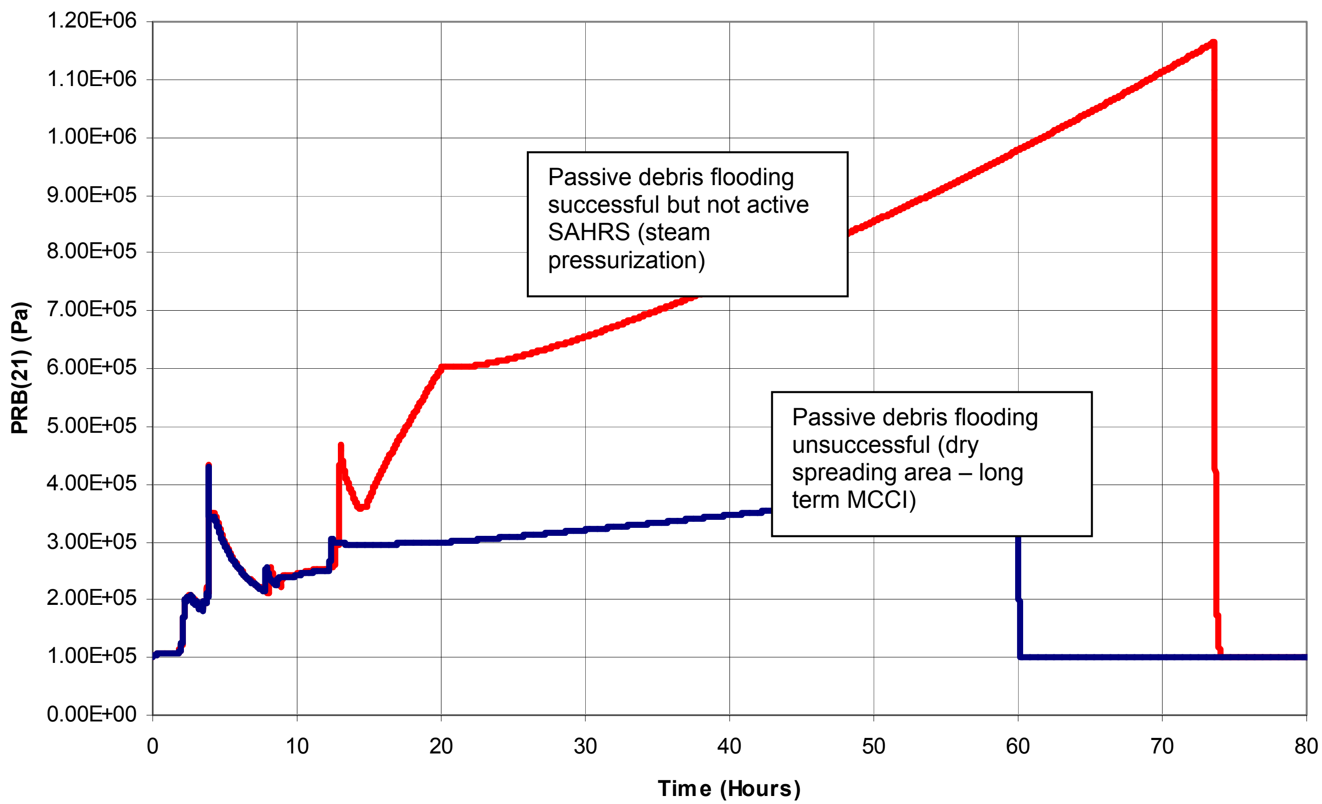


Figure 19-110-2—Long Term Containment Pressure Transient for Cases with Flooding but No Heat Removal, and without Flooding



Question 19-111:

The CMSS is located in the Seismic Category I Reactor Containment Building, but is not designed for seismic events. Please explain whether this system fails, during a seismic event that initiates a severe accident.

Response to Question 19-111:

The core melt stabilization system (CMSS) is a non-safety-related system and is not seismically qualified (NSC). Safety grade systems designed to withstand an earthquake event are charged with safely bringing the plant to a shutdown condition following a seismic event. Therefore, the CMSS was not designed for a severe accident that is initiated by a seismic event. The combination of the low probability of a severe accident coupled with the low probability of a seismic event, results in an acceptable low frequency scenario, which is screened from further consideration for the U.S. EPR. The only portion of the CMSS designated with a Seismic Category II classification is the melt discharge channel, which partially supports the reactor cavity structural concrete.

FSAR Impact:

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

Question 19-112:

The core melt plug must be removed from the lower cavity floor to allow for certain shutdown inspection activities. Please state whether it would be a condition for return to power that this plug be satisfactorily back in place.

Response to Question 19-112:

The functions of the melt plug are:

- To provide a removable separation between the reactor cavity and the spreading area of the core melt stabilization system (CMSS).
- To maintain temporary retention and accumulation of the corium in the reactor cavity.
- To fail with an adequate open cross section after melt contact.

The melt plug can be removed through the melt discharge channel to give access to the lower part of the reactor cavity to enable outside ultrasonic examination of the reactor pressure vessel (RPV) during shutdown inspection activities.

The separation of reactor cavity and spreading area allows the cooling structure in the spreading compartment to be safe from potentially critical loads related to the failure of the reactor pressure vessel (RPV) in case of a severe accident. Likewise, the separation of reactor cavity and spreading area protects the reactor cavity from an unintentional flooding of the spreading area during power operation. Based on this, the melt plug must be returned to its position in the bottom of the reactor cavity and locked in place prior to the plant returning to full power.

FSAR Impact:

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

Question 19-113:

Please provide a detailed severe accident equipment survivability evaluation, consistent with directions in SECY-90-016 and SECY-93-087.

Response to Question 19-113:

In accordance with SECY-90-016 and SECY-93-087, features provided solely for severe accident protection need not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirements of 10 CFR Part 50, Appendix B, or redundancy/diversity requirements of 10 CFR Part 50, Appendix A. However, the guidelines go on to state that reasonable assurance must be provided proving that mitigation features will operate in a severe accident environment (e.g., pressure, temperature, radiation) for which they are intended, and over the time span for which they are needed.

As outlined in FSAR Tier 2 Section 19.2.4.4.5, the U.S. EPR is designed to cope with a severe accident in such a way to limit the radiological consequences to the plant. This can be established and monitored with appropriate instrumentation and components to:

- Perform operator actions.
- Survey the effectiveness of the installed mitigation measures.
- Survey the overall plant conditions including possible releases to the environment during a severe accident.

Design features are provided to mitigate the consequences of a severe accident. A minimal number of components are relied upon to function during the sequence of events following the onset of severe accident conditions. While safety-related systems used to address design basis accidents (DBA) may become available at a later time, they are not required to operate or prevent containment or basemat failure. Should such systems become available, the operator may choose to employ them to address conditions of the accident. However, their operation is not credited and they are assumed to be inoperable and the main cause for leading to a severe accident. The Severe Accident Management Guidelines (SAMG) will address the use of such systems should they become available and how they may be used to enhance the mitigation of the severe accident.

Systems specifically designed to address the environmental conditions during a severe accident within the RCS and the containment are:

- Primary depressurization system valves (PDS).
- Core melt stabilization system (CMSS).
- Combustible gas control system (CGCS).
- Severe accident heat removal system (SAHRS).

The PDS, CMSS, and CGCS components are located inside the containment and must therefore be qualified for local ambient conditions of pressure, temperature, and humidity. While the SAHRS is used to limit the pressure and temperature inside the containment, its main components, the heat exchanger and pump, are not located inside the containment. These components only need to be qualified for elevated temperature and radiation doses inside the compartments in the Safeguard Building where they are located.

Containment isolation valves, containment penetrations, air locks, hatches and gaskets are required to maintain their leak-tightness during a severe accident. Therefore, this equipment must be qualified to withstand the severe environmental conditions during a severe accident.

The U.S. EPR severe accident instrumentation concept is mainly passive. Therefore, a few instruments are needed to monitor the mitigation path and to establish limited radiological consequences. The severe accident mitigation path assumes that systems lost leading to a severe accident are not recovered before RPV failure. Instrumentation used to survey the effectiveness of the installed mitigation measures and to survey overall plant conditions is not necessary to achieve the control of radiological releases. Most of the measurements necessary or useful for severe accident management and monitoring can be performed with instrumentation already provided by operational and safety I&C. However, the measurement range of this instrumentation may need to be adjusted to severe accident environmental conditions.

Whereas all containment systems need to maintain their leak-tightness and integrity, only those containment penetrations associated with the operation of the SAHRS and with the severe accident instrumentation need to retain their full functionality during a severe accident. The instrumentation and control functions necessary during a severe accident are implemented in a dedicated severe accident I&C system.

The environmental conditions that severe accident-related equipment is exposed to during a severe accident, can be differentiated into four location categories:

- In-Core/ In-RCS.
- Inside containment.
- SAHRS-compartments (in Safeguard Building 4).
- Annulus.

Any component necessary during a severe accident located inside the core or RCS should remain operable until RPV failure. This time span, referred to as the in-vessel phase, can be divided into the following phases:

- Start of accident and departure of RCS pressure and temperature from normal operating conditions.
- Loss of accident control functions leads to a core outlet temperature increase up to 1202°F.
- Actuation of the PDS valves and start of depressurization at the 1202°F signal.
- End of depressurization.
- Further increase of core outlet temperature from 1202°F up to start of core oxidation at approximately 2012°F (1100°C).
- Accelerated heat up of core temperature beyond 4532°F (2500°C) and core melt onset.
- Relocation of the core melt to the lower RPV head.

- Increase of RPV outside wall temperature beyond 1472°F (800°C) and subsequent RPV failure.

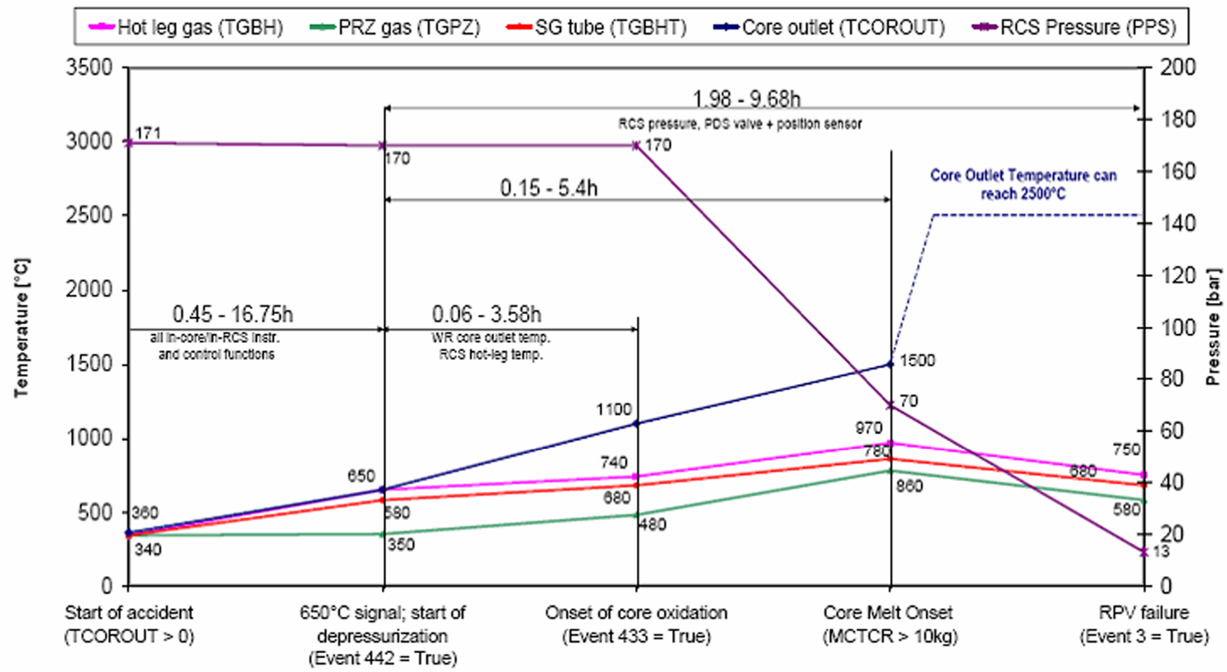
The instrumentation required and relied upon for these phases of the accident progression are:

- Wide-range core outlet temperature thermocouples.
- RCS pressure sensors.
- PDS valves.

The different requirements for operability duration (mission time) are summarized in Figure 19-113-1. The time between each of the events depends on the scenario. The values are the predicted minimum and maximum time spans, based on the results for the relevant scenarios modeled in MAAP for the uncertainty analysis. The temperatures represent the maximum temperatures among the different scenarios at the corresponding event time. Figure 19-113-1) is a synthesis giving the bounding temperature values of the many possible scenarios. The time development of the temperature between events is not necessarily linear.

The variables to plot were chosen to represent the environmental conditions of the releant instrumentation and equipment. Therefore, the RCS pressure (PPS), the hot leg gas (TGBH) and core outlet (TCOROUT) temperature were modeled to provide bounding conditions for qualificationof the core outlet thermocouples and RCS pressure sensors. The gas temperature in the pressurizer (TGPZ) was modeled to provide input for the qualification of the PDS valves and position sensors. While the temperature in steam generator tubes (TGBHT) are not currently used for qualification purposes, they are provided for future reference should instrumentation be deemed necessary, and to present a complete picture of the temperature development of the RCS during a severe accident.

Figure 19-113-1—Bounding In-Core Temperature and Pressure Development during Severe Accident



The displayed measurement range of the wide-range core outlet temperature thermocouples should be at least 2282°F (1250°C) with a qualification up to 1832°F (1000°C). With respect to pressure, the thermocouples should be qualified to 2901 psi (200 bar). As the severe accident progresses after depressurization of the RCS, the temperature at the location of the core outlet thermocouples will increase beyond their qualification range, thus leading to their anticipated failure before RPV failure.

To have continuous information on the efficiency of depressurization, which is to decrease RCS pressure to below 290 psi (20 bar) at RPV failure, the pressure sensors should remain available until failure occurs. The sensors should deliver correct values even if the hot leg (TGBH) and PRZ gas (TGPZ) temperatures exceed 2192°F (1200°C).

The qualification requirement for the temperature on the inside of the PDS valves, located at the top of the pressurizer, is 1112°F (600°C) to establish reliable opening of the valves, which occurs only once at the latest at the 1202°F (650°C) core outlet temperature signal. The PDS discharge capability is required until RPV failure. The PDS valves should remain open and their position reliably indicated up to PRZ gas temperatures of 1832°F (1000°C). On the outside, the PDS valves and their position sensors will be qualified to withstand the environmental conditions inside the containment during a severe accident as discussed below. No additional severe accident related requirements relative to cumulative radiation dose are set for the in-core instrumentation.

Inside the containment, the equipment and instrumentation associated with severe accidents are:

- Containment isolation valves and position sensors.
- PDS valves and position sensors.
- Containment pressure sensors.
- H2 monitors.
- Hydrogen mixing dampers and position sensors, PARs, Convection and Rupture foils.
- IRWST water level and temperature.
- Dose rate measurement (i.e., gamma-sensitive detector).
- Severe accident sampling system.
- Thermocouples inside insulation liner to measure temperature of RPV lower head.
- Flooding valves and position sensors.
- Thermocouples in spreading area main cooling channel and steam chimney.
- Containment spray nozzles.

Environmental conditions inside the containment that may be more severe during a severe accident than during a DBA are:

- Static pressure.
- Potential short term pressure spikes due to H2 combustion.
- Temperature.
- Presence of gases such as steam (i.e., humidity, hydrogen, carbon monoxide)
- Radiation and deposition of radioactive aerosols.

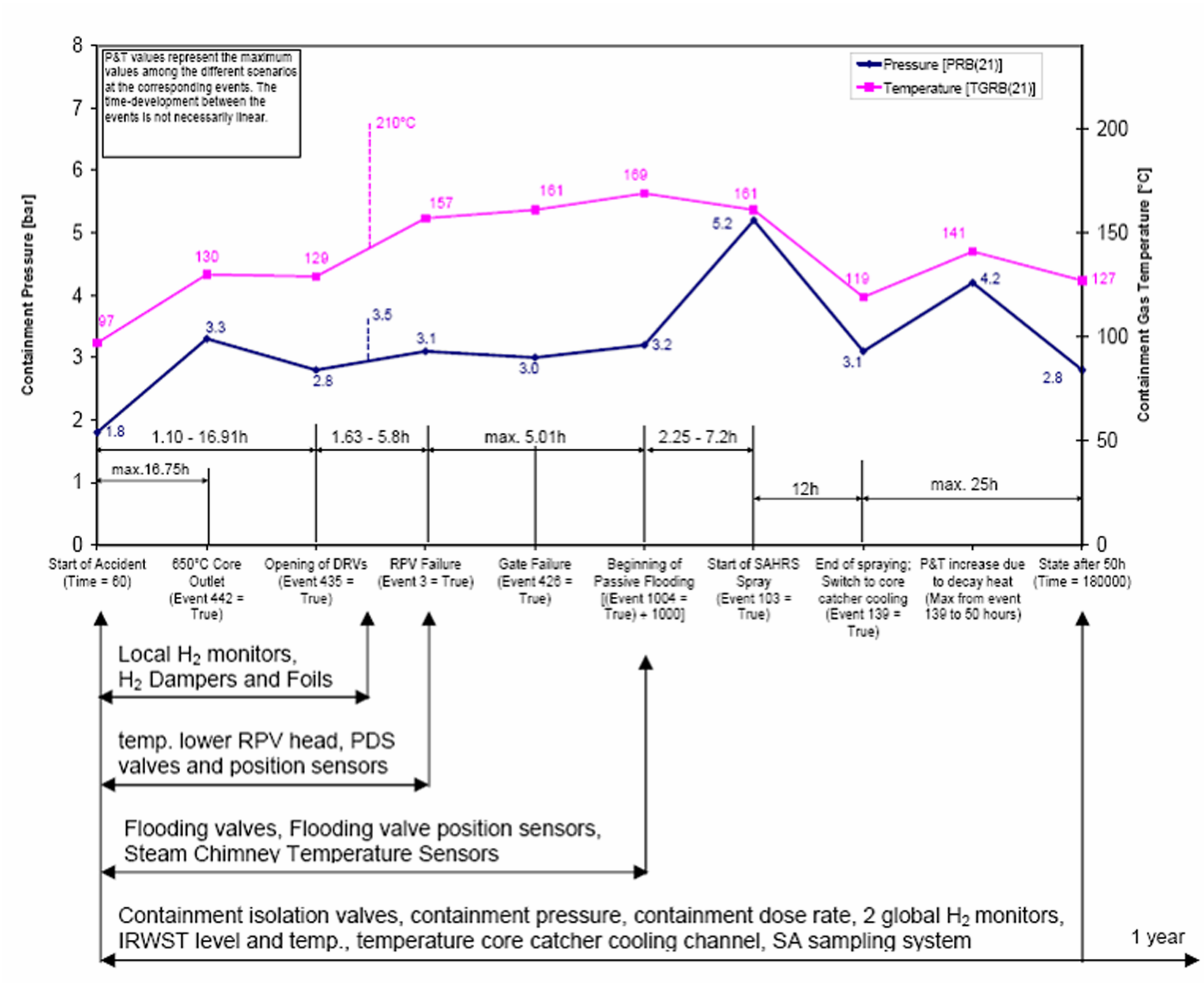
The time span for which the equipment is to remain operable (mission time) is important for the design of equipment. This is described in Figure 19-113-2, which gives a schematic representation of the course of main events and maximum pressure and gas temperature in containment during a severe accident. The time between each of the events depends on the scenario. The values shown represent the maximum value among the different scenarios at the corresponding event time. Figure 19-113-2 is again a synthesis giving the bounding values of the many possible scenarios. The time development of pressure and temperature between events are not necessarily linear.

The course of events and the corresponding environmental conditions inside the containment during in- and ex-vessel phases of a severe accident are visualized in detail for the different scenarios, and can be divided into the following phases:

- Start of accident: pressure, temperature, and humidity increase inside the containment due to steam release from the RCS.
- Closing of containment isolation valves, actuation of the PDS valves, and start of depressurization at the 650°C signal.

- Hydrogen production and fission product release due to start of core oxidation and degradation.
- Escape of hydrogen into the containment via break and/or depressurization valves; increase of average hydrogen concentration to max. 10 vol%.
- Increase of radiation level due to fission product release.
- Reduction of hydrogen concentration by passive autocatalytic recombiners (PARs).
- Pressure and temperature increase due to H2 recombination.
- RPV failure.
- MCCI in reactor pit and continued increase of containment temperature.
- Gate failure and spreading of core melt in the core catcher.
- MCCI in the core catcher: Hydrogen production and increase of pressure and temperature inside the containment.
- Pressure increase due to quenching of melt by opening of flooding valves and inflow of water from IRWST.
- Continued pressure increase due to evaporation of water on melt because of decay heat.
- Start of SAHRS spray.
- Reduction of pressure and temperature due to heat removal.

Figure 19-113-2—Bounding Pressure & Temperature Development



The pressure, temperature, and humidity time developments during a severe accident are based on data obtained from the 59 cases run for the uncertainty analysis. Based on Figure 19-113-2, the maximum "global" containment pressure and temperature equipment and instrumentation may be exposed to during the progression of a severe accident are 75.4 psi (5.2 bar) and approximately 410°F (210°C), respectively. The maximum humidity inside the containment experienced by the equipment can conservatively be assumed to be 100% after the commencement of spray. However, due to the existence of other gases inside containment, the steam concentration approaches a conservative value of 80%. Finally the IRWST reaches a maximum temperature of 257°F (125°C). The SAHRS system itself is conservatively designed for a maximum IRWST water temperature of 320°F (160°C).

For the U.S. EPR, it has been shown that for the relevant scenarios, localized hydrogen detonation and deflagration can be reliably excluded. However, as the highest adiabatic, isochoric complete combustion (AICC) pressure and temperature resulting from the analysis of the data obtained from the 59 cases is 105.9 psi (7.3 bar) and 1634°F (890°C), respectively; an assessment of the extended operational range with respect to a pressure spike of 105.9 psi (7.3

bar) and temperature of close to 1652°F (900°C) will be made for relevant equipment and instrumentation that needs to remain operational after a potential hydrogen combustion. It should be kept in mind that AICC pressure is purely a theoretical value that in reality cannot be reached, as the combustion is neither adiabatic, nor isochoric, nor complete.

While all equipment and instrumentation inside containment may be exposed to such pressure and temperature spikes, only equipment relied upon to actively mitigate the consequences of hydrogen in the containment atmosphere is required to survive such occurrences per 10 CFR 50.44. Therefore, the hydrogen mixing dampers, PARs, and the rupture and convection foils in the steam generator ceiling should be shown to be capable of surviving such short lived pressure and temperature spikes.

During testing of the AREVA NP specific PARs under a collaborative research agreement between EPRI and EdF, PAR outlet temperatures of well above 932°F (500°C) were observed. The temperature increase during the test was monitored at the active plate and was attributed to a deflagration that occurred during the test. An inspection afterwards revealed no damage to the PAR or the plates. It is therefore consistent to assume that the operational capacity of the PARs is well within the range of AICC temperatures. The AICC scenario that was considered in the combustion evaluation also assumed that the hydrogen concentration is allowed to accumulate such that ignition occurs at the worst possible time (hydrogen is not allowed to burn until the concentration reaches a maximum). This is an unrealistic scenario given that the atmospheric conditions in the containment during a severe accident are quite favorable for hydrogen combustion. It is therefore likely that the median AICC temperatures will be well within the range of the PARs. Based on the design, the PARs are not pressure retaining components. The PARs are open at the bottom and the top, and therefore unaffected by localized pressure increase. Similar arguments can be made for the hydrogen mixing dampers and the rupture and convection foils. As those open on pressure differential and, in the case of the convection foils, on temperature differential their operation is not affected by localized pressure and temperature increase due to hydrogen combustion.

A quantitative analysis of the direct dose radiation environment in the U.S. EPR buildings, as well as the submersion dose for the Reactor Building accident conditions for use in the environmental qualification of equipment, was performed for the U.S. EPR. For DBA conditions, the analysis is based on a large break loss of coolant accident (LBLOCA) which provides the bounding dose rates within the Reactor Building and Safeguard Buildings. To establish radiation doses for equipment relied upon to monitor and mitigate the consequences of such a severe accident, scaled results from a similar plant were used. The results were scaled based on a comparative study of the core inventories for the similar plant and the U.S. EPR.

The radiation levels relevant for qualification inside the containment during a severe accident assume that the fraction of the core inventory released to the containment is distributed uniformly both in the atmosphere and on the walls. The corresponding cumulative dose values based on highly conservative assumptions (e.g., no wash out) were taken from OL3 analysis, adjusted by the weighted average factor derived from the comparative study, and are presented as follows:

- Dose due to air-borne gamma radiation after 24 hours: *399 kGy*
- Dose due to air-borne gamma radiation after 1 year: *6670 kGy*
- Dose due to deposition gamma radiation after 24 hours: *385 kGy*

- Dose due to deposition gamma radiation after 1 year: *6815 kGy*
- Dose due to gamma radiation after one year due to the activity inventory of the IRWST (calculation dose point is at the water level): *4640 kGy*

These dose values are maximum values based on conservative assumptions. The calculated dose for one year implies that for one year the conditions inside the containment are nearly constant. Only radioactive decay is considered as the effect leading to the decrease of the activity inventory. Other effects like transport processes of nuclides from the containment atmosphere into the IRWST water, where a part of radioactive nuclides would be retained, are not considered. The equipment inside the containment which is required to function after a severe accident (core melt) is expected to be located in such a way that direct exposure from IRWST water surface is excluded. That means the local dose rates will be lower due to shielding by existing walls and structures.

The SAHRS, with exception of the spray system, is not exposed to any severe accident related conditions until started. After start of the SAHRS, the contaminated IRWST water, which has a a maximum temperature of 257°F (125°C), flows through the system. The nominal boron concentration in the IRWST is 1700 ppm ± 100 ppm with a maximum value of 7000 ppm.

Two cases were analyzed for calculating the cumulative dose in the SAHRS compartments:

1. Gamma radiation dose due to a pipe of the SAHRS system in operation (water is assumed to be drawn from the IRWST).
2. Gamma radiation dose in a SAHRS compartment caused by spread fluid on the floor due to an assumed pipe leakage.

Adjusting those values based on the comparative study, the main results are:

- Dose due to gamma radiation from the pipe in 1 m distance after one year: *131 kGy*
- Dose due to gamma radiation in a SAHRS compartment caused by leaked fluid on the floor after 100 hours: *334 kGy*

The selection of a reference dose point 1 m away from the pipe takes into consideration the layout of the pipe branches and the SAHRS room location. The simultaneous radiation from two pipes, one on the pump suction side and one on the pump pressure side, is considered for the expected gamma dose. Inside the heat exchanger and valve room a similar situation exists with regard to the pipes running to and from the heat exchanger. Additionally, injection into the containment is possible via different paths depending on the SAHRS operating mode – spraying, active cooling of the spreading area, or back flushing.

Taking these items into account, it is assumed that irradiation can take place from two sides. Qualification in these rooms take this into account by doubling the dose from 131 kGy to 262 kGy. The dose caused by spilled fluid after a pipe break or leakage is not relevant for equipment qualification of SAHRS components because such an event would result in long-term inoperability of the affected SAHRS components.

The systems, instrumentation, and equipment identified in the severe accident equipment survivability evaluation are relied upon as a minimum to function during the environmental conditions specified. In the event of a severe accident, environmental conditions inside the containment, the RCS, the SAHRS compartments, and the annulus can be harsher than during a DBA. The environmental conditions during a severe accident presented in the severe accident equipment survivability evaluation serve as the basis for defining qualification profiles for instrumentation and equipment that are necessary for containment isolation and managing and monitoring a severe accident. The corresponding pressure and/or temperature values presented in Figure 19-113-1 and Figure 19-113-2 are bounding values, which take the different possible event scenarios into account.

FSAR Impact:

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

Question 19-114:

Please provide a detailed description of the combustible gas control system and a detailed analysis of its performance during representative and bounding severe accident scenarios. Include a detailed description of the U.S. EPR hydrogen mitigation strategy.

Response to Question 19-114:

A response to this question will be provided by August 8, 2008.

Question 19-115:

For purposes of the source term calculations, please provide and justify the maximum number of broken tubes that were assumed in the case of a temperature-induced SGTR, considering the fact that it is not clear that a single broken tube would necessarily depressurize the primary side of the system to an extent that would mitigate further tube failures.

Response to Question 19-115:

A response to this question will be provided by August 8, 2008.

Question 19-116:

Please provide the basis for quantification of and uncertainty with respect to the decontamination factor (DF) assumed in the steam generators for SGTR scenarios with scrubbing available in the Level 2 PRA. If the basis for this quantification involves assumptions regarding the aerosol size distribution, please also provide the assumed distribution and its basis.

Response to Question 19-116:

The quantification of the decontamination factor (DF) assumed in the steam generators for SGTR scenarios with scrubbing available was performed as follows:

- A global DF of 20 was initially assumed for all particles in SGTR sequences with feedwater available.
- A scrubbing evaluation was performed using the MAAP pool scrubbing model to confirm the adequacy of this assumption.

Important MAAP4.0.7 Models

Important MAAP4.0.7 models used in this response are shown in Table 19-116-1. Full descriptions of the modeling can be found in the descriptions of the associated subroutines in Volume 2 of Reference 1.

Pool scrubbing is modeled in MAAP4.0.7 via a series of lookup tables with data originating from analysis with the SUPRA® pool scrubbing code. DFs are interpolated for each aerosol particle size range and for various thermal hydraulic conditions, as shown in Table 19-116-2.

Estimation of SG DFs

Interpolations were done within a spreadsheet with data chosen to be typical of an SGTR severe accident, as shown in Table 19-116-3.

Figure 19-116-1 shows the results for “downcomer” injection mode with various pool depths. If the 3m deep pool with zero subcooling is used, a wide range of DF is apparent, with all particle sizes larger than 0.3 – 0.4 micron radius experiencing DF>20 (corresponding to Log(DF)>1.3).

To estimate the aerosol size distribution, Figure 19-116-4 was used (distribution with aerosol source taken since it has more small particles which have lower DFs). To read this plot, the non-dimensional particle size parameter VR has to be converted back to a radius in meters (refer to Table 19-116-4). This was done using the data shown in Table 19-116-3 with the result shown in Table 19-116-5. It can be seen that at about a 0.5 micron radius, the value of VR is 1.0. Comparing now with the size distribution shown in Figure 19-116-4, it is clear that a large fraction of the aerosol mass will be in the form of particles with DF>20. On this basis, for the conditions assumed, a value of 20 was chosen.

Sensitivity

Results show a strong sensitivity to the parameter “injection mode.” In the steam generator, MAAP4.0.7 uses a side vent injection mode instead of a downcomer injection. Interpolations were repeated using the side vent data results in Figure 19-116-2 and Figure 19-116-3. Particles above about 1 micron are likely to be scrubbed with DF>20 provided the pool is at least approximately 1.8 m deep. The sensitivity analysis confirms the adequacy of the value chosen for the DF.

Conclusion

For the characterization of the scrubbed SGTR source term (RC701), a DF of 20 has been applied to the unscrubbed SGTR source term (RC702) from MAAP4.0.7. This value is found to be conservative for most aerosols, and overall adequate to model source term scrubbing in the SGs.

Table 19-116-1—MAAP4.0.7 Models of Pool Scrubbing

Name	Type	Description
POOLD	**Blockdata**	**Contains all the look-up data based on SUPRA® analysis of pool scrubbing**
POOLDF	Subroutine	Calculates overall pool DF
AERODF	Subroutine	Calculates overall pool DF
AMDEF	Subroutine	Sets up mass / size distributions for aerosols

Table 19-116-2—Independent Parameters and their Respective Values for the DF Data Tables in MAAP4.0.7

Mode of Gas Injection[1]	Parameter	Range of Values Used to Create DF Tables
Sparger	Aerosol Particle Radius Steam Mass Fraction Pool Height Pressure Subcooling of Pool Gas Consumption	Ten Values[2] 0, 0.5, 0.9, 0.99 0.5, 1, 2, 4, 6 m 1, 2, 5 atm 0, 1, 1.15, 1.5, 10, 10.15, 10.5, 30, 30.15, 30.5 K Hydrogen
Downcomer	Aerosol Particle Radius Steam Mass Fraction Pool Height Pressure Subcooling of Pool Gas Consumption	Ten Values[2] 0, 0.5, 0.9, 0.99 0.5, 1, 3 m 1, 5 atm 0, 1, 1.15, 2.15, 2.5, 10.15, 10.5, 30.15, 30.5 K Air, Hydrogen
Side Vent	Aerosol Particle Radius Steam Mass Fraction Pool Height Pressure Subcooling of Pool Gas Consumption	Ten Values[2] 0, 0.5, 0.9, 0.99 0.5, 0.8, 1.8 m 1 atm 0, 1.15, 2.15, 10.15, 30.15 K Air, Hydrogen

Notes:

1. The nominal geometry used in the SUPRA® calculation for each type of injector is as follows. The DFs are a strong function of the mode of injection, but are not very dependent on the injector diameter.

Sparger	1500 orfices per sparger
	0.5 in. (1.27 cm) diameter orfices
Downcomer	2 foot (0.61 m) diameter vertical pipe
Side Vent	2.3 foot (0.70 m) diameter opening in wall of pool

2. The ten particle radii are 0.01, 0.035, 0.06, 0.08, 0.1, 0.2, 0.4, 0.6, 0.8, 1.0 microns.

Table 19-116-3—Values Used in the DF Interpolations

Parameter	Value Used	Units	Reference / description
gamma	2.5	-	Parameter file, variable GSHAPE
g	9.81	ms^{-2}	MAAP source, S/R AMDIST
k	1.38E-23	J/K·molecule	Boltzmann constant
Assumed gas conditions:			
T_{gas}	900	K	Temperature Avg of hottest tube gas temp and saturation
P	1.00E+05	Pa	Pressure - depressurized SGs
rho	1000	kg/m^3	Density of aerosol particles, S/R FPTRAN
mu	3.37E-05	Pa·s	Viscosity of gas

Table 19-116-4—Nondimensional Scalings for the Macroscopic Properties of a Sedimentation Aerosol (Reference 1)

Nondimensional Mass, M	Nondimensional Particle Volume, $\overline{V}$	Nondimensional Source, M_p
$\left(\frac{g\ h^4\ \gamma^9}{\rho^3\ \mu\ k_o\ \alpha^3}\right)^{1/4} m$	$\left(\frac{\gamma\ g\ \rho}{\alpha^{1/3}\ \mu\ k_o}\right)^{3/4} v$	$\left(\frac{\gamma^{11}\ \chi^4\ \mu\ h^8}{\alpha^5\ g\ k_o^3\ \rho^5}\right)^{1/4} \dot{m}_p$

with

$$k_o = \frac{4\ k\ T_{gas}}{3\ \mu}$$

where h - equivalent length for deposition
γ - agglomeration enhancement factor
α - particle porosity factor
χ - Stoke's law correction factor
v - particle volume
m - aerosol concentration (kg m/m³)
$\dot{m}_p$ - aerosol source concentration (kg m/m³ sec)
k - Boltzmann's constant
T_{gas} - gas temperature
μ - gas viscosity

Table 19-116-5—Relation Between Nondimensional Particle Size, VR and Particle Radius

VR	r (m)
1.00E-02	1.21E-07
1.00E-01	1.21E-06
1.00E+00	5.62E-07
1.00E+01	1.21E-06
1.00E+02	2.61E-06
1.00E+03	5.62E-06
1.00E+04	1.21E-05
1.00E+05	2.61E-05
1.00E+06	5.62E-05
1.00E+07	0.000121

Figure 19-116-1—Pool Scrubbing Data for Downcomer Venting

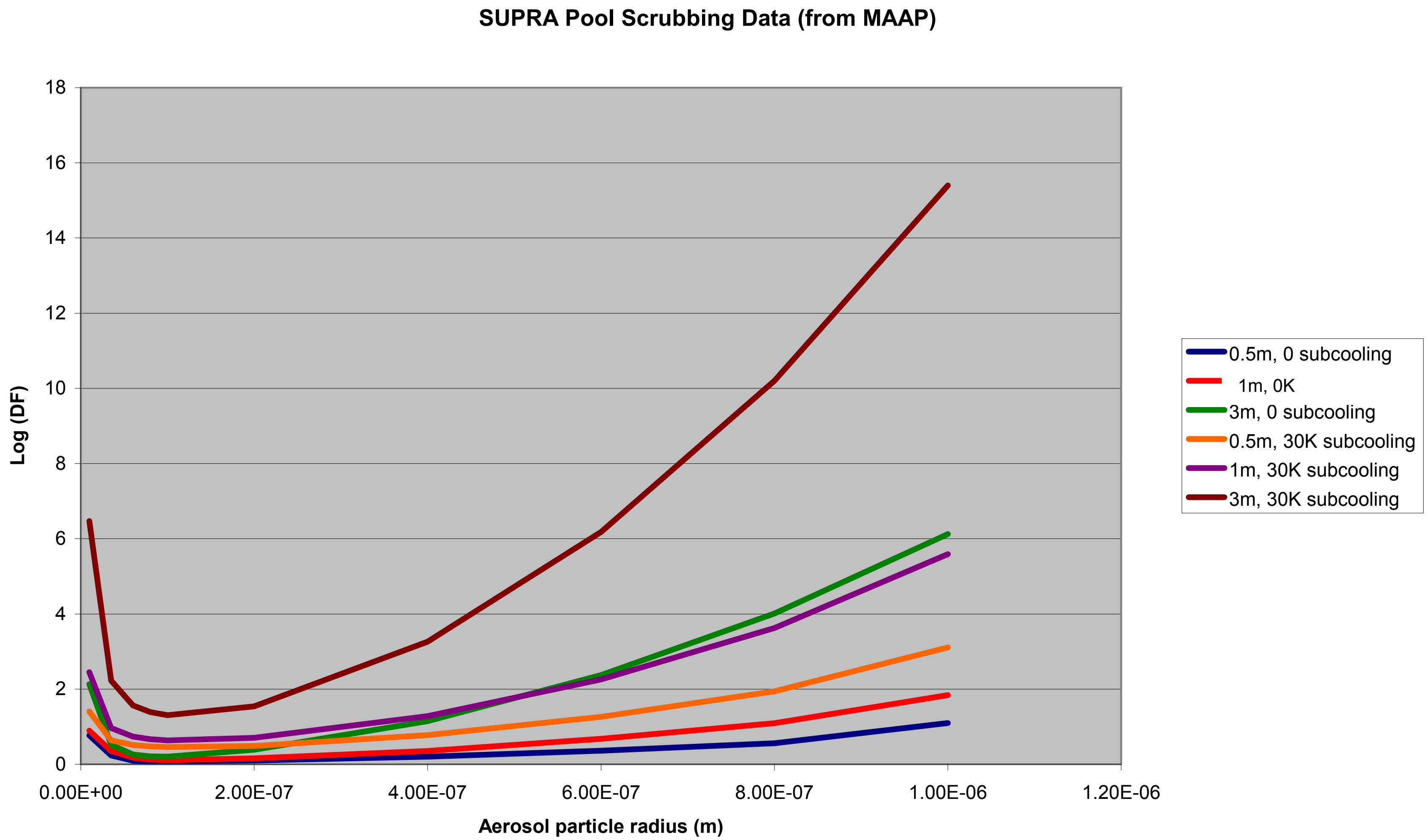


Figure 19-116-2—Pool Scrubbing Data for Side Venting

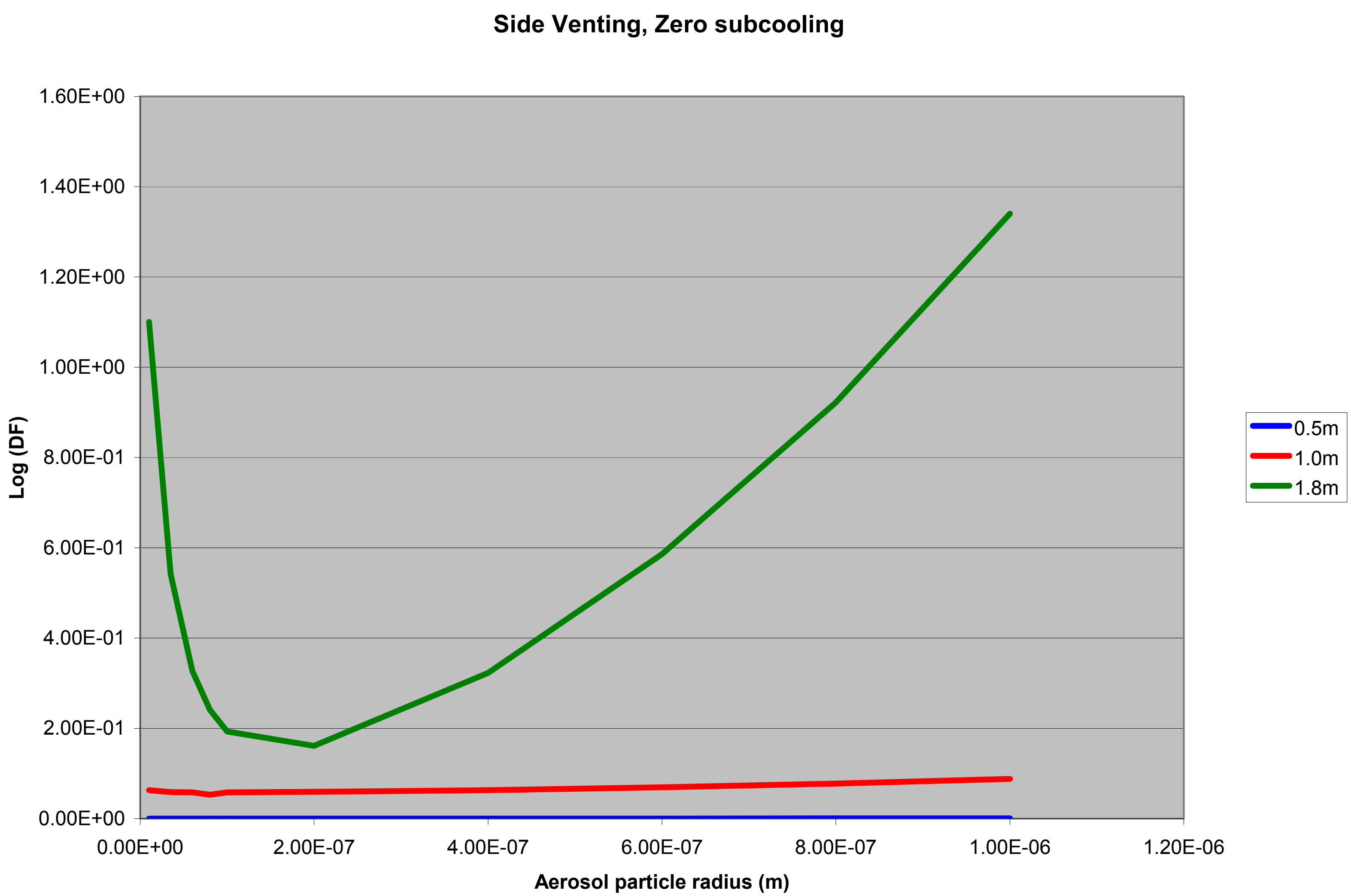


Figure 19-116-3—Log DF Vs. Pool Depth – 1 micron particle – Side Venting

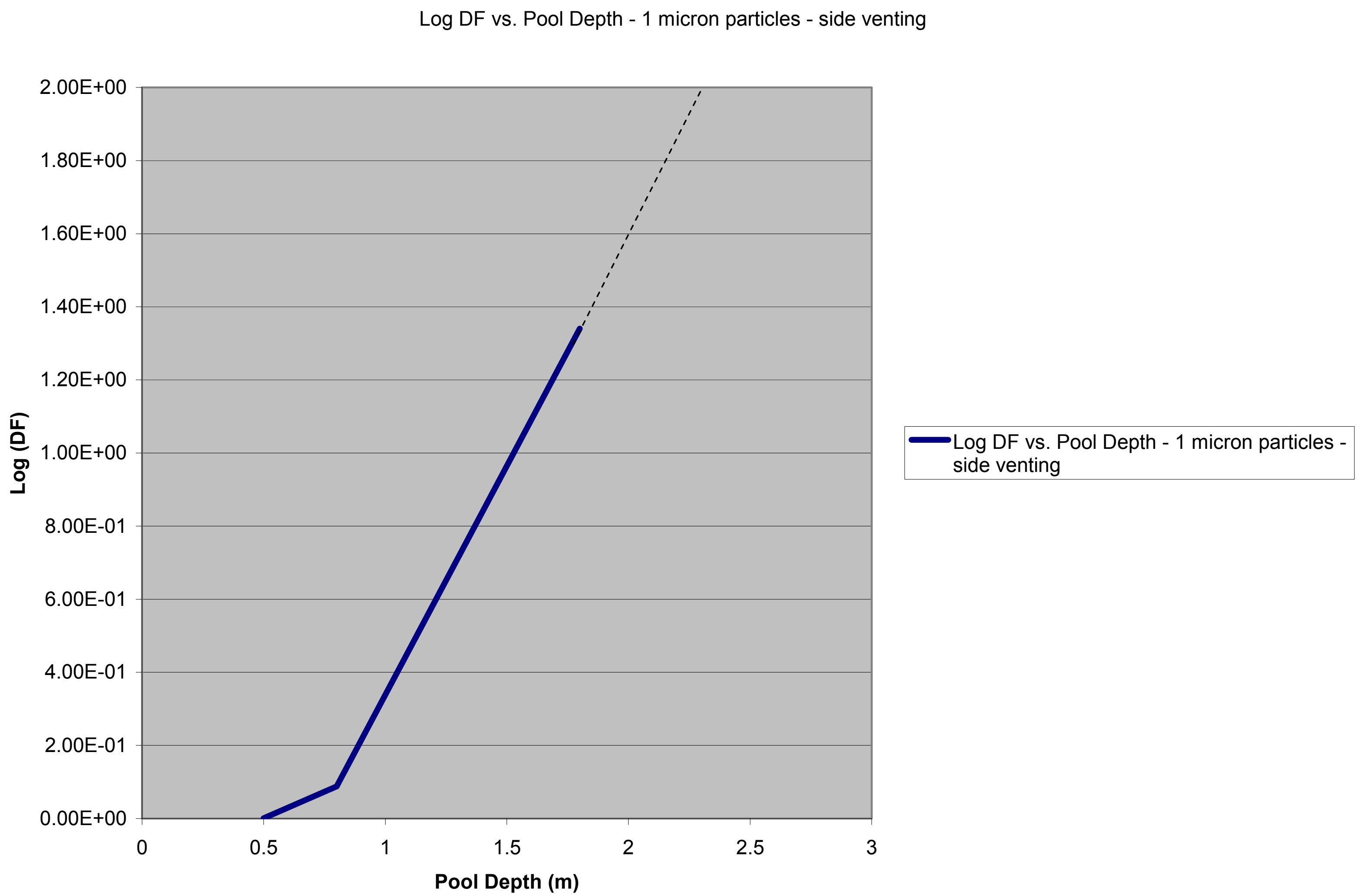
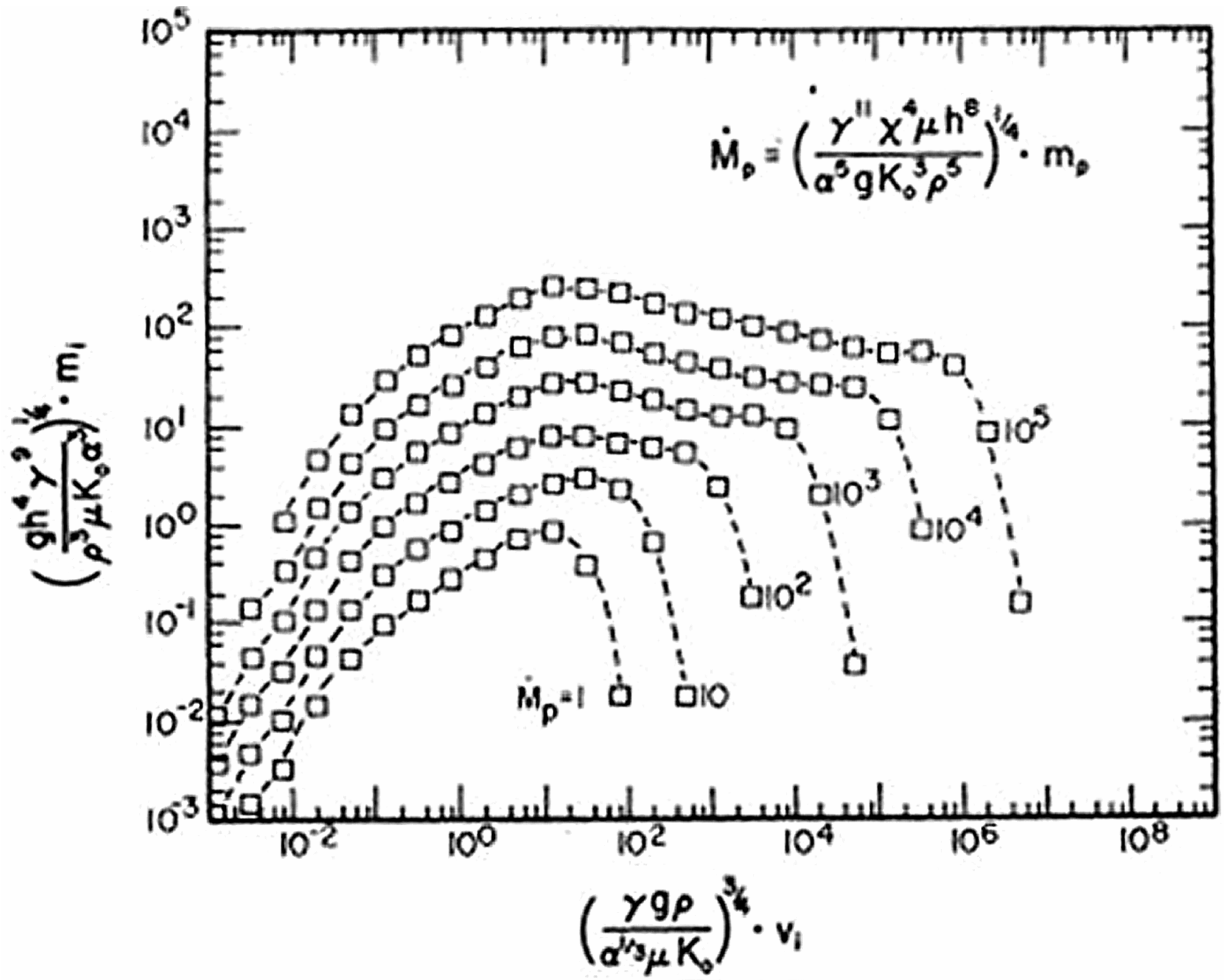


Figure 19-116-4—Steady State Aerosol Size Distribution for Different Non-Dimensional Source Rates (from Ref. 1)



References for 19-116:

1. Electric Power Research Institute, MAAP 4 Computer Code Manual, May 1994.

FSAR Impact:

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

Question 19-117:

In Table 19.1-20 of the PRA, release category 802 (ISLOCA) shows only about 0.028 maximum release fraction to the environment for aerosols and 0.39 for noble gases. If the radiological release were modeled as being directly to the environment, these figures would seem very low. Please justify the low values of radiological release calculated for this RC. If the reactor building was credited for decontamination, please justify the large DF (approximately 30 inferred), and include the volume, environmental leakage area or characteristics, and deposition or sedimentation surface area inside of the reactor building as used in this MAAP calculation. Please also provide a figure showing the pressure in the reactor building as a function of time as calculated for this scenario, and the basis for any assumptions regarding reactor building failure by overpressurization.

Response to Question 19-117:

A response to this question will be provided by August 8, 2008.

Question 19-118:

Table 19.1-20 shows release fractions to the environment calculated for various release categories. Represented by Cs, releases for the containment isolation failure release categories (i.e., RC2xx) are reported to be:

RC201	CIsF with melt retained in-vessel	0.084
RC202	CIsF, RV fails, dry MCCI with sprays	0.022
RC203	CIsF, RV fails, dry MCCI w/o sprays	0.023
RC204	CIsF, RV fails, wet MCCI with sprays	0.019
RC205	CIsF, RV fails, wet MCCI w/o sprays	0.026
RC206	CIsF (2" or less)	0.0082

Please explain why the case of isolation failure with melt retention results in the highest release magnitude (even higher than for vessel failure with dry MCCI and no sprays).

Response to Question 19-118:

A response to this question will be provided by August 8, 2008.

Question 19-119:

Please provide the results of the independent peer review of the Level 2 PRA, and a judgment regarding the capability categories of the model in key areas.

Response to Question 19-119:

A response to this question will be provided by August 8, 2008.

Question 19-120:

The Large Release classification specification utilized in the FSAR is adapted from guidance listed in Appendix A of NUREG/CR-6595 Rev.1. One limit is that any predicted I, Cs, or Te release fraction above approximately 2.5 to 3 percent is classified as large. At the time of development of that report, the highest licensed MWt for an operating commercial plant was (and is) that for the Palo Verde units at 3990 MWt (NUREG-1350, Vol. 19, Appendix A). The U.S.PWR plant design objective is presented as a rated output of 4614 MWt.

a. Please provide a listing of the equilibrium mid-cycle core radioisotope inventory (e.g., the 60 isotopes normally provided by ORIGEN).

b. Postulating that the NUREG/CR-6595 numbers were intended to provide an early fatality surrogate only for the spectrum of operating reactors and their releases, please provide a discussion of the results that would occur if the limit numbers were linearly scaled to reflect the above differences in plant thermal ratings.

Response to Question 19-120

Response to Question 19-120a:

Within the Level 2 PRA analysis, the fission product inventory used to perform the MAAP 4.0.7 source term runs was not an equilibrium mid-cycle core radioisotope inventory, but rather a bounding core inventory for the U.S. EPR. This bounding core inventory is shown in Table 19-120-1.

There are two isotopes missing from the 60 isotopes normally provided by ORIGEN: Co-58 and Co-60. These isotopes are in the core as a result of activation, not as a fission product, and were not included in the results of the ORIGEN runs for the U.S. EPR. . These two isotopes are not included in the 12 fission product groups modeled in MAAP 4.0.7, and therefore were not included in the source term analysis and subsequent offsite consequence analysis. Previous sensitivity studies for operating plants have shown no sensitivity in the offsite consequences to these two isotopes.

Response to Question 19-120b:

A response to this question will be provided by August 8, 2008.

FSAR Impact:

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

Table 19-120-1—U.S. EPR Core Inventory in Curies

Radioisotope	Bounding Core Inventory (curies)	Radioisotope	Bounding Core Inventory (curies)
Kr-85	2.10E+06	Te-132	1.98E+08
Kr-85m	4.50E+07	I-131	1.39E+08
Kr-87	9.02E+07	I-132	2.01E+08
Kr-88	1.28E+08	I-133	2.90E+08
Rb-86	5.80E+05	I-134	3.18E+08
Sr-89	1.61E+08	I-135	2.69E+08
Sr-90	1.69E+07	Xe-133	2.89E+08
Sr-91	2.07E+08	Xe-135	9.26E+07
Sr-92	2.14E+08	Cs-134	6.48E+07
Y-90	1.79E+07	Cs-136	1.61E+07
Y-91	1.96E+08	Cs-137	2.47E+07
Y-92	2.14E+08	Ba-139	2.62E+08
Y-93	2.34E+08	Ba-140	2.52E+08
Zr-95	2.29E+08	La-140	2.54E+08
Zr-97	2.43E+08	La-141	2.41E+08
Nb-95	2.29E+08	La-142	2.35E+08
Mo-99	2.59E+08	Ce-141	2.24E+08
Tc-99m	2.27E+08	Ce143	2.28E+08
Ru-103	2.42E+08	Ce-144	1.70E+08
Ru-105	1.96E+08	Pr-143	2.26E+08
Ru-106	1.43E+08	Nd-147	9.44E+07
Rh-105	1.75E+08	Np-239	3.82E+09
Sb-127	1.80E+07	Pu-238	1.46E+06
Sb-129	4.85E+07	Pu-239	6.14E+04
Te-127	1.79E+07	Pu-240	1.40E+05
Te-127m	2.43E+06	Pu-241	2.53E+07
Te-129	4.78E+07	Am-241	2.88E+04
Te-129m	7.08E+06	Cm-242	1.31E+07
Te-131m	2.04E+07	Cm-244	6.94E+06

Question 19-121:

In 10 CFR 52.47(b)(2) and 10 CFR 51.55 (a), the NRC requires the applicant to prepare an environmental report that includes the cost and benefits of severe accident mitigation design alternatives (SAMDA). The environmental report supporting the SAMDA analysis refers to a Level 3 PRA that was performed to determine the overall risk perspective of the U.S. EPR design. The analysis uses a standard method (template) as provided in the NEI 05-01 for SAMDA analysis in support of license renewal. Review of the methods and assumptions has identified a number of questions related to details of the analyses and conclusions that follow. These include:

a. The report does not provide any details on assumptions and data used in the Level 3 analysis. Please provide the assumptions and data used, including discussion of assumptions related to equilibrium core inventory (considering the potential for high burn up), source term, meteorology, population distribution, evacuation, sheltering, and any other data necessary as part of the input to the MACCS2 program. In addition, please supply the MACCS input that was used to perform the analysis

b. FSAR Figure 19.1-29 provides a range of values for mean and point estimates for internal events core damage frequencies. The core damage frequency used in the SAMDA analysis is 5.3E-07, which is a point estimate value. The mean CDF values given on the same figure has a maximum value 5.1E-06, almost an order of magnitude higher than the point estimate value that was used. Please justify the use of point estimate for the SAMDA analysis, and evaluate the impacts of using the mean CDF on the results of the cost benefit analysis.

c. NUREG/BR-0184 provides a range of doses and costs for occupational exposure, onsite clean up costs, and replacement power costs. The report used only the best estimate values with 3 and 7 per cent discount rates to estimate the range of potential averted costs. Given that these estimates are dated (1992 circa), please elaborate why the analysis did not consider the potential uncertainties in these values. Please evaluate the impact of uncertainties on the results of the cost benefit analysis.

d. In the screening of potential SAMDA, the analysis used the cost estimates from the license renewal submittals, in order to justify the cost effectiveness of implementation of specific SAMDAs. In an operating plant, a backfit may not be cost effective, but this may not be true for the new design. For the new design, the cost would be incremental addition, whereas, for a backfit is a total cost. Considering these observations, please describe if any one of the items screened out could become cost –beneficial.

e. Another screening criterion refers to “not required for design certification.” These are SAMDA items related to procedures and training. Please identify the screened-out list that needs to be considered by COL holders of US EPR design.

f. The basis statements for screening under the “Not Applicable” criterion are not always justified. For example, for SAMDA item “vent MSSVs in containment,” the basis for not being applicable is given as “Such a modification would pose design drawbacks, such as increased pressure loading and water inventory within containment, which would exceed the intended benefit of this SAMDA. ….” Please provide the impacts in terms of in containment pressure increase and water inventory, and show how these would be different than those of other severe accidents analyzed. Also elaborate on the intended benefit of this SAMDA item for the U.S. EPR design.

Response to Question 19-121:

A response to this question will be provided by August 8, 2008.

Question 19-122:

The MAAP 4.07 input deck specifies a total pressurizer free volume of 75 m3, which appears very close to the value that would be estimated from its inside geometry assuming it was completely empty and without accounting for internals such as pressurizer heaters. Please confirm whether pressurizer internals were considered in calculating the pressurizer free volume and, if not, then provide the volume associated with these internals.

Response to Question 19-122:

The free volume of the pressurizer was calculated using the simple geometry of the vessel and includes the water and steam (gas) volumes. The volume occupied by the sprays and heaters is neglected. This is a reasonable modeling simplification since the volume occupied is very small compared to the pressurizer free volume, as shown below.

The volume associated with the pressurizer heaters is **[ ]**, the heater support plate is **[ ]**, and the sprays occupy approximately **[ ]**. Thus, the total free volume of the pressurizer would be reduced by **[ ]** to account for the volume of the pressurizer heaters and sprays.

FSAR Impact:

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

Question 19-123:

Please estimates of the pressure drops and/or form loss coefficients in the reactor core region at normal operating conditions. These should include the values for the lower core support plate/structure, along the fuel rods, and across the upper core plate.

Resonse to Question 19-123:

The requested information was provided in the response to RAI 3 Question 19-47 part 2.

Question 19-124:

Please provide examples of several typical MAAP-calculated results for risk-dominant U.S. EPR accident scenarios, including the time evolution of:

- RCS pressure
- Hydrogen produced in-vessel
- Hydrogen mass inside containment (mass and concentration)
- Hydrogen consumed by PARs
- Oxygen (mass and concentration) inside containment
- Steam (mass and concentration) inside containment)
- Other non-condensable gases (mass and concentration) inside containment
- Level of water inside IRWST
- Level of water on the spreading floor, the melt discharge channel, the reactor pit, and other applicable elevations inside containment (showing water drainage into the IRWST)
- Core debris mass inside various containment regions (i.e., reactor pit, discharge channel and the spreading floor)
- Debris penetration distance (axial and radial) in the reactor pit, the discharge channel and the spreading floor
- The rate of release of various fission product groups inside the reactor, and on the containment floor.
- The airborne mass of various fission product groups inside the containment
- The mass of various fission product groups leaking out of the containment
- Total containment pressure (sum of all partial pressures).

Response to Question 19-124:

A response to this question will be provided by August 8, 2008.