ArevaEPRDCPEm Resource

From: WELLS Russell D (AREVA NP INC) [Russell.Wells@areva.com]
Sent: Monday, December 08, 2008 6:43 PM
To: Getachew Tesfaye
Cc: John Rycyna; Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19
Attachments: RAI 133 Response US EPR DC.pdf

Getachew,

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

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

Question #	Start Page	End Page
RAI 133 — 19-230	2	2
RAI 133 — 19-231	3	3
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RAI 133 — 19-246	53	79

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

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-232 (c)	February 13, 2009
RAI 133 — 19-233	February 13, 2009
RAI 133 — 19-236	January 30, 2009
RAI 133 — 19-237	January 30, 2009
RAI 133 — 19-238	February 13, 2009
RAI 133 — 19-240 (2)	February 13, 2009
RAI 133 — 19-242	January 30, 2009
RAI 133 — 19-243	June 19, 2009
RAI 133 — 19-244	March 6, 2009

(Russ Wells on behalf of)

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

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 29, 2008, and discussed with your staff on November 6, 2008. Draft RAI Questions 19-229 was deleted and Draft RAI Questions 19-238 was modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information wiwasll impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
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Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 18

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Response to

Request for Additional Information No. 133

11/07/2008

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

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

Question 19-230:

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

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

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

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

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

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

Response to Question 19-230:

A response to this question will be provided by June 19, 2009.

Question 19-231:

(Follow-up to Question 19-189) In response to Question 19-189, the applicant discusses material testing that has been performed for the U.S. EPR Zirconia bricks, including erosion tests for contact with oxidic molten core debris.

Please provide the results of these tests, including the specification of the stabilized Zirconia that will be used for U.S. EPR versus those tested.

Response to Question 19-231:

As stated in AREVA NP's response to RAI 45, Question 19-189, testing for the zirconia brick assembly identified for the U.S. EPR has been performed. The results report is available for NRC inspection at AREVA NP's Twinbrook, MD office.

The exact specification for the stabilized zirconia for the U.S. EPR will be developed later in the design process.

FSAR Impact:

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

Question 19-232:

(Follow-up to Question 19-84d) The response to Question 19-84d discusses the results of sensitivity cases that were investigated for source term calculations.

a. In discussions of the "effect of the SAHRS" (severe accident heat removal system), "isolation failure size," and "small leakage," the text refers to Figures 19-84d-1 through 19-84d-6 that are missing in the response. Please, provide these figures.

b. In discussions under "small leakage," the containment leak rate for 2- and 3-inch failure diameter (Figure 19-84-9d [SIC]), the response indicates that the flow rate for the 3-inch diameter case is conservatively calculated as (9/4) x (the leak rate for a 2-inch diameter). However, the figure illustrating the results show an inconsistency, namely, where the leak rate is lower for the 3-inch diameter case as compared to that for the 2-inch diameter break. Please, explain this apparent inconsistency.

c. In addition, in discussing the effects of molten corium-to-concrete interaction (MCCI), the response refers to Figures 19-84d-11 and 13 depicting lower values of SrO releases for cases involving MCCI as compared to accidents where MCCI is prevented. Please explain the phenomenological processes causing this apparent anomaly.

Response to Question 19-232:

Response to Part a

Figures 19-84d-1 through 19-84d-6 will be supplied in the revised response to RAI 6, Question 19-84. As noted in the response to Question 19-245, the revised response to RAI 6, Question 19-84 is submitted separately along with this response.

Response to Part b

As with part a above, the revised response to RAI 6, Question 19-84 contains the discussion of sensitivity cases for small leakage. This particular figure has been deleted and an alternative method for determining the size of small containment leakage is presented.

Response to Part c

A response to this question will be provided by February 13, 2009.

Question 19-233:

(Follow-up to Question 19-84e) The response to Question 19-84e discusses the assumptions applicable to analysis of steam generator tube rupture (SGTR) source terms (i.e., Release Category 702). The response confirms that the analysis assumes the failure of a single tube, whereas the results of a preliminary sensitivity study by AREVA shows the increase in fission product source term under the postulated condition of more than a single tube rupture. Please address the likelihood for multiple SGTRs and provide associated source terms.

Response to Question 19-233:

A response to this question will be provided by February 13, 2009.

Question 19-234:

(Follow-up to Question 19-84c) The response to Question 19-84c indicates alarm timing errors that were used in the consequence analysis, and provides a table comparing the results for "Base" and "All." The error in alarm timing affects many release categories. Please provide an explanation of the meaning of "Base" and "All".

Response to Question 19-234:

In RAI 6, Question 19-84, Table 19-84c-5, the results for the "Base" case refers to the original Base Case that used the incorrect OALARM number. The "All" case refers to Sensitivity Case A11 that used the correct OALARM values, and is documented in a revised analysis. The effect of changing the OALARM value on the overall risk is negligible.

The supporting release category runs were re-run to address the core inventory input error. When these cases were re-run, the corrected OALARM value was used. The updated information is provided in the re-submittal of RAI 6, Question 19-84, which is addressed in the response to Question 19-245.

FSAR Impact:

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

Question 19-235:

(Follow-up to Question 19-114) The response to Question 19-114 discusses flame acceleration and analysis approach on the uncertainty of zirconium oxidation. The method relied on realistic values of hydrogen production, which were lower than 100% zirconium oxidation. Please explain the potential for flame acceleration and detonation, if one were to consider hydrogen masses corresponding to 100% of the equivalent in-vessel zirconium oxidation (see 10 CFR 50.44(c).

Response to Question 19-235:

The analysis that shows flame acceleration and detonation is avoided, which is discussed in the response to RAI 6, Question 19-114, does account for hydrogen masses corresponding to 100 percent of the equivalent in-vessel zirconium oxidation. U.S. EPR FSAR Tier 2, Section 19.2.4.2.1 discusses the methods used to address the regulatory considerations of 10 CFR 50.44. It is not possible using MAAP4, to allow all fuel cladding to interact with the coolant to produce all necessary hydrogen. Therefore, to simulate at least a 100 percent cladding-coolant hydrogen reaction an alternate method was employed. This alternate method applies several model biases to promote hydrogen production while prohibiting the auto-ignition of hydrogen created from the molten core-coolant interaction (MCCI), which would physically auto-ignite. The additional hydrogen added to the containment through this method more than compensates for the lack of in-vessel hydrogen generation. U.S. EPR FSAR Tier 2, Section 19.2.4.4.1.2 discusses this method in further detail, indicating that the total hydrogen generated in-vessel when 100 percent of the cladding reacts with coolant is 2000 lbm. To meet the requirement of 10 CFR 50.44, a minimum of 2000 lbm should be generated for U.S. EPR severe accident scenarios.

The response to RAI 6, Question 19-114 was not intended to imply that it was necessary to produce less hydrogen than the equivalent of 100 percent cladding-coolant oxidation to pass the sigma criteria. For those scenarios that did not originally pass the sigma criteria, they produced hydrogen in excess of the 100 percent cladding-coolant oxidation requirement. The addition of hydrogen to the containment through MCCI is negligible because of the auto-ignition phenomenon. The ex-vessel hydrogen was merely added to the original 59 scenarios of the uncertainty analysis to produce 100 percent cladding-coolant oxidation equivalent hydrogen for all scenarios. Therefore, for those few scenarios that did not pass the sigma criteria with the excess hydrogen, the input parameters affecting its production were modified to values consistent with the 100 percent cladding-coolant oxidation requirement. The results of these scenarios still produce more than 2000 lbm of hydrogen to satisfy 10 CFR 50.44, yet no longer violate the sigma criteria. Because no scenarios violate the sigma criteria while still satisfying the requirements of 10 CFR 50.44, there is no potential for flame acceleration or detonation.

FSAR Impact:

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

Question 19-236:

(Follow-up to Question 19-121b) The response to Question 19-121b discusses the effect of using the mean core damage frequency (CDF) as opposed to using point estimate CDF value for cost-benefit analysis of SAMDAs, and it indicates a potential increase in maximum benefit of a factor of 8 if the mean CDF values were used. The NRC guidance, as well as the American Society of Mechanical Engineers (ASME) PRA standard, specifically requests mean values rather than point estimates (see Regulatory Guide 1.174, Section 2.2.5.5; Regulatory Guide 1.206, Section C.I.19, Appendix A; SRP 19.0, Section III; ASME RA-Sb-2005, Supporting Requirement QUA2b). Therefore, please explain why the mean CDF was not used in the evaluation.

Response to Question 19-236:

A response to this question will be provided by January 30, 2009.

Question 19-237:

(Follow-up to Question 19-121c) The response to Question 19-121c discusses five sensitivity analyses that were performed to assess the uncertainties in the maximum benefit calculations. The calculations were performed on a segregated basis; i.e., considering individual effects one at a time. This method would not provide the true effects of combination of uncertain inputs. For example, updating the constant, representing a string of replacement power costs provided in terms of “1993 dollars” in the NUREG/BR-0184, to represent a value in “2008 dollars” is not a sensitivity analysis, and perhaps needs to be used as the basic input for the analysis.

In addition, the present value of the replacement power for a single event of $1.4E+9 (in “1993 dollars”) with a discount rate of 3 percent in the NUREG/BR-0184 corresponds to an average reactor life of 24 years. This value should be adjusted for the reactor life of 60 years, which is applicable to the U.S. EPR. Therefore, please explain why a consolidated calculation using the upper bound estimates for the onsite clean up cost, the onsite plant personnel dose, and the inflation adjusted replacement power costs in combination with a 3 percent discount rate (as recommended in NUREG/BR-0058, and used in the ANP-10290) was not performed to estimate the maximum benefit. Note that the final results should consider the mean CDF as discussed in the follow–up to Question 19-121b.

Response to Question 19-237:

A response to this question will be provided by January 30, 2009.

Question 19-238:

(Follow-up to Question 19-121f) The response to Question 19-121f indicates that if the released water/steam through the safety relief valve following a SGTR accident were to be injected (through a specially designed system) back into the containment, the resulting containment pressure would be comparable with containment loads for other severe accident scenarios that have been analyzed (see Figure 19-121-1 and 19-121-3). In fact, the associated loads both in terms of pressure and water loading on the containment should be similar to that of a small steam line break, with a known release path to a specially designed in-containment quench/scrub system. Therefore, at least from the safety point of view, this appears to be a viable alternative that should have been considered as part of the SAMDA cost benefit studies. Please perform an analysis of this SAMDA to determine if it merits implementation into the design, or explain why it is not necessary to do so.

Response to Question 19-238:

A response to this question will be provided by February 13, 2009.

Question 19-239:

(Follow-up to Question 19-157) The response to Question 19-157 discusses containment isolation failure and its contribution to large early release. While the split between large and small isolation failures is deducible from careful reading of the response, the bulleted specifications for the top events conflict with the rest of the text and would benefit from some rewriting.

a) The top event “CI Large” description in the first bulleted item, as written, defines leakage from two or more lines (of any size however small) as a large release. The bands for single line leaks are not contiguous (currently 2” or less for CI Small and greater than 3” for CI Large for single line leaks.)

b) The response indicates that the annual frequency of release due to large containment isolation failures (defined as release category [RC] 201—205) is 9.30E-10, contributing 0.2 percent of LRF. This LRF frequency is the sum of frequencies from internal events (internal, fire and floods) as given in Tables 19.1-24, 19.1-50, and 19.1-75. The total LRF value from these tables is 2.67E-08, which makes contribution from large containment isolation failures to be about 3.5 percent of the internal events LRF, and not 0.2 percent as indicated in the response. Please clarify whether the total LRF values for internal events have changed, if so please provide the new and revised estimates and please update the affected tables.

c) In addition, in the response to Question 19-84a, RC 206 is identified as large early release, and considered as consequence dominant for Level 3 analysis. However, Table 19.1-24 of the Final Safety Analysis Report (FSAR), neither identifies this release category as large release, nor provides its frequency. Please clarify this inconsistency, and if RC 206 is considered as part of LRF, then the affected FSAR tables for internal events need to be revised.

Response to Question 19-239:

Response to Question 19-239a:

As described in the response to RAI 22, Question 19-157, the examination of the containment isolation failures sizes resulted in the following definitions:

Small containment isolation failures: includes all releases of fission products from a single line with a diameter equal or less than 2 inches.

Large containment isolation failures: includes any leak from two or more lines independent of their diameters, or leakage from one line larger than 2 inches in diameter.

The description of containment isolation sizes presented in the response to RAI 22, Question 19-157 appears to be discontinuous because the actual lines which are modeled in the containment isolation are either 2 inches internal diameter (ID) lines, or larger than 3 in, (i.e., there is no line in the 2 to 3 inch ID range).

Based on this definition, the large containment isolation failures are assigned to release categories (RC) RC 201, RC 202, RC 203, RC 204, and RC 205. Small containment isolation failures are assigned to RC 206.

Response to Question 19-239b:

As indicated in the response to RAI 22, Question 19-157, the annual frequency of release due to large containment isolation failure is 9.3E-10/yr. This corresponds to:

- 0.2 percent of the total core damage frequency (CDF) (5.3E-07/yr)
- 3.5 percent of the total large release frequency (LRF) (2.6E-08/yr)

The percentage presented in the response to RAI 22, Question 19-157 reflects the containment isolation failure probability given core damage, not the contribution to total LRF.

Therefore, the correct contribution of large containment isolation failures to the total LRF is 3.5 percent. Thus, the response to RAI 22, Question 19-157 should have stated the percentage of large and small isolation failures as a fraction of the total CDF, not LRF, as follows:

"The annual frequency of a release due to a large containment isolation failure is 9.3E-10/yr (0.2 percent of CDF). The annual frequency of a release due to a small containment isolation failure is 1.6E-08/yr (3.1 percent of CDF)."

Response to Question 19-239c:

RC 206 is not considered a large release category, based on the criterion of the predicted I, Cs, and Te percentage presence in the source term. This criterion is provided in U.S. EPR FSAR Tier 2, Section 19.1.1.2.1.3 and further discussed in response to RAI 6, Question 120.

The response to RAI 6, Question 19-84a, which identifies all RC 20x as large releases should have identified only RC201 to RC205 as large releases. The U.S. EPR FSAR Tier 2 Table 19.1-24 is correct.

There is an inconsistency between Level 3 PRA results for the number of early fatalities and the large release criterion above. The criterion for classifying a release category as "large" is derived from Appendix A of NUREG/CR-6595, which uses this criterion as a substitute for "one mean fatality at one mile." The Level 3 PRA shows that the early fatalities are less than one at one mile for each one of the RC200s. Based solely on the "one mean fatality at one mile" criterion, one could exclude all the RC 200s from the calculation of LRF. Therefore, the use of this criterion as the primary criterion in defining large release is conservative. Based on this criterion, RC 201 through 205 are selected as large releases.

FSAR Impact:

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

Question 19-240:

(Follow-up to Question 19-79)

1. Please confirm the relationship for the stress and Larson-Miller parameter given in Table 19-79-1 of the response to RAI 19-79. Specifically, should LMP be multiplied by the fitting parameter bfit or added to it?
2. More information is needed to clarify the response to question 19-79a, as summarized below.
   - Since cases where the secondary side is depressurized dominate the risk in current PWRs, please run a variation of the MAAP base case with degraded tubes where the steam generators are depressurized and report the damage fractions and the times of predicted hot leg failure, tube failure, and when the core outlet temperature reaches 650 C. Also provide plots of the steam generator tube temperatures.
   - If there are any relevant PRA sequences that would include the 2" diameter breaks, please identify them, and state their core damage frequencies. For MAAP case 1.1i (depressurized SGs), tube failure is estimated to occur about 25 minutes after the core outlet temperature reaches 650 C (11918 sec. vs. 10440 sec.). How does this relatively short time interval affect potential severe accident management high-level actions and strategies? What would be the tube failure times for variations of this case where tubes are damaged, in particular for 1/2, 2/3, and 3/4 through-wall volumetric degradation (MAAP parameter FERSGT=0.5, 0.33 and 0.25). Such damage can realistically be expected to occur from foreign object wear above the top of the tube sheet.
3. The response to Question 19-79e states that system-related top events in the containment event trees are used to model the status of containment isolation, safety injection, severe accident heat removal, and other systems. But FSAR Section 19.1.4.2.1.3 states that the CDES are not, however, directly transferred to Level 2 CETs. Rather, each individual end state is transferred through an intermediate event tree, referred to as CDES link event tree, prior to transfer to a Level 2 CET. Please provide the details of these link event trees, including diagrams, tables of grouping and quantifications, how they are applied to each of the CDESs and how they are linked to the Level 2 CETs that are shown in FSAR Appendix 19C.

Response to Question 19-240:

Response to 19-240, 1:

The correct relationship for the stress and Larson-Miller parameter is:

$$Ln(\sigma) = m_{fit} \cdot LMP + b_{fit}$$

Where:

$\sigma = \dfrac{\Delta P \cdot r}{x}$ is the stress for cylindrical tubes or pipes

With:

ΔP = pressure difference across the pipe wall (Pa)

r = pipe radius (m) and

x = pipe wall thickness (m)

The formula presented in response to Question 19-79 incorrectly reversed m_{fit} and b_{fit}. The analyses were performed with the correct formula.

Response to 19-240, 2:

A response to this question will be provided by February 13, 2009.

Response to 19-240, 3

Each of the core damage end states (CDES) listed in U.S. EPR FSAR Tier 2, Table 19.1-16 is associated with a link tree which transfers the core damage sequences to the adequate Level 2 containment event tree (CET), while preserving the Level 1 accident sequence information.

The structure of a link tree is as follows:

- The initiating event is the CDES.
- A flag basic event is added to the sequence in order to label it with the appropriate CDES.
- Additional function events are evaluated, if needed, in order to bin the sequence into the appropriate CET.
- The consequence is the corresponding CET for the sequence.

Link trees used for CDES: AT, ATI, IS, LL, SG, SG3, SL, SLD, SS, SSD, TR and TRD only add a flag to the sequence before sending it to the appropriate CET. The correspondence between CDES and CETs is shown in U.S. EPR FSAR Tier 2, Table 19.1-16.

For the other CDES, link trees are also used to select the appropriate CET and gather additional information, such as:

- Recovery of offsite power within the different severe accident timeframes for loss of offsite power (LOOP) sequences.
- Success criteria for limited core damage sequences.
- Availability of feedwater for steam generator tube rupture (SGTR) sequences

Table 19-240-1 describes the function of these more complex link trees. The link trees are shown in Figure 19-240-1.

Table 19-240-1—Link Tree Description

Event Tree "Initiating" Top Event	Input CDES	Top Events Description
#CDES-LL1	LL1	If LHSI (injection and long term heat removal) is successful the sequence is sent to #CET LIMITED CD otherwise it is sent to #CET LO PRESSURE
#CDES-ML	ML	
#CDES-PL	PL	
#CDES-SG1	SG1	If both LHSI (injection and long term heat removal) and feed and bleed are successful the CDES is sent to #CET LIMITED CD otherwise it is sent to the #CET SGTR
#CDES-SL1	SL1	If both LHSI (injection and long term heat removal) and feed and bleed are successful the CDES is sent to #CET LIMITED CD otherwise it is sent to the #CET1 HI PRESSURE
#CDES-SL1D	SL1D	
#CDES-SS1	SS1	
#CDES-SS1D	SS1D	
#CDES-TR1	TR1	
#CDES-TR1D	TR1D	
#CDES-SP	SP	The link tree asks for recovery of off site power within 2 to 7 hrs (given no recovery in 1 hour asked in the level1) or between 7 and 31 hours.
#CDES-SPD	SPD	
#CDES-TP	TP	
#CDES-SP1	SP1	The success of LHSI and feed and bleed leads to #CET LIMITED CD. All other sequences are sent to #CET1 HI PRESSURE. Recovery of LOOP between 2-7hrs and recovery of LOOP between 7-31hrs is asked
#CDES-SP1D	SP1D	
#CDES-TP1	TP1	
#CDES-SG2	SG2	If feed water to the ruptured SG is successful the CDES is sent to #CET SGTR FW otherwise it is sent to #CET SGTR

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Figure 19-240-1—U.S. EPR PRA CDES Link Trees
Sheet 1 of 29

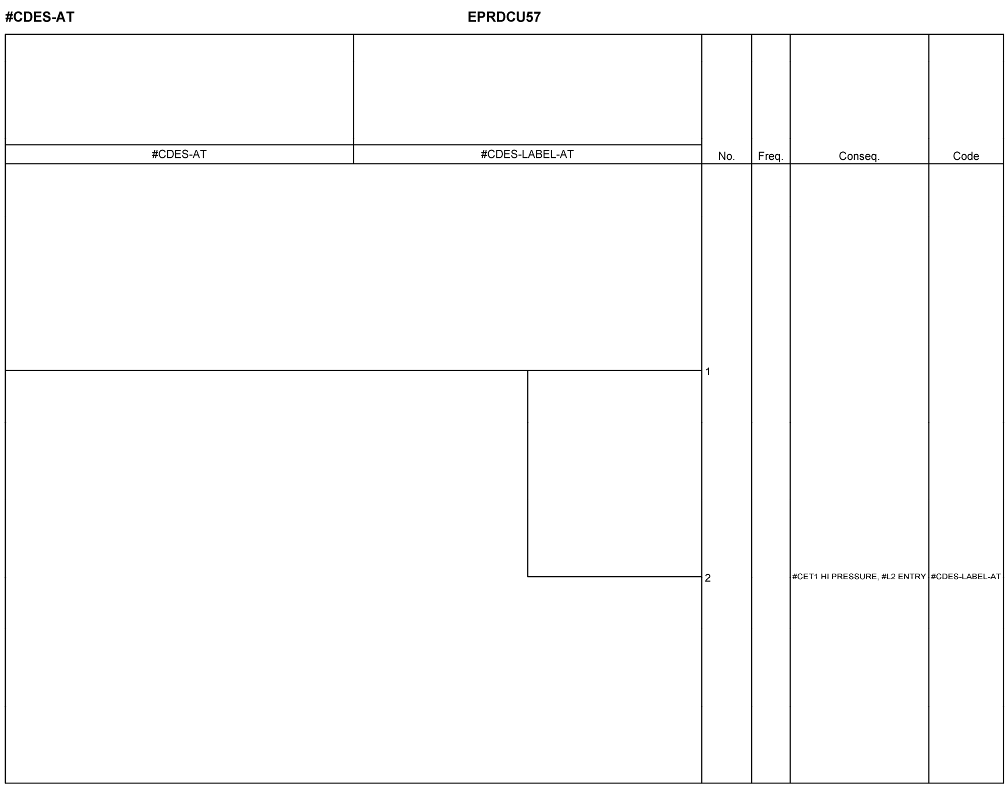

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
Sheet 2 of 29

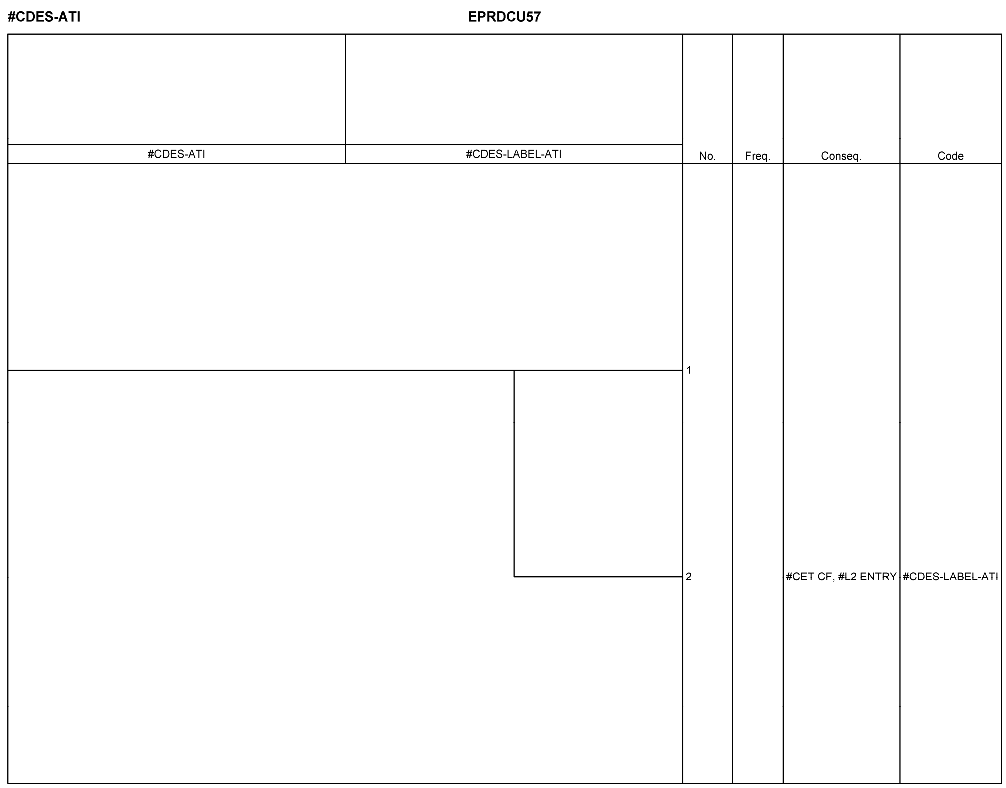

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
Sheet 3 of 29

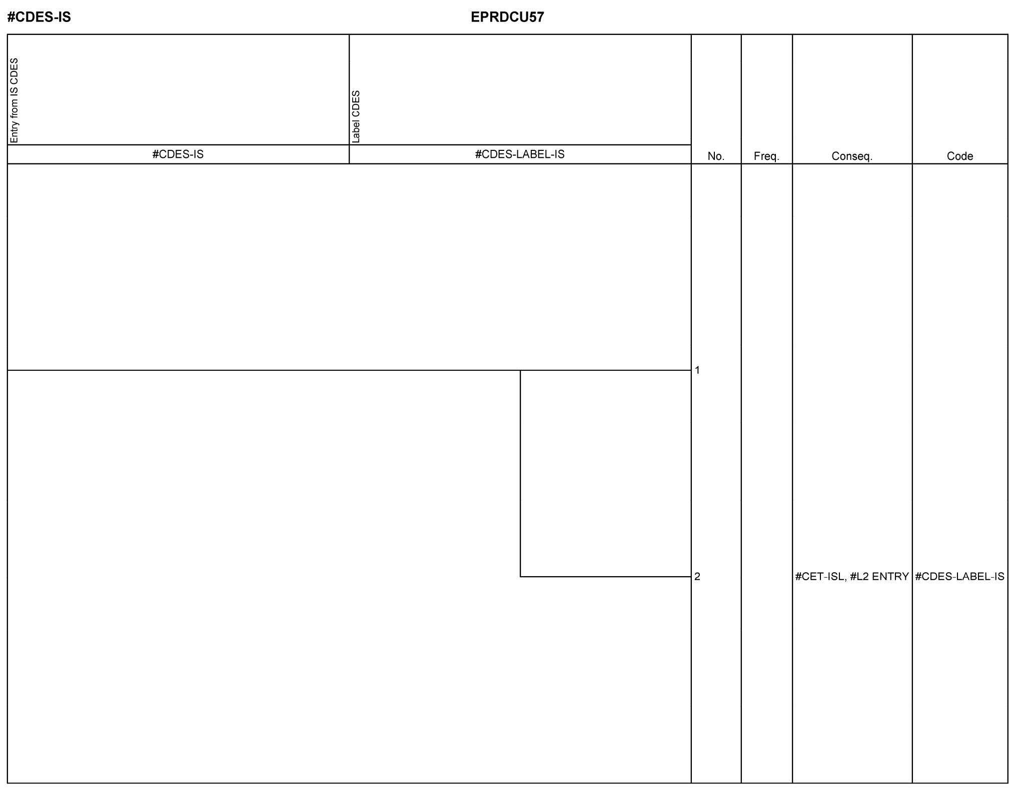

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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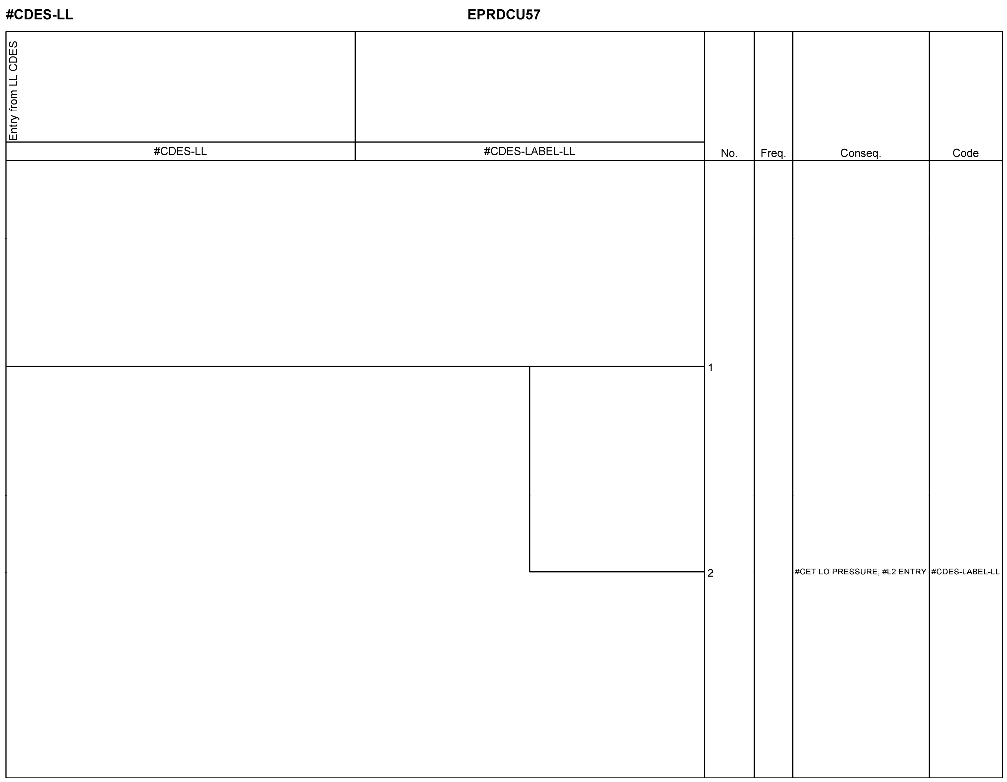

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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#CDES-LL1 **EPRDCU57**

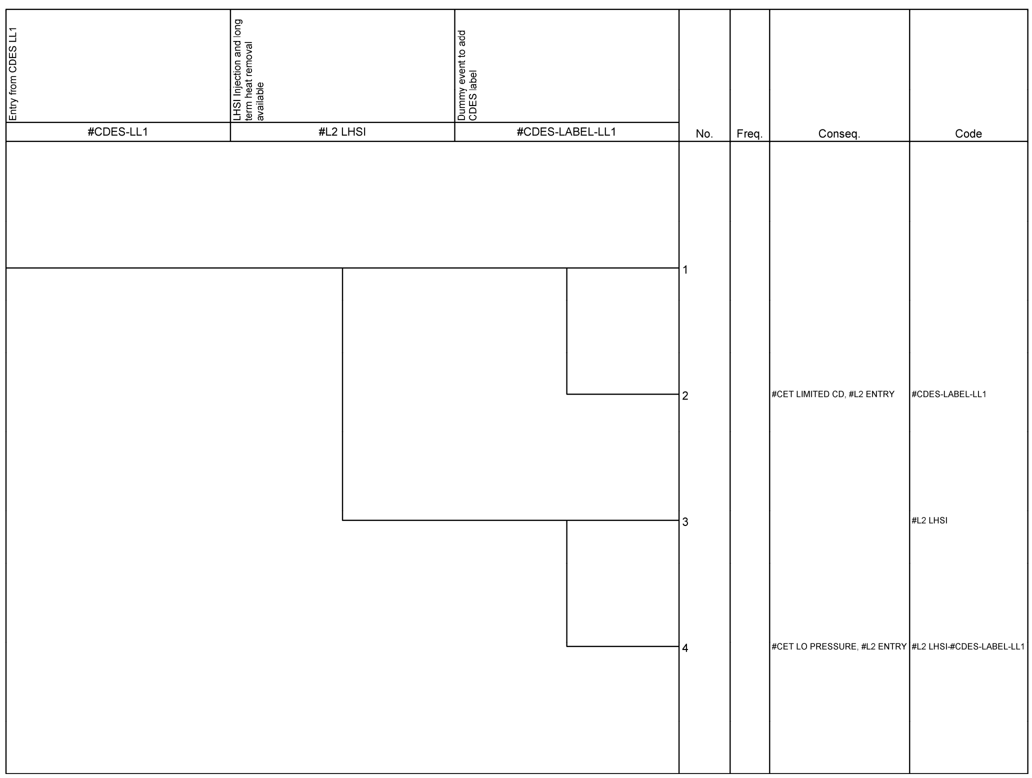

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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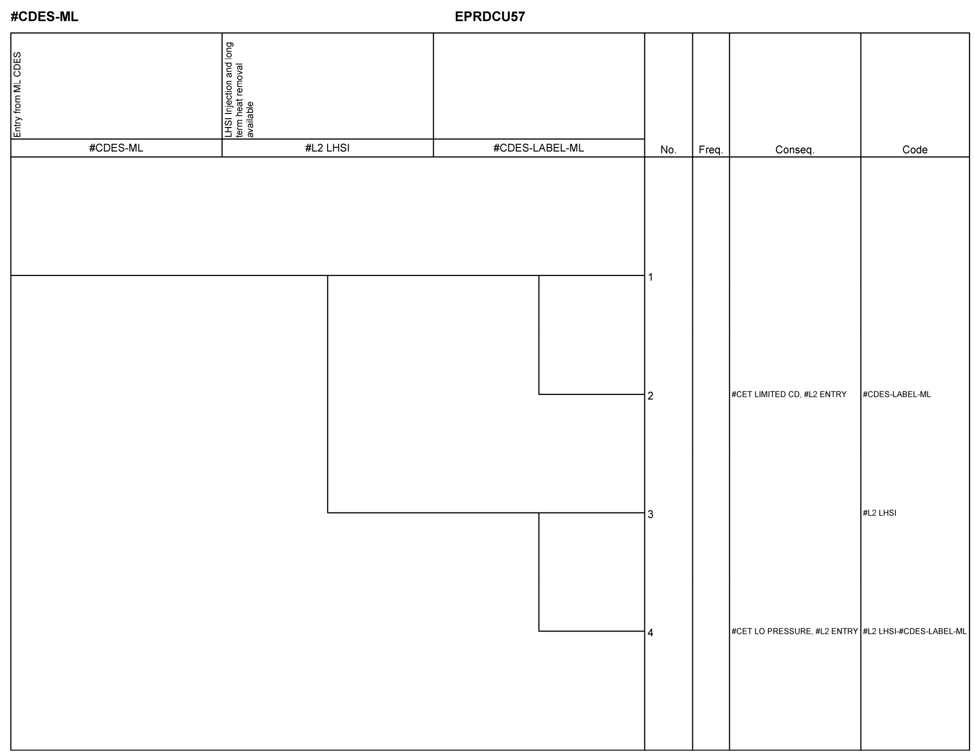

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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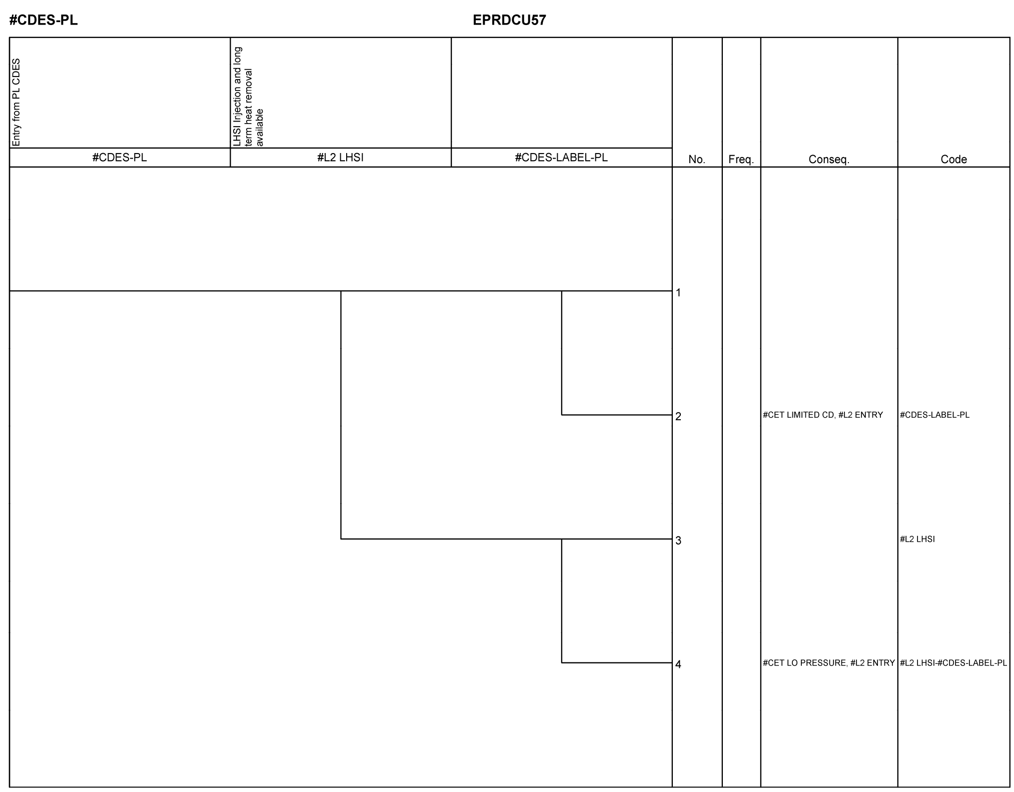

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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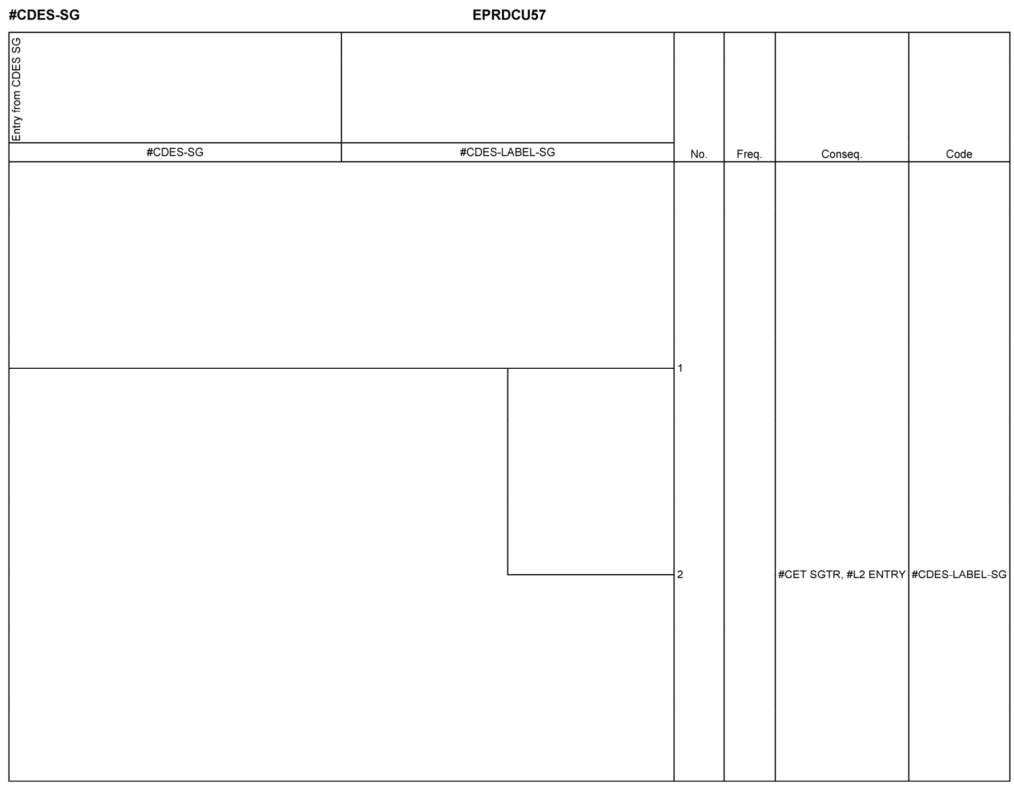

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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#CDES-SG1 **EPRDCU57**

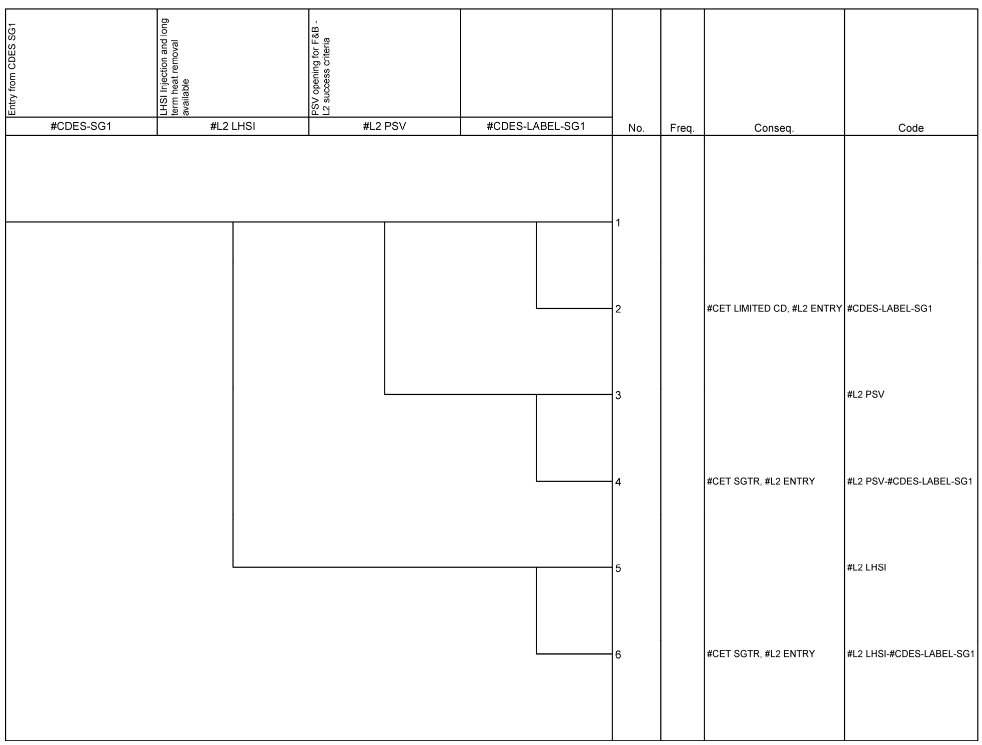

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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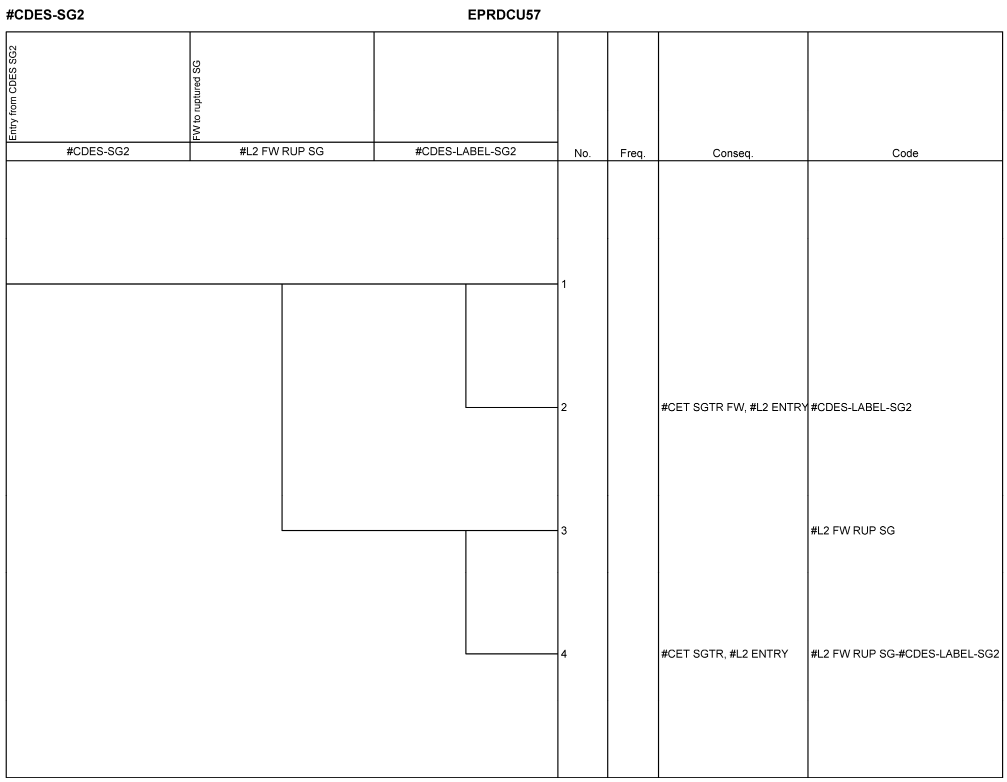

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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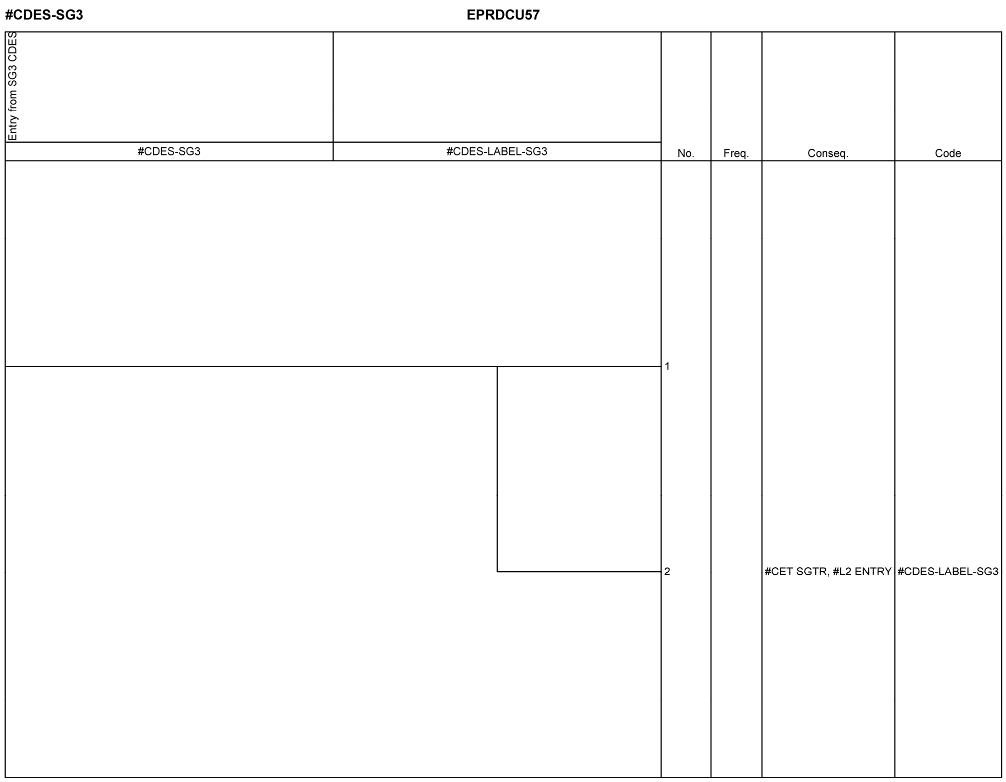

**Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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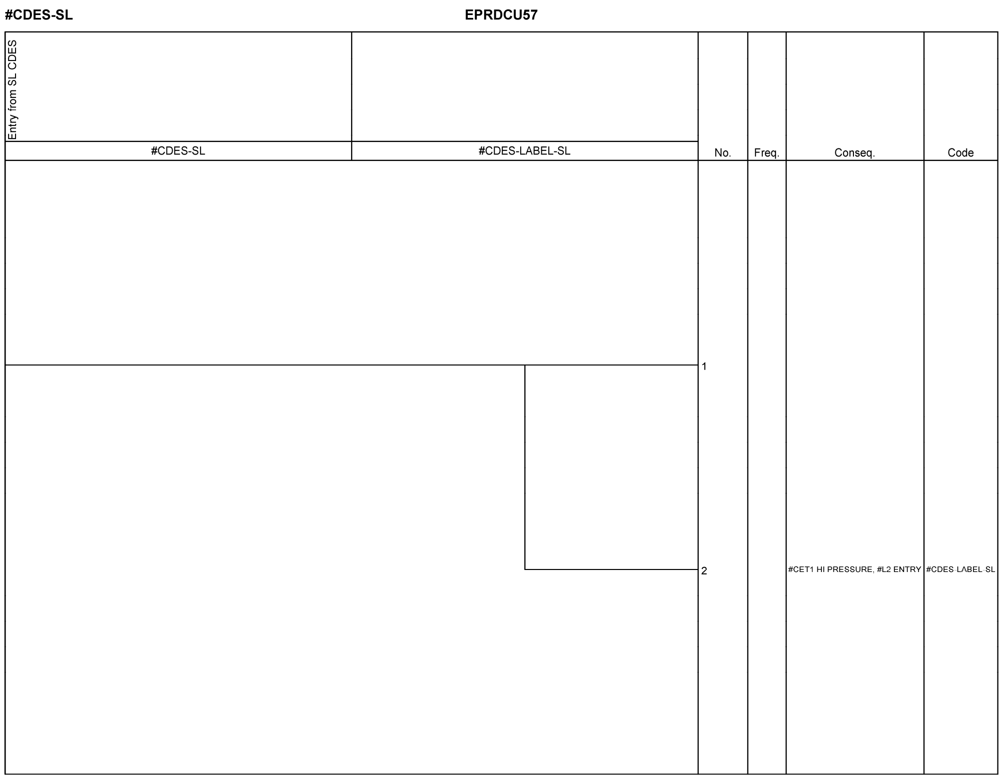

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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#CDES-SL1 **EPRDCU57**

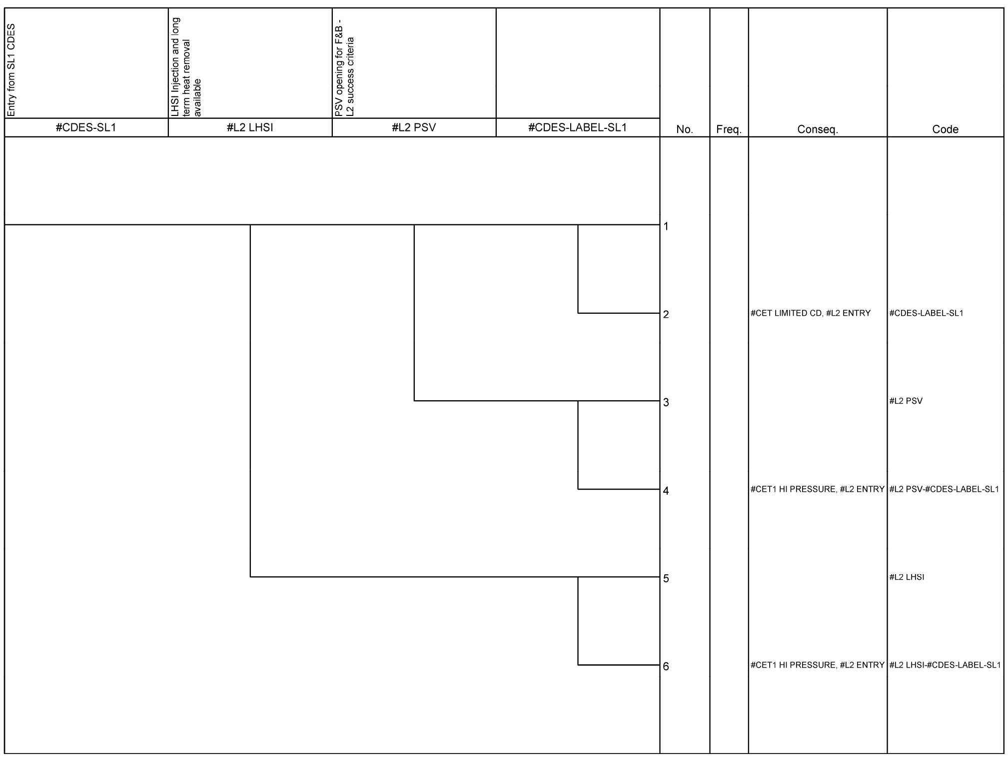

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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#CDES-SL1D **EPRDCU57**

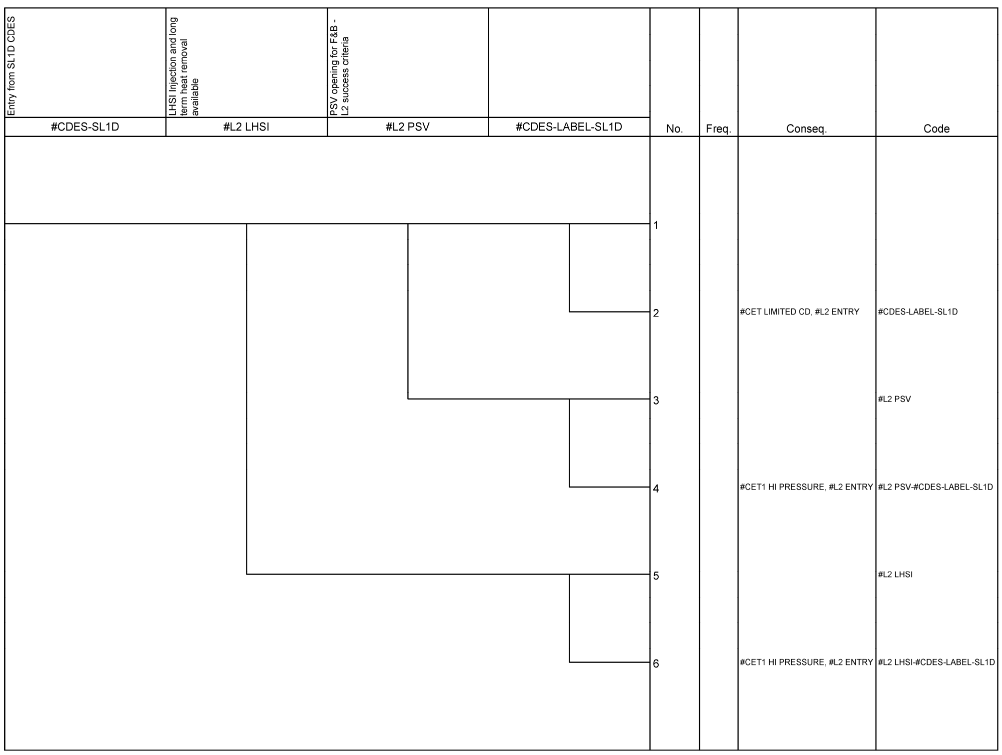

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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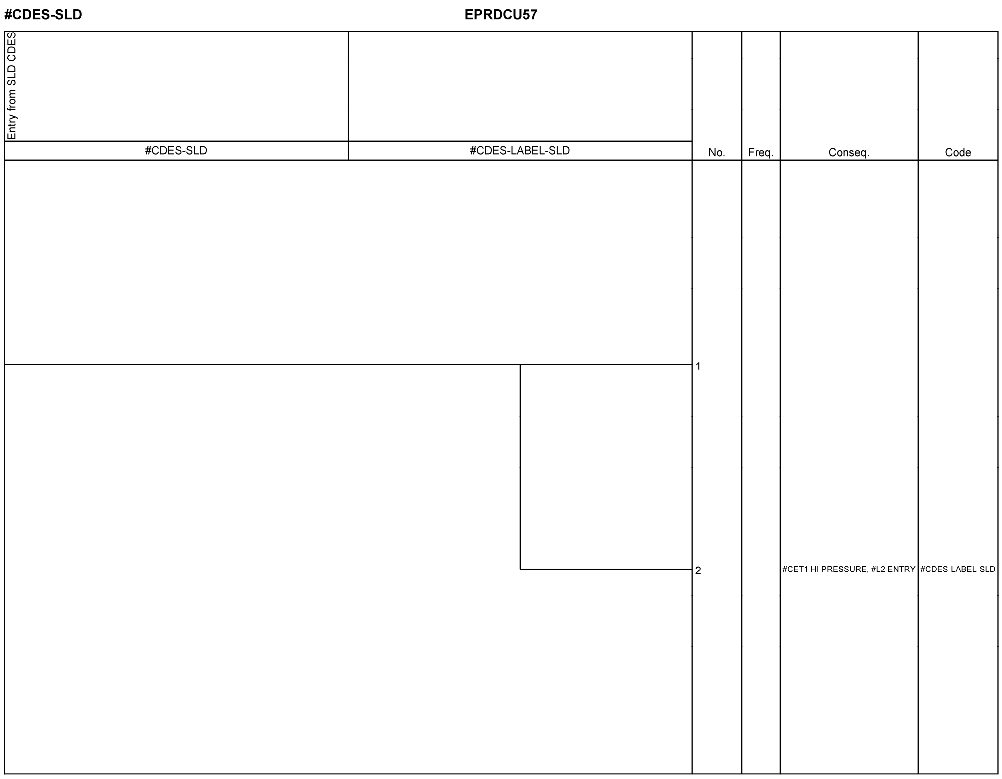

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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#CDES-SP **EPRDCU57**

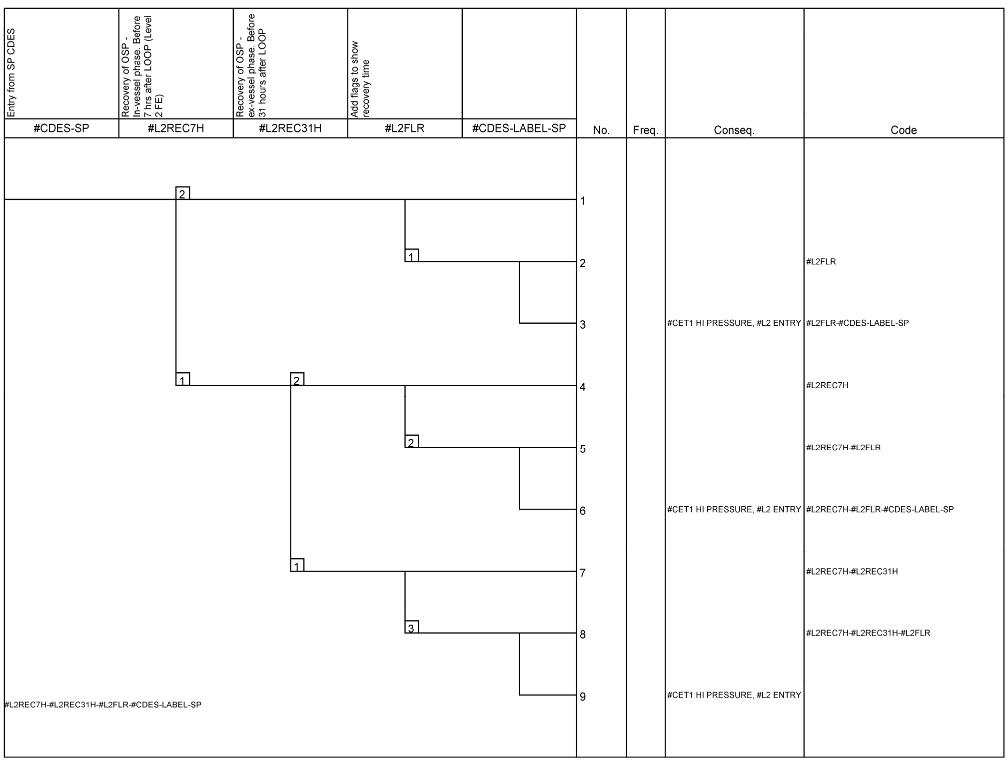

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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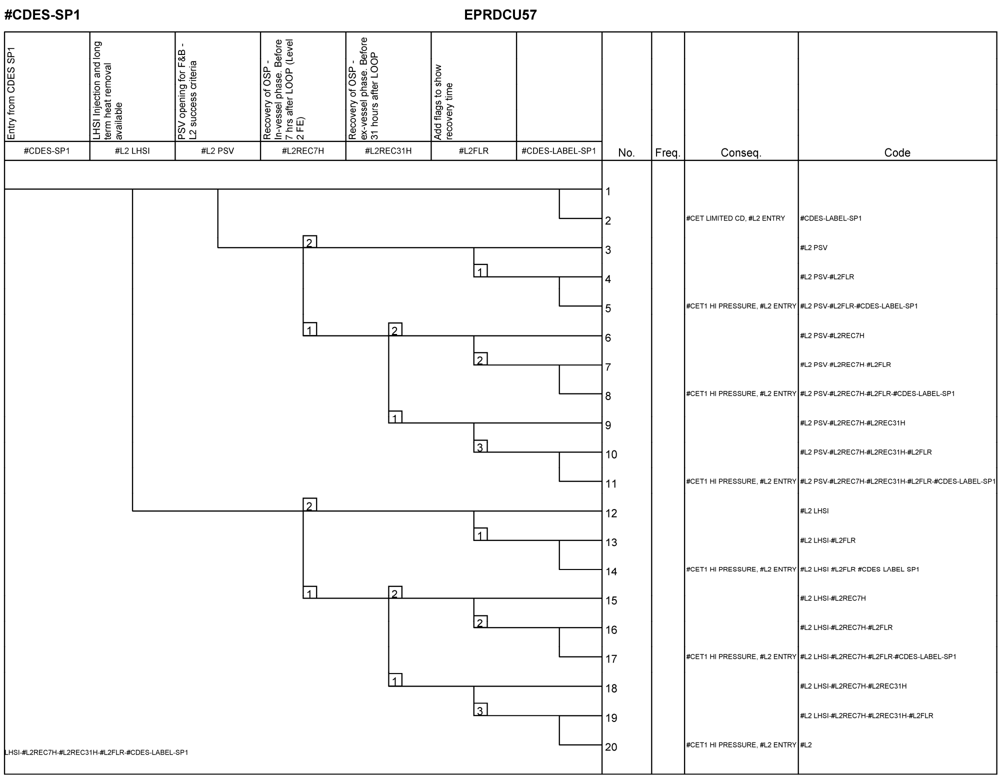

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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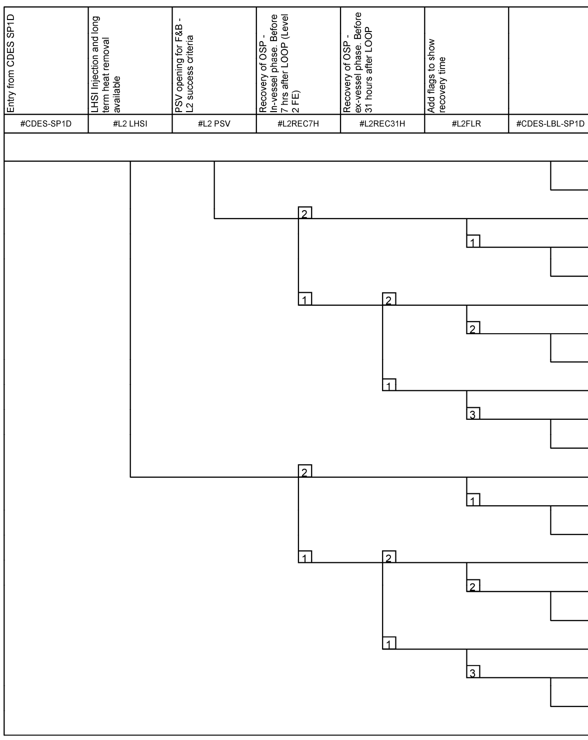
#CDES-SP1D **EPRDCU57**



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5		#CET1 HI PRESSURE, #L2 ENTRY	#L2 PSV-#L2FLR-#CDES-LBL-SP1D
6			#L2 PSV-#L2REC7H
7			#L2 PSV-#L2REC7H-#L2FLR
8		#CET1 HI PRESSURE, #L2 ENTRY	#L2 PSV-#L2REC7H-#L2FLR-#CDES-LBL-SP1D
9			#L2 PSV-#L2REC7H-#L2REC31H
10			#L2 PSV-#L2REC7H-#L2REC31H-#L2FLR
11		#CET1 HI PRESSURE, #L2 ENTRY	#L2 PSV-#L2REC7H-#L2REC31H-#L2FLR-#CDES-LBL-SP1D
12			#L2 LHSI
13			#L2 LHSI-#L2FLR
14		#CET1 HI PRESSURE, #L2 ENTRY	#L2 LHSI-#L2FLR-#CDES-LBL-SP1D
15			#L2 LHSI-#L2REC7H
16			#L2 LHSI-#L2REC7H-#L2FLR
17		#CET1 HI PRESSURE, #L2 ENTRY	#L2 LHSI-#L2REC7H-#L2FLR-#CDES-LBL-SP1D
18			#L2 LHSI-#L2REC7H-#L2REC31H
19			#L2 LHSI-#L2REC7H-#L2REC31H-#L2FLR
20		#CET1 HI PRESSURE, #L2 ENTRY	#L2 LHSI-#L2REC7H-#L2REC31H-#L2FLR-#CDES-LBL-SP1D

**Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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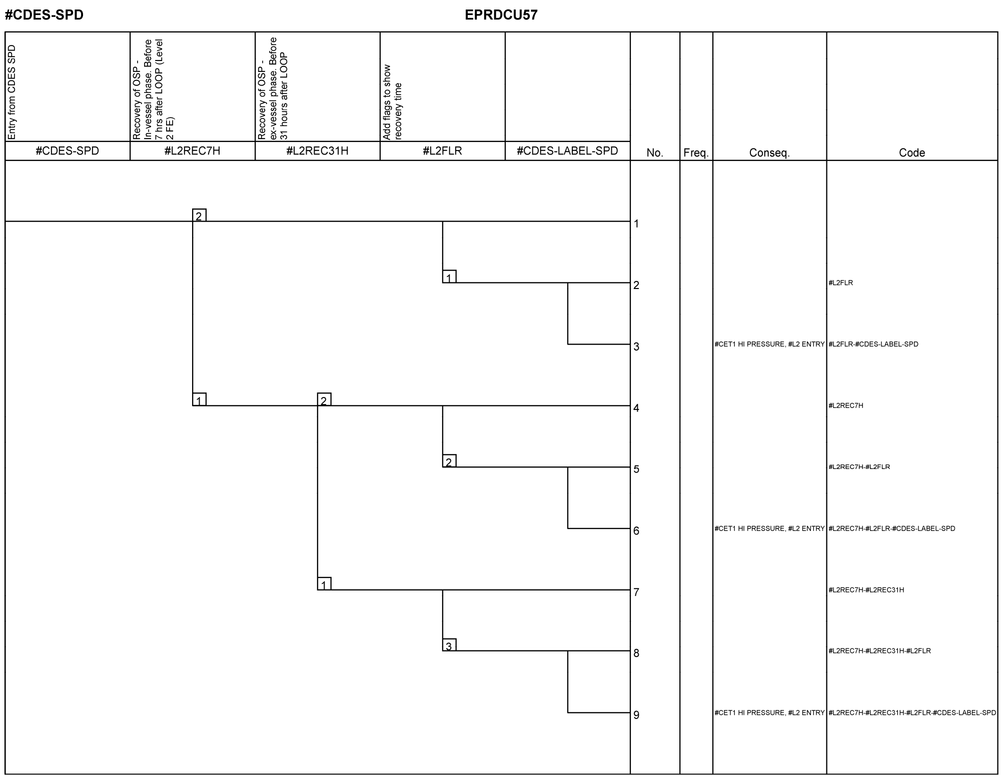

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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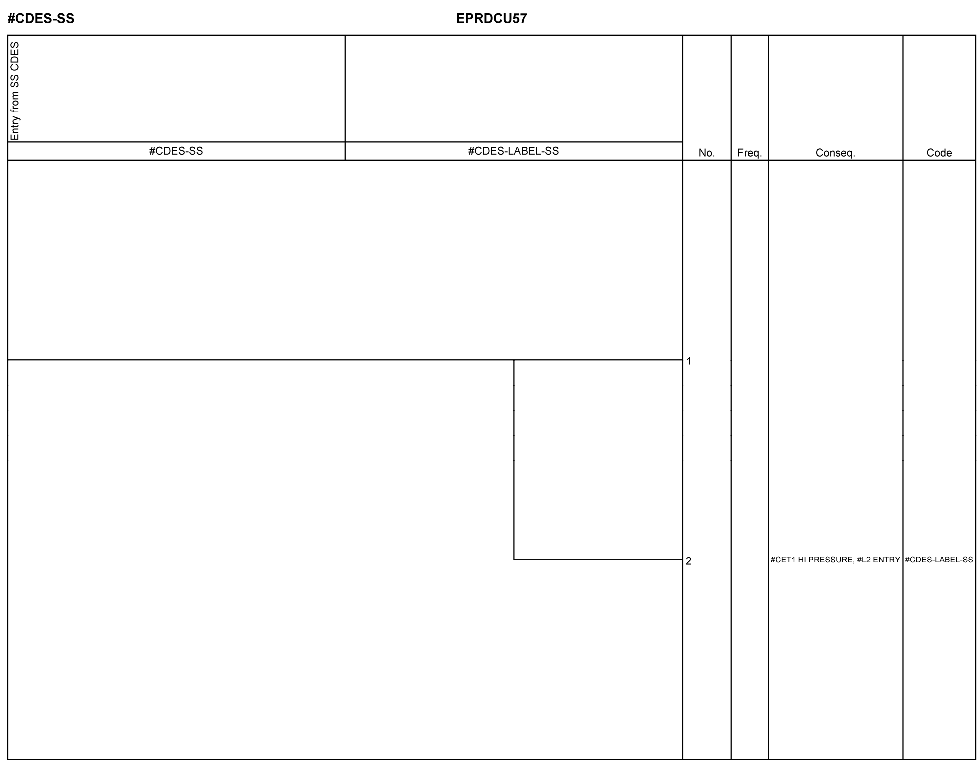

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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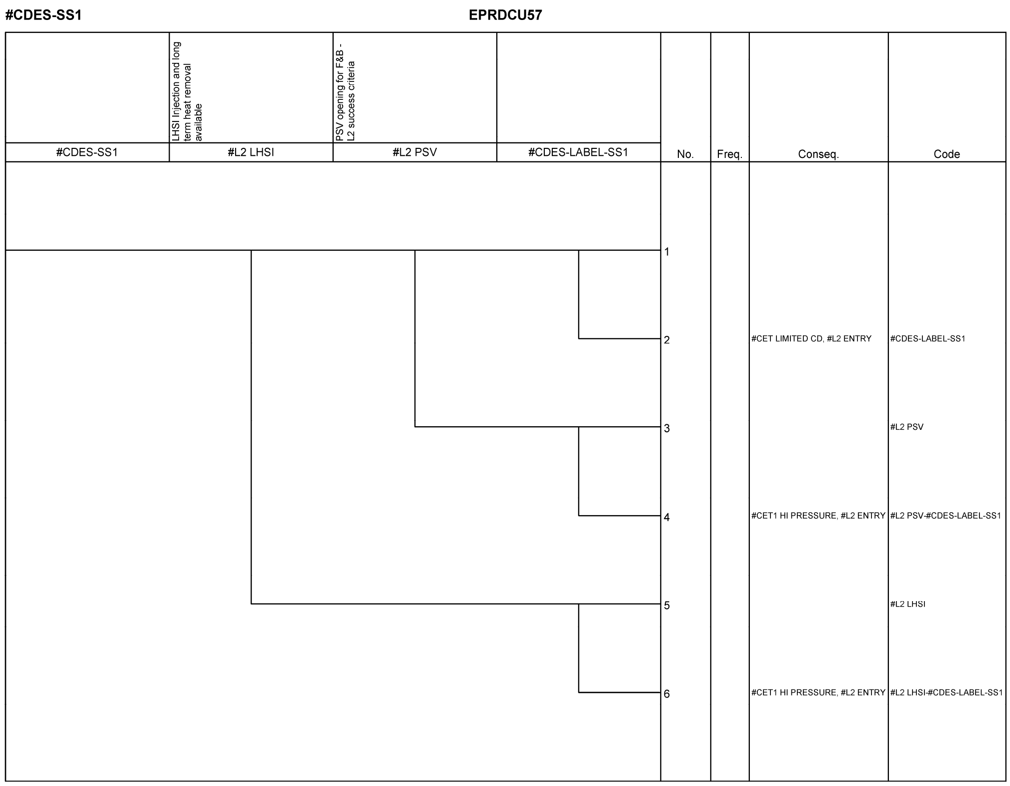

**Figure 19-240-1—U.S. EPR PRA CDES Link Trees
Sheet 22 of 29**

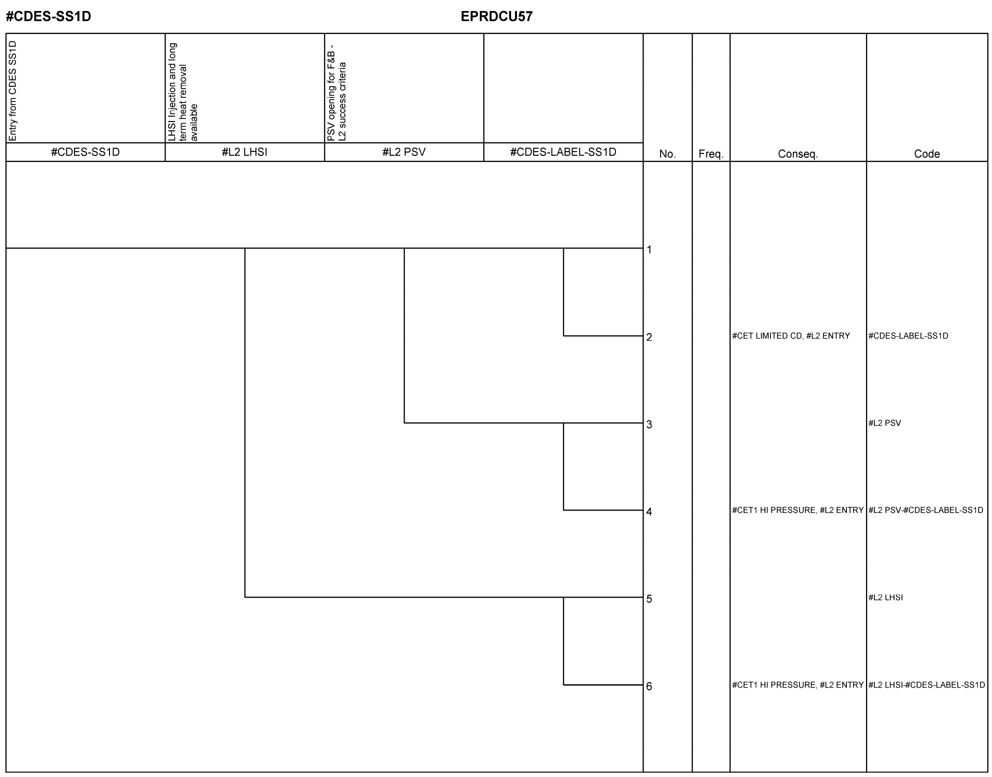

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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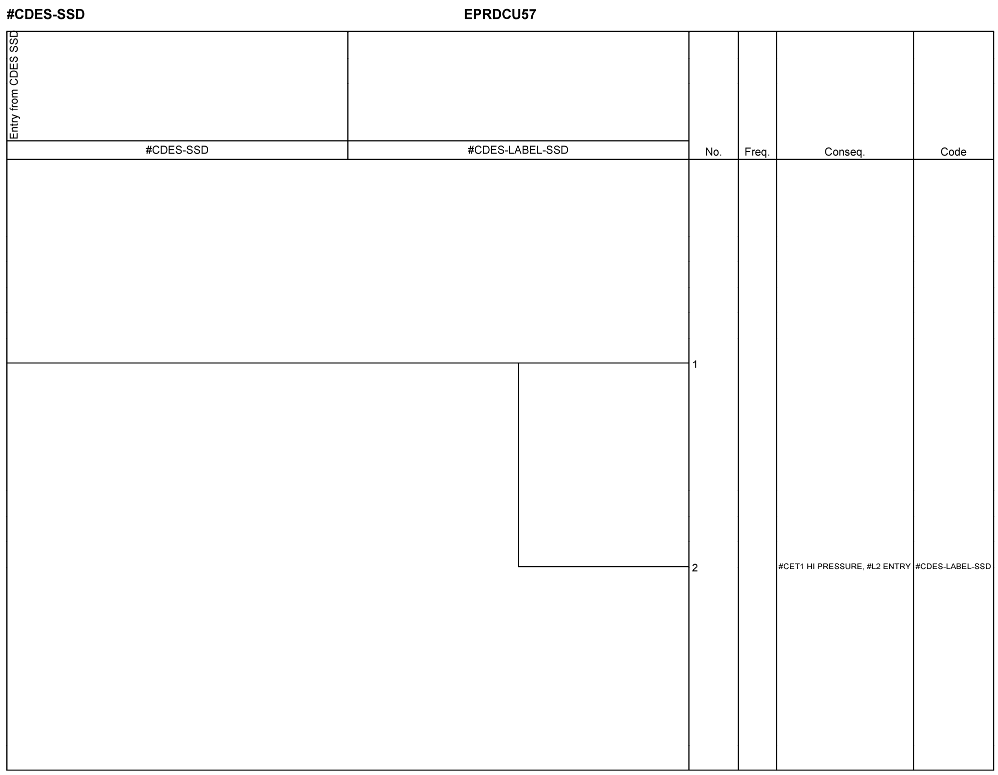

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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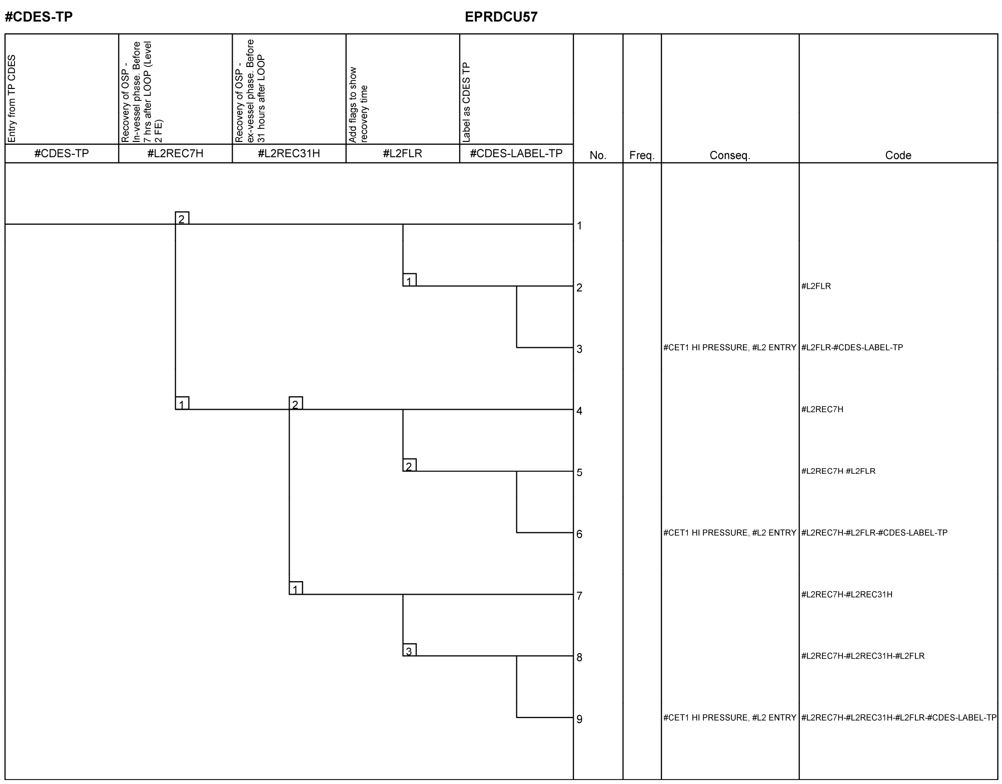

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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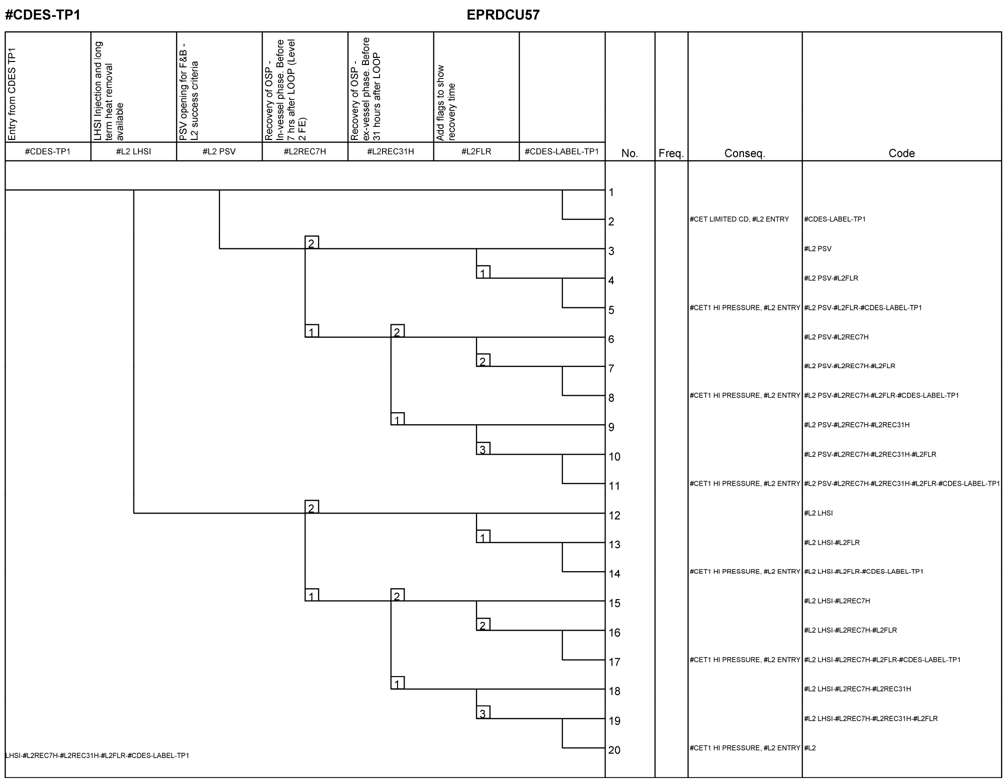

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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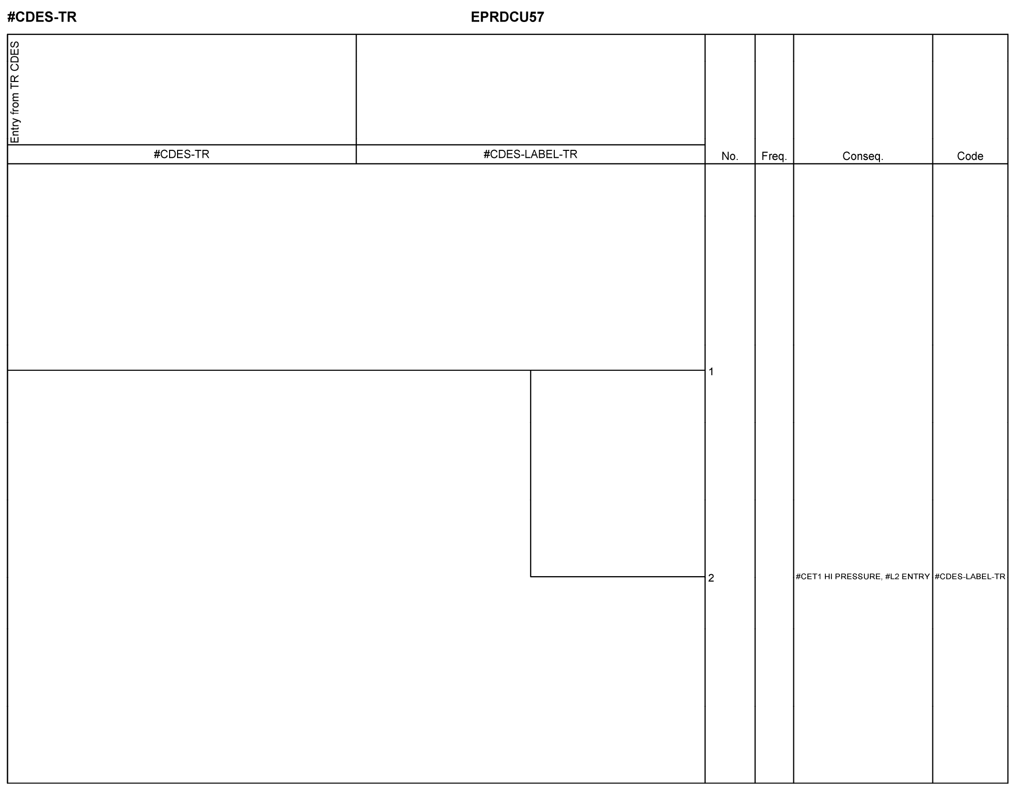

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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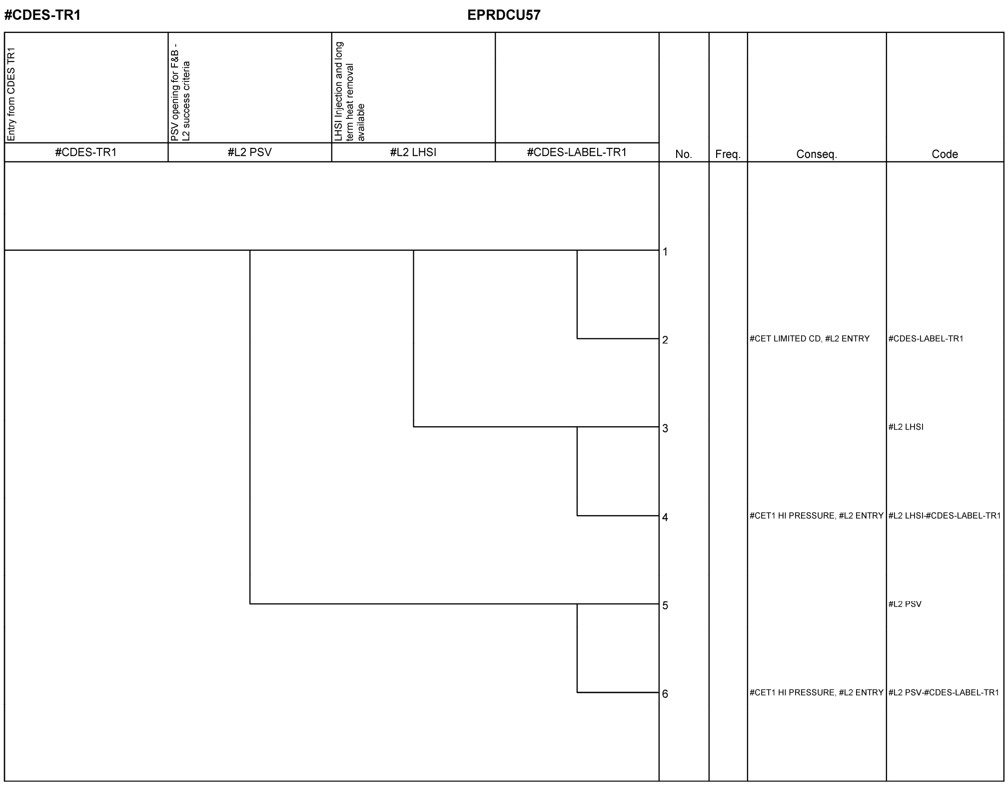

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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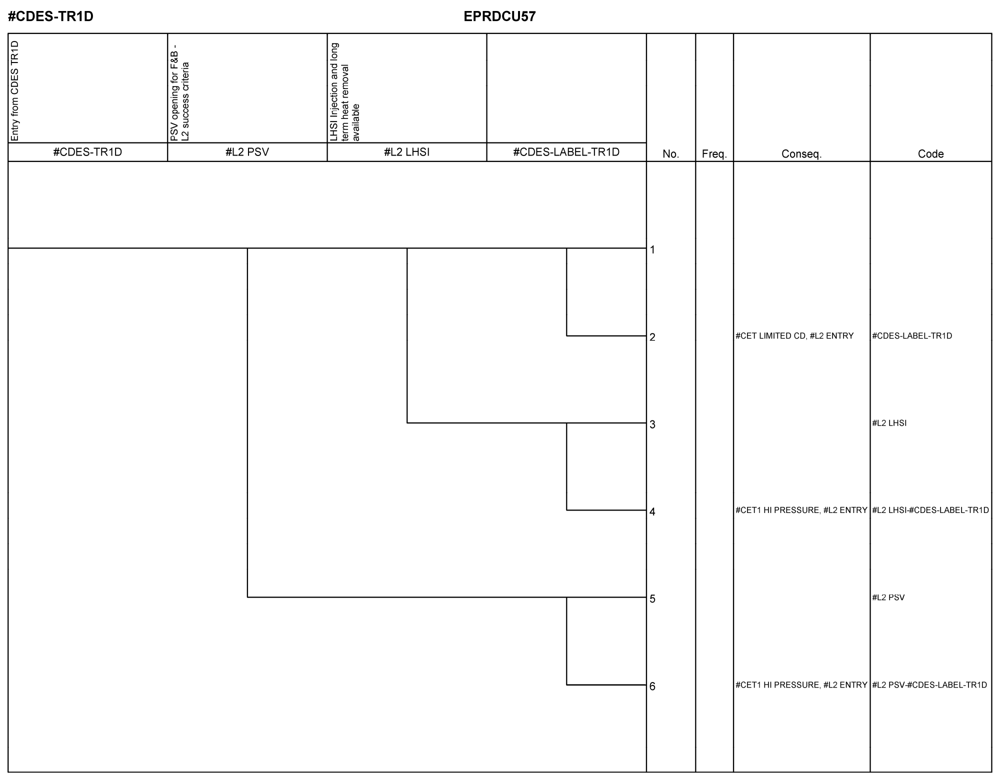

Figure 19-240-1—U.S. EPR PRA CDES Link Trees
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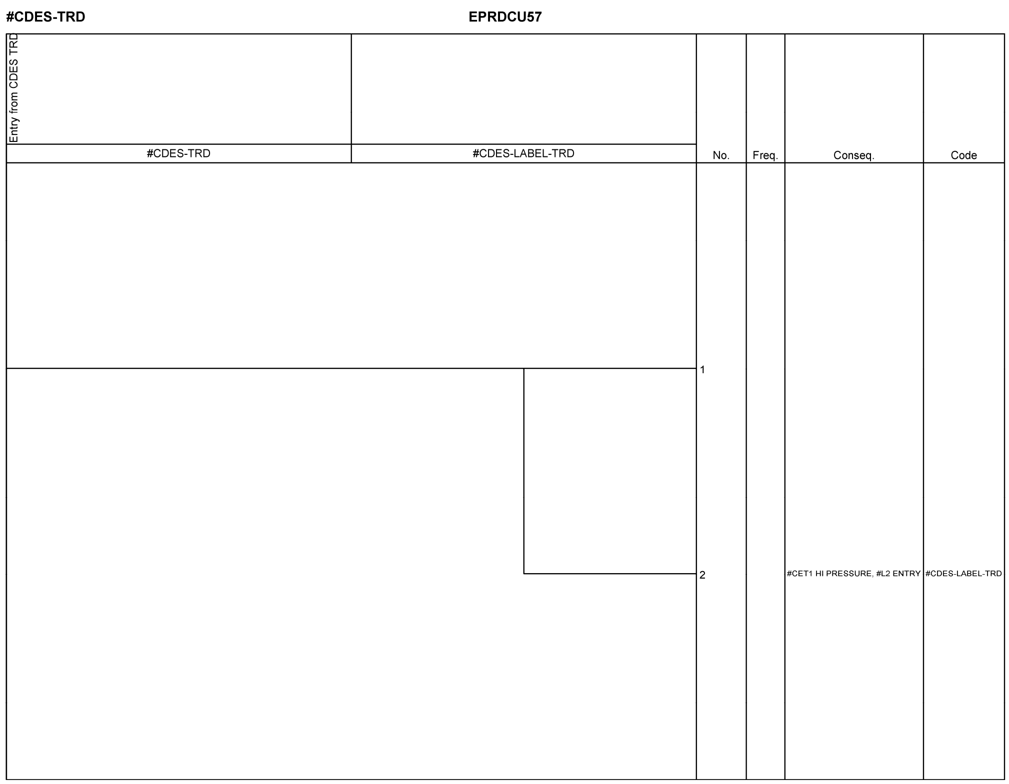


FSAR Impact:

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

Question 19-241:

Please provide the inner diameter of the control rod guide assembly (CRGA) in the location where it penetrates the RPV upper head.

Response to Question 19-241:

The CRGA does not penetrate the reactor pressure vessel (RPV) upper head (See Figure 19-241-1). The control rod drive mechanism (CRDM) adaptor nozzle penetrates the RPV upper head. A thermal sleeve is installed inside the CRDM adaptor nozzle and the CRDM drive rod is inserted through the thermal sleeve into the CRGA.

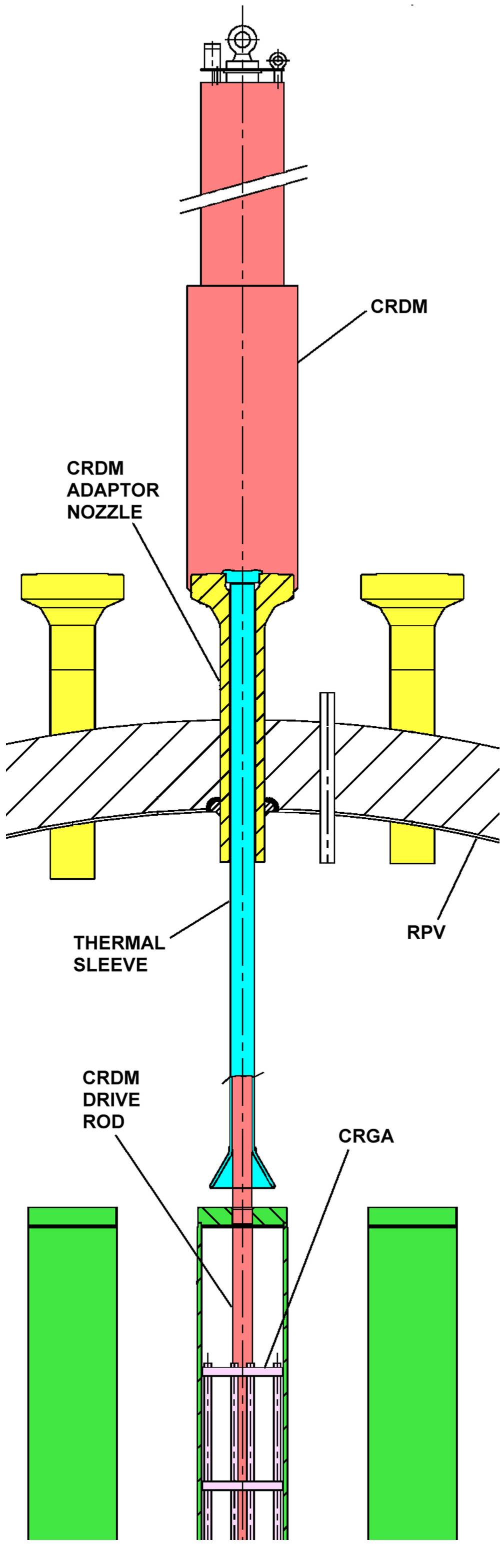
The following components penetrate the RPV upper head.

- CRDM adaptor nozzle: inside diameter = 2.95 inches (75 mm).
- Thermal sleeve: Inside Diameter = 2.44 inches (62 mm) at the top tapering out to 2.60 inches (66 mm) at the bottom funnel.

FSAR Impact:

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

Figure 19-241-1—CRGA Details



Question 19-242:

(Follow-up to Question 19-150) With respect to the distributed heat sink data provided in Table 19-150-1 of the response to RAI 19-150:

a. Please provide the thickness of each heat sink.

It is stated in the response that the heat sink data in Table 19-150-1 are used as the source for the MAAP input deck. Pease clarify the reasons for differences in the total surface areas of various heat structure material as calculated using the data in Table 19-150-1 and those of the MAAP input deck A.

Response to Question 19-242:

A response to this question will be provided by January 30, 2009.

Question 19-243:

(Follow-up to Question 19-192) It is stated in the response to Question 19-192 that AREVA does not understand the regulatory basis of the question. This basis is described in NEI 91-04 (originally NUMARC 91-04), "Severe Accident Issue Closure Guidelines," which summarizes the important information pertaining to the agreement between the nuclear industry and the NRC on severe accident management, and contains the binding implementing guidance in Section 5. As such, it is an industry initiative that was endorsed by the NRC.

The current industry technical basis stems from EPRI's Severe Accident Management Technical Basis Report (EPRI TR-101869), which was used in developing vendor-specific guidance. This guidance was provided to the NRC by the various owners groups and constitutes the technical basis of severe accident management for existing plants. What the NRC staff is now looking for is the revised technical bases for the new plants, which would consider the new features for accident prevention and mitigation. The staff needs to review this material before any COL is issued to ensure that no unexpected problems arise that would be difficult to resolve after the fact.

After a COL is issued, it is imperative that the new plants also have their SAMG implementation in place, including procedures and training, prior to initial fuel load.

The purposes of Question 19-192 were:

1. To obtain recommendations from AREVA, for the various accident scenarios being considered, regarding how best to prevent the accidents from progressing to core damage, terminate core damage once it begins, maintain the capability of the containment as long as possible, and minimize on-site and off-site releases and their effects.
2. To request AREVA to identify a COL action item that would require each COL applicant to provide documentation of the severe accident technical basis.

The NRC staff is not asking for specific severe accident management guidelines. Instead, the staff needs to evaluate the technical bases that would support the guidelines. Specifically, the staff needs to review the results obtained during the first two phases identified in the last paragraph of the initial response to question 19-192. In that regard, please provide documentation of the results of these first two phases, culminating in the overall mitigation strategies that would be developed based on symptoms and phenomena. Also provide any available supplemental analyses that demonstrate the effectiveness of the operator actions to the plant response. Please provide, to the extent practicable, a discussion of the accident sequences considered and the results of the supporting MAAP 4.0.7 analyses. Finally, please provide the necessary COL action item or a justification for not providing one.

Response to Question 19-243:

A response to this question will be provided by June 19, 2009.

Question 19-244:

(Follow-up to Question 19-148) The response to Question 19-148 contained results for a single instrument tube failure. The NRC Staff is also interested in what would be the consequences of failing multiple instrument tubes, from oxidation of the Zircaloy cladding.

1. Please provide an analysis of the consequences of failing all of the AMS probes in the region of the core where the Zircaloy oxidation takes place, for each of the relevant scenarios. Discuss how the hydrogen, steam, and fission products would then flow through the AMS probes and out into the instrumentation rooms. Please identify the recipient containment node in the MAAP model, and discuss the potential for a large hydrogen deflagration or detonation there and in adjacent compartments. In addition, please describe the gas flows from there into the rest of the containment building. Please present similar plots (mole fractions of hydrogen, steam, oxygen, and nitrogen; containment pressures, and temperatures; and fission product releases into the containment) to those presented in the initial response to the question.
2. For both the single probe and multiple probe failure cases, provide plots of the core-to-upper plenum natural circulation, countercurrent natural circulation flow rates in the hot leg and steam generator tubes, and damage fractions in the hot leg and steam generator tubes.

Response to Question 19-244:

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

Question 19-245:

The most recent revision to the severe accident source term presents the results of the MAAP cases that were re-calculated in order to address core inventory input error that was discovered by AREVA (WebCAP 2008-3905-CR and 2008-4034-CR).

Please address the following questions:

1) Do these revised source term results require any modification to previously provided responses to RAIs by AREVA? If this is the case, please provide a list of the affected RAI responses, and resubmit your responses.
2) Are there any other significant analyses that have been revised as a result of the revised source term data? If so, please provide a list of the affected analyses.
3) Does this revision affect the information that has been transmitted by AREVA to the current COL applicants? Please elaborate.

Response to Question 19-245:

1) The revised source term results require modification to previously submitted RAIs. The RAIs affected include RAI 6, Supplement 1, Question 19-84 and Question 19-124. Revised versions of these questions will be submitted separately.

Changes to RAI 6, Supplement 1 questions are summarized as follows:

- 19-84a - The response was updated to clarify that release category RC206 is not considered a large release fraction (LRF) case. In light of this, the figures related to Modular Accident Analysis Program (MAAP) run st1.8f have been removed. The MAAP run results reflect the updated core inventory and the additional ventilation path out of the Fuel Building and Safeguards Buildings described in the response to RAI 19-117a.

- 19-84b - The statements made in the original response to this question do not change as a result of the updated Level 2 probabilistic risk assessment (PRA) core inventory. No update to this response is necessary.

- 19-84c - Table 19-84c-1, 2, and 3 have been updated to reflect the information from the MAAP runs performed with the updated core inventory.

- 19-84d - The text and figures of this response have been updated to reflect the impact of the updated core inventory on the MAAP parametric sensitivity calculations.

- 19-84e, g, h - The statements made in the original responses to this question do not change as a result of the updated Level 2 PRA core inventory. No update to these responses is necessary.

- 19-84f - The text and figures in this response were updated to impact of the updated core inventory on the source term as well as the additional ventilation path out of the Fuel Building and Safeguards Buildings described in the response to RAI 6 Supplement 1, Question 19-117a.

- 19-124 - The MAAP run results reflect the updated core inventory. MAAP run st1.8f was not included since this case is not considered a LRF case.

2) The Level 3 PRA associated with U.S. EPR Environmental Report ANP-10290 will be revised as a result of the revised source term (see Questions 19-236 and 19-237).

3) AREVA NP believes this question is not applicable for the design certification review.

FSAR Impact:

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

Question 19-246:

Please provide detailed results of the MAAP 4.0.7 and S-RELAP5 calculations to determine Level 1 success criteria. Also include the results of benchmarking MAAP 4.0.7 against S-RELAP5 and explain how the benchmarking was performed. As part of your discussion:

1. Please provide results of all of the analytical comparisons made, and include any cases that provide a direct core heatup comparison.
2. Please explain why phenomena associated with these events are expected to extend to the remaining events of interest, such as steam generator tube rupture, steam line breaks (inside or outside containment), and loss of off site power.
3. Please list and discuss the key assumptions made either in model development activities or in processing the individual calculations.
4. Please explain why an alternative acceptance criterion of a MAAP 4.0.7 calculation of 1800 F for peak cladding temperature should be used, and why calculations yielding peak cladding temperatures between 1400 F and 1800 F should be evaluated qualitatively or by an independent calculation before being accepted.
5. Please list and discuss the major conclusions from the comparison.
6. Please present the success criteria results obtained with MAAP 4.0.7 for all of the events considered.

Response to Question 19-246:

1. From the 155 MAAP4 calculations performed as part of the probabilistic risk assessment (PRA) Level 1 mission success criteria study, seven scenarios were identified for further study with S-RELAP5. These particular scenarios were selected based on the presence of a broad spectrum loss of coolant accident (LOCA) and of non-LOCA phenomena impacting both primary and secondary side conditions. Among the seven scenarios, four were loss of main feedwater (LOMFW) and three were LOCAs. By design, six cases are “success” cases; that is, core damage was not predicted in the original MAAP4 simulations. One case (Case 4i) was a “failure” in the original MAAP4 simulations. Also, one LOMFW and one LOCA case among those simulations did not exhibit core heat up.

   Table 19-246-1 is a summary of the selected plant scenarios examined in detail. Event permutations include the status of reactor coolant pumps (RCP), main steam relief valves (MSRV), main steam safety valves (MSSV), emergency feedwater (EFW), medium head safety injection (MHSI), low head safety injection (LHSI), accumulators, the actuation of partial cooldown or fast cooldown using the main steam relief train, and the actuation of the primary depressurization system (PDS) valves.

   The results of the MAAP4 and S-RELAP5 runs for five cases are discussed in the following sections. The results from these cases provided insight into the applicability of MAAP4 to simulate plant thermal-hydraulic behavior prior to core damage. The results from the remaining two cases reinforce the conclusions drawn from the other five cases. The implication of the highlighted code-to-code differences presented in the presentation of results is addressed in the responses to parts 4 and 5 of this RAI.

Table 19-246-1—Summary of MAAP4 Cases Examined with S-RELAP5

Case / Trait	3c	4g	12a	13d	16a1	4i	4cc
Initiating Event	LOFW	LOFW	SBLOCA (2in)	SBLOCA (3 in)	MBLOCA (6 in)	LOFW	LOFW
RCPS	Trip	Run	Trip	Trip	Trip	Run	Run
MSRV	All fail closed	All fail closed	1/4	1/4	Off	All Fail Closed	All Fail Closed
PCD/FCD	NA	NA	PCD	FCD	NA	N/A	N/A
MSSV	1/8	1/SG	Off	Off	Off	1/SG	1/SG
EFW	1/4	Off	1/4	1/4	Off	Off	Off
MHSI	Off	1/4	1/4	Off	1/4	1 of 4	1 of 4
ACC	Off	1/4	Off	1/4	Off	1 of 4	1 of 4
LHSI	Off	Off	Off	1/4	1/4	Off	Off
Operator	NA	Open PDS valves at 1.1 hr	NA	Open 1 MSRV at 40 min (FCD)	NA	N/A	Open PDS valves at 90 min

LOMFW Case 3c

The eventual consequence of the initiating event, a loss of main feedwater to all four steam generators, is the complete loss of primary-to-secondary heat transfer. In this scenario primary-to-secondary heat transfer is further degraded by the failure to open the MSRVs and MSSVs associated with three steam generators. Normally, a reactor trip would occur on low steam generator level narrow range; however, in the MAAP4 and S-RELAP5 calculations, reactor trip is assumed coincident with the initiating event along with RCP trip and main steam isolation valve (MSIV) closure. The success of this event is attributed to the one available EFW which adequately removes core decay heat.

Figure 19-246-1 presents the maximum core temperature from MAAP4 and compares that to a similarly modeled S-RELAP5 core region. Because the S-RELAP5 core region model is finer than MAAP4, the S-RELAP5 result is not a maximum value for the calculation; instead, as shown in Figure 19-246-2, the maximum core temperature is given from a modeled high power rod within a high power or “hot” assembly. Figure 19-246-3, average core water temperature, illustrates a difference between MAAP4 and S-RELAP5: primary-to-secondary heat transfer. The lower temperature in the MAAP4 calculation suggests that more heat is leaving the primary than predicted in the S-RELAP5 calculation. This also helps explain the difference between fuel temperatures in Figure 19-246-2. Specifically, heat transfer from the fuel to the coolant is obviously lower in the MAAP4 calculation, and is attributed to different heat transfer regimes predicted by the two codes. It does not have significant influence on the event simulation.

Figure 19-246-4 shows the steam generator liquid level. This provides the best evidence of the primary-to-secondary heat transfer difference. Both simulations show the impact of high and low primary-to-secondary heat transfer. MAAP4 shows an initial high primary-to-secondary heat transfer period, as evidenced by the steep drop in level, followed by a low primary-to-secondary heat transfer period as the steam generator empties. S-RELAP5 shows an initial low primary-to-secondary heat transfer period prior to a period of high primary-to-secondary heat transfer then, like MAAP4, a period of low primary-to-secondary heat transfer follows.

A follow-on sensitivity study was performed to investigate the inconsistency between MAAP4 and S-RELAP5 with regard to primary-to-secondary heat transfer. To compensate for this apparent inconsistency, S-RELAP5 was rerun delaying the Loop 3 RCP trip for 20 minutes (1200 s). In doing this, convective heat transfer to the secondary is enhanced. Convective heat transfer is also enhanced in the core region.

Agreement is improved between the codes. With the one RCP running for 20 minutes in the revised calculations, S-RELAP5 results do not show an equivalent primary-to-secondary heat transfer rate. In fact, the S-RELAP5 simulation exhibits more primary-to-secondary heat transfer than MAAP4 (e.g., lower early temperatures). Comparison of the liquid level in the steam generator is in better agreement as shown in Figure 19-246-5.

The performance of the MAAP4 code against the S-RELAP5 code in this study, as measured by both relative magnitudes and trends of key results (e.g., cladding temperature), is good. A major difference between the codes is the primary-to-secondary heat transfer as identified in the initial benchmark. This is attributed to the timing of fully developed natural circulation, as demonstrated by the sensitivity study.

Figure 19-246-1—MAAP4 vs. S-RELAP5 Core Temperature (Best Estimate)

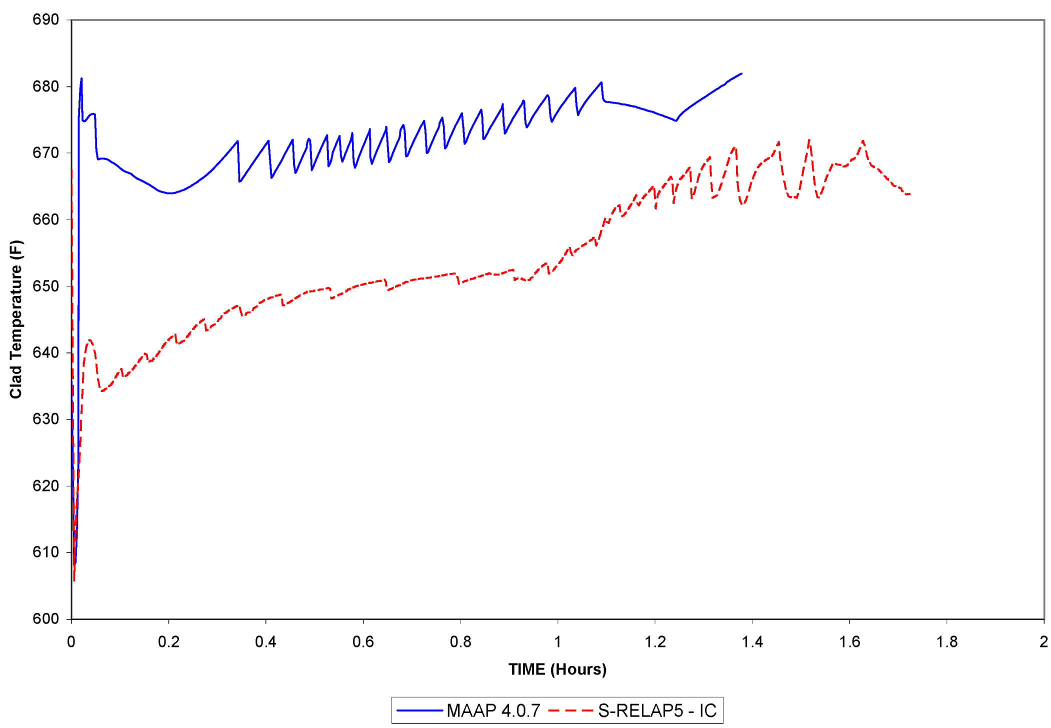


Figure 19-246-2—S-RELAP5 Cladding Temperatures (Best Estimate)

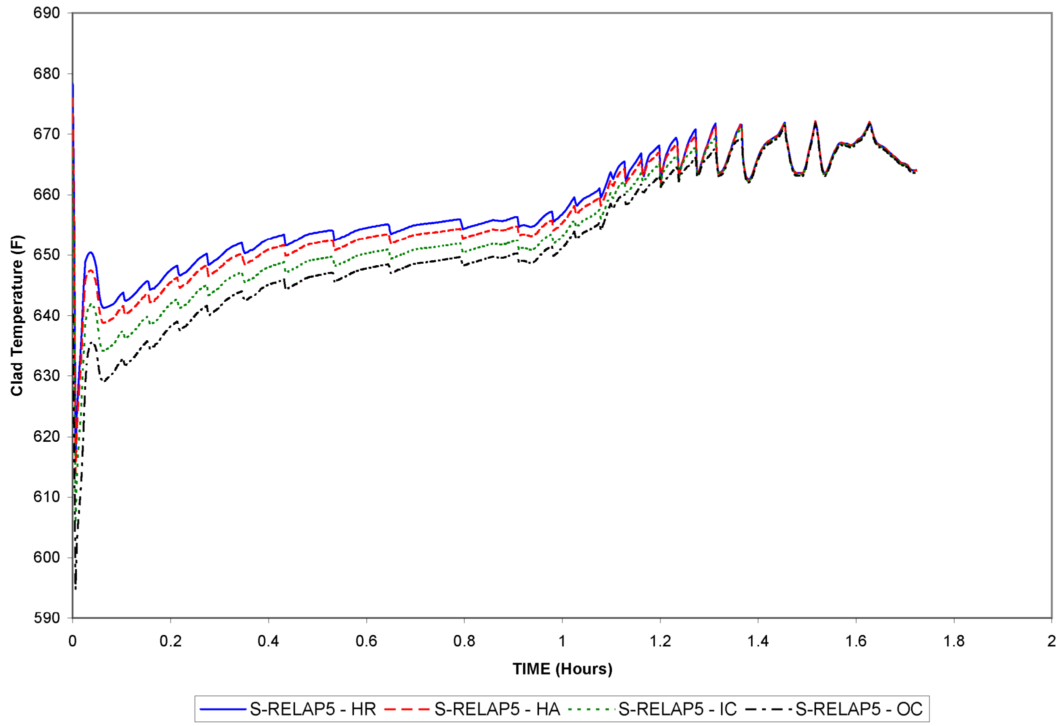


Figure 19-246-3—Average Core Water Temperature (Best Estimate)

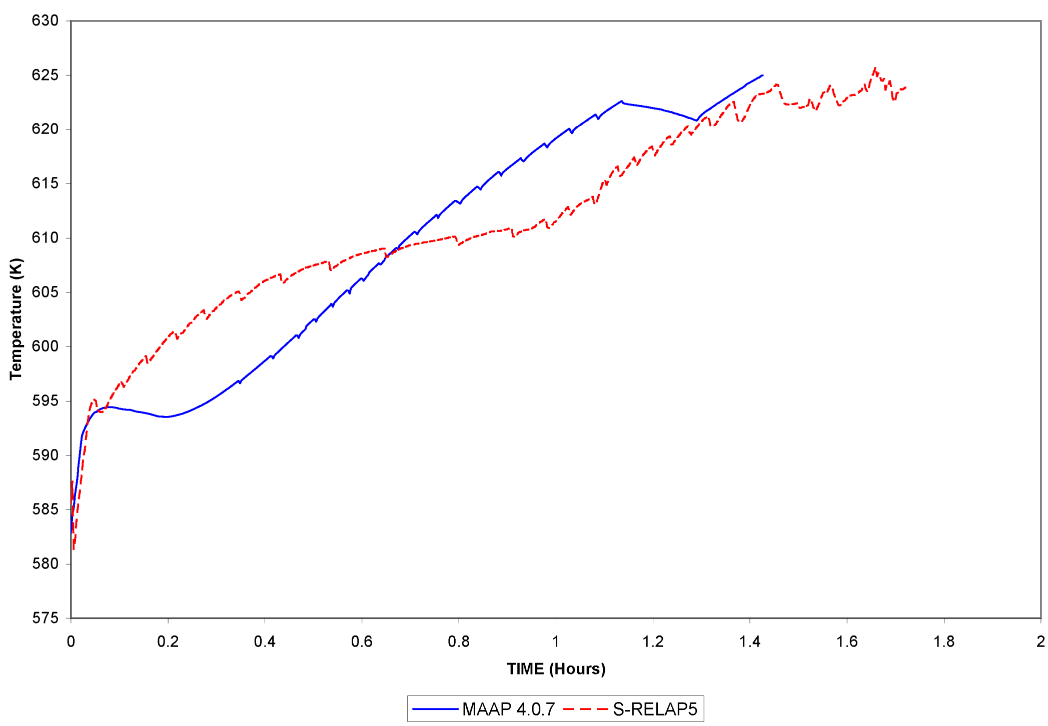


Figure 19-246-4—Steam Generator Water Level (Best Estimate)

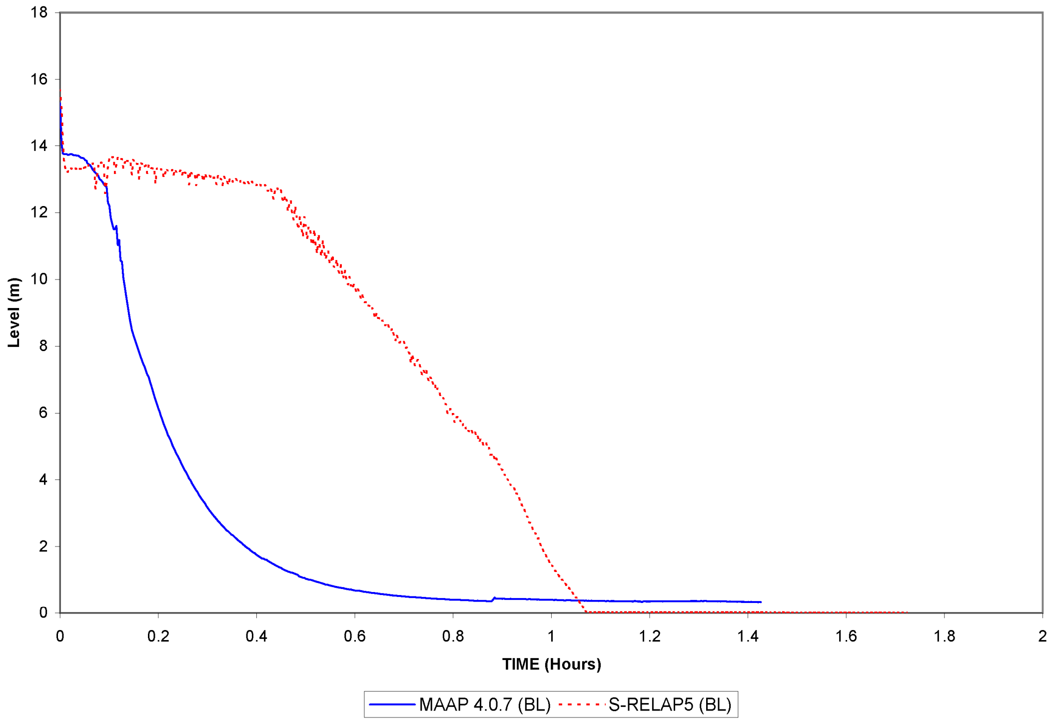
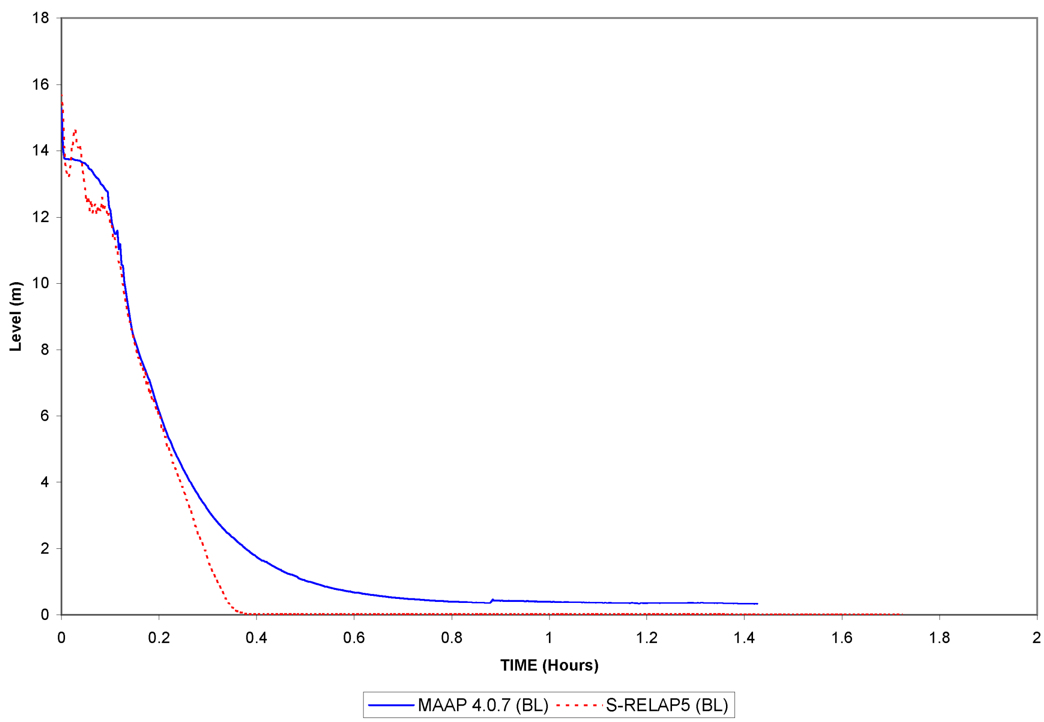


Figure 19-246-5—Steam Generator Water Level (Tuned)



LOMFW Case 4g

The eventual consequence of the initiating event, a loss of main feedwater to all four steam generators, is the complete loss of primary-to-secondary heat transfer. In this scenario the plant model assumes normal reactor protection systems and a reactor trip occurs on low steam generator level (narrow range) shortly after the event begins. In both the MAAP4 and S-RELAP5 calculations, MSIV closure is assumed coincident with the initiating event; however, the RCPs are allowed to continue to operate for the duration of the simulation. With the coincident LOMFW and loss of EFW, the success of this scenario is attributed to the available one accumulator and one medium head safety injection.

Figure 19-246-6 presents the maximum core temperature from MAAP4 and compares that to a similarly modeled S-RELAP5 core region. As before, the S-RELAP5 result is not a maximum value for the calculation; rather, as shown in Figure 19-246-7, the maximum core temperature is given from a modeled high power rod within a high power or “hot” assembly. Figure 19-246-8 shows the average core region water temperature. With RCP operation and the initial steam generator liquid level as a buffer, fluid and fuel temperatures are maintained close to the steady state value until the secondary heat sink is lost.

A difference is observed between the S-RELAP5 and MAAP4 results with regard to fuel temperature shortly after the steam generators are dry. The difference is attributed to interphase drag, which is explicitly modeled in the S-RELAP5 but not in MAAP4, and contributes to the homogenization of the system two-phase conditions, providing improved core heat transfer in the S-RELAP5 simulation. MAAP4 does track a “two-phase natural circulation” condition in the RCS loops, which improves core heat transfer; however, when void fraction exceeds a predefined threshold (a MAAP4 user option), this improved heat transfer condition ends. This assumption, while briefly providing accurate core heat transfer, is a compensating error. The two-phase condition in the core is not driven by natural circulation; rather, it is the consequence of the RCS blowdown condition resulting from the open pressurizer power operated relief valve (PORV). Since the conditions for “two-phase natural circulation” end in the MAAP4 simulation, phase separation occurs as evidenced by the sharp drop in the reactor vessel liquid level. This separation has the consequence of leaving the core region completely dry. The more homogenized condition predicted in the S-RELAP5 calculation results in improved heat transfer, relative to MAAP4, when the reactor vessel loses coolant inventory.

A follow-on sensitivity study was performed to assess the impact of a RCP trip at 1 hour. In general, the pump trip at 1 hour has little impact on the fuel temperatures predicted by S-RELAP5. Without the pumps, the coolant phases are expected to be more separated than in the base calculation. However, the revised S-RELAP5 calculations did not show a strong sensitivity to this change.

The performance as measured by trends of key results of the MAAP4 code closely follows the S-RELAP5 code. For Case 4g. The magnitude of a key simulation measure, maximum fuel temperature, is conservative. The differences between MAAP4 and S-RELAP5 are explained by the role of interfacial drag in this scenario and the limitations of the MAAP4 code to simulate the core heat transfer condition during a RCS blowdown from open PDS valves. MAAP4 provides a conservative prediction for this type of scenario.

The sensitivity study supports this conclusion. With the RCP trip in the S-RELAP5 model, more phase separation is expected. While the RCP trip is observed as a penalty, it did not result in a significant degradation of core heat transfer as observed in the MAAP4 simulation. It is the blowdown condition that drives the core heat transfer and not the RCPs.

Figure 19-246-6—MAAP4 vs. S-RELAP5 Core Temperature (Best Estimate)

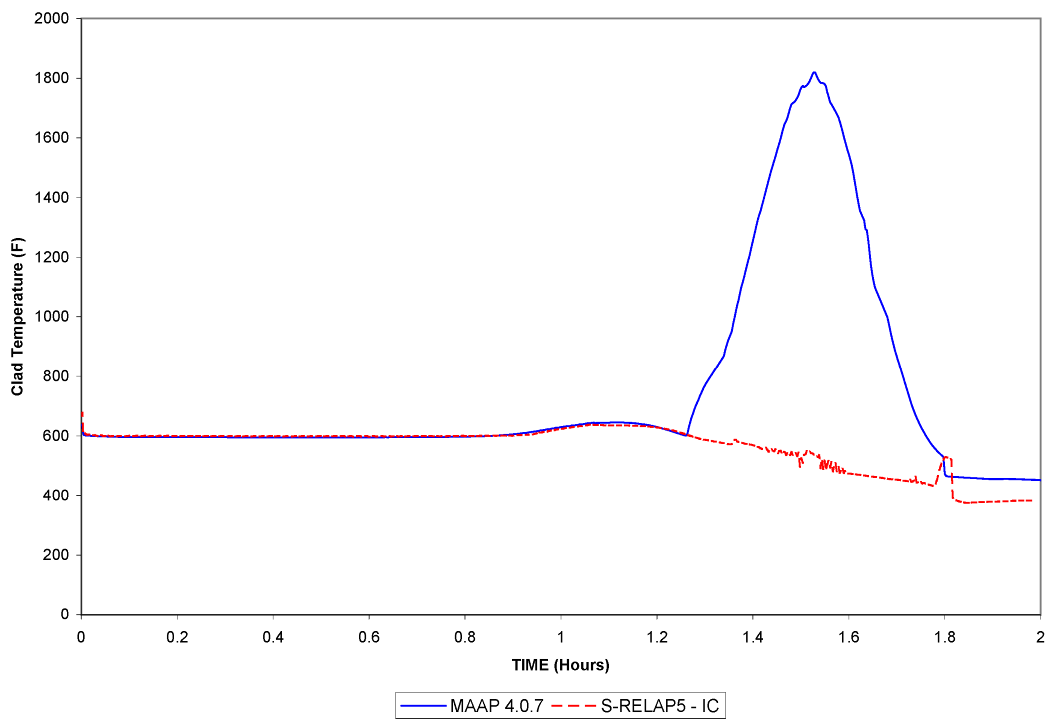


Figure 19-246-7—S-RELAP5 Cladding Temperatures (Best Estimate)

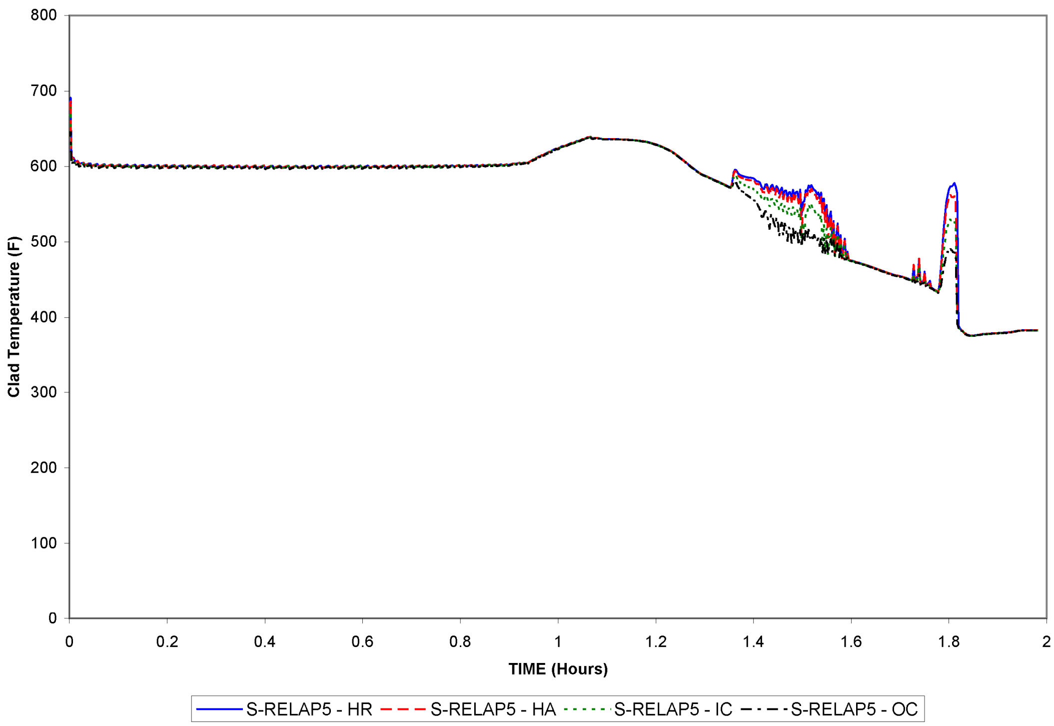
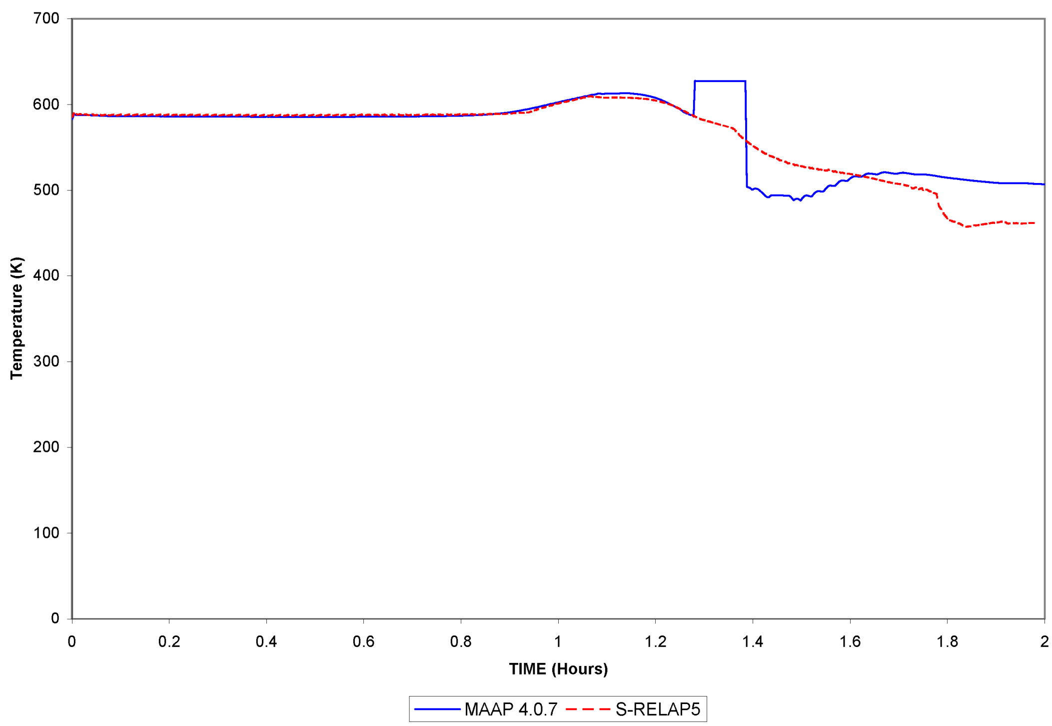


Figure 19-246-8—Average Core Water Temperature (Best Estimate)



LOMFW Case 4i

The only difference between Case 4g and 4i is that the PDS valves are not actuated in Case 4i. As a consequence, primary system pressure remains high for the duration of the simulation. As with Case 4g, the eventual consequence of the initiating event is the complete loss of primary-to-secondary heat transfer. This is shown in Figure 19-246-9, the wide range (WR) Level in steam generator. The level decreases steadily due to the boil-off of the steam generators. Once dry, the steam generators remain dry because there is no emergency feedwater supply. From Figure 19-246-9, primary-to-secondary heat transfer is similar between the MAAP4 and S-RELAP5 simulations.

When the steam generators dry out, secondary heat removal is completely lost and the reactor coolant system pressure begins to increase. Subsequently, the pressurizer safety valve cycles open/closed maintaining the reactor coolant system pressure between the opening/closing setpoints. The mass flow rate through pressurizer safety valve is shown in Figure 19-246-10.

The loss of primary system inventory resulting from the opening of the pressurizer safety valve leads to core uncovery and heatup. This is reflected in the cladding temperature given in Figure 19-246-11. As shown, MAAP4 conservatively estimates the cladding temperature. This is a consequence of early phase separation.

Figure 19-246-9—Steam Generator Water Level

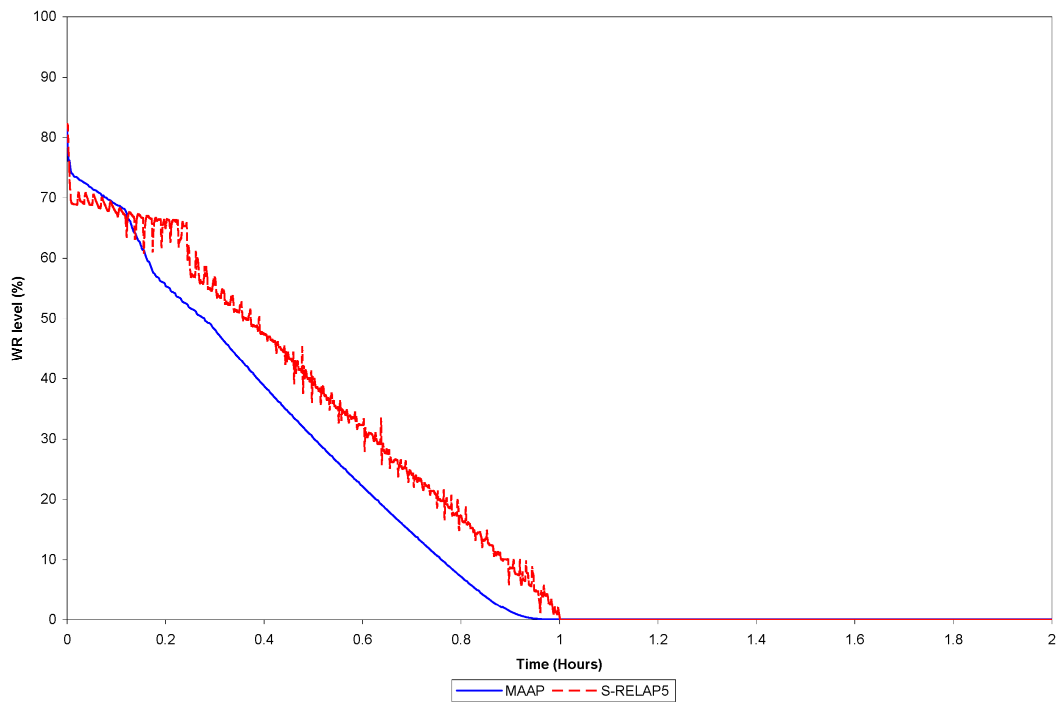


Figure 19-246-10—Pressurizer Safety Valve Mass Flow Rate

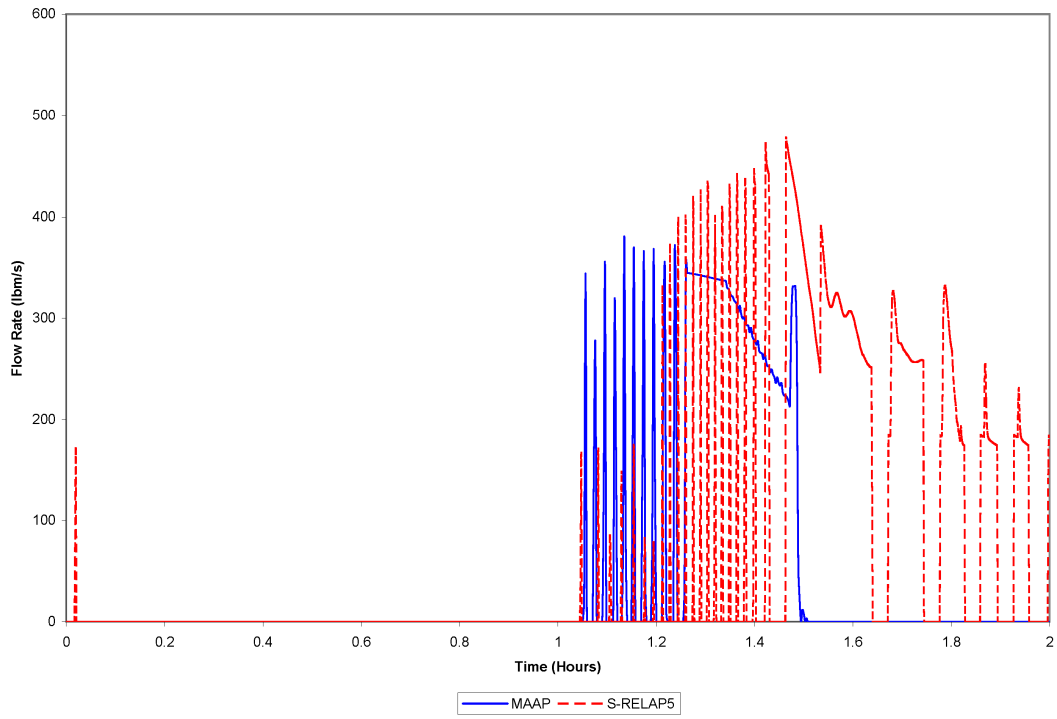
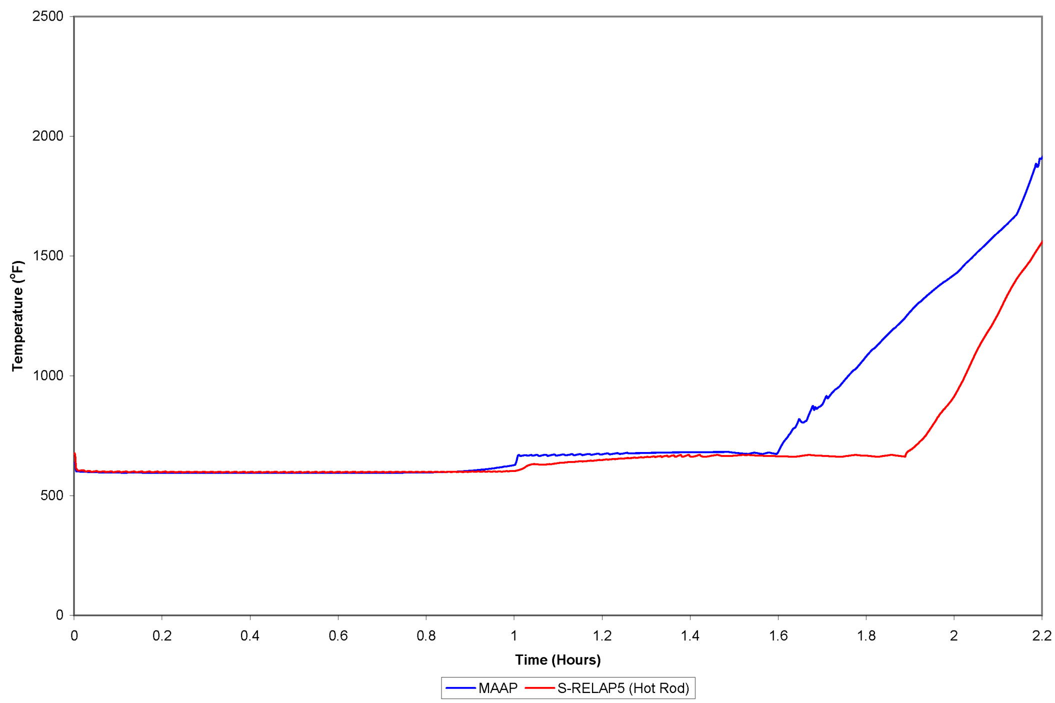


Figure 19-246-11—MAAP4 vs. S-RELAP5 Cladding Temperature



3” LOCA Case 13d

The immediate consequence of the initiating event, a 3 inch cold leg break loss of coolant accident, is a rapid decrease in primary system pressure. Normally, a reactor trip would occur on low system pressure; however, in the MAAP4 and S-RELAP5 calculations, reactor trip is assumed coincident with the initiating event along with RCP trip and MSIV closure.

Figure 19-246-12 shows the primary system pressure. Both the trends and magnitudes of pressure are similar. Figure 19-246-13 shows break flow and, like pressure, shows good agreement in magnitude and trend. The eventual consequence of the LOCA is to reduce the RCS inventory and similarly the reactor vessel inventory.

Core uncovery occurs in both MAAP4 and S-RELAP5 simulations, which results in rising fuel temperatures as shown in Figure 19-246-14. The core temperature is not reduced until operators initiate fast cooldown by opening an MSRV. Steam generator liquid level, shown in Figure 19-246-15, reflects the consequence of the fast cooldown in terms of coincident steam generator liquid level drop.

The main differences between the MAAP4 and S-RELAP5 in this case are the magnitude of the core temperature excursion and the steam generator liquid level. In Figure 19-246-15, the steam generator liquid level, the code result comparison indicates a significant difference in primary-to-secondary heat transfer. In the MAAP4 calculation, an initial high primary-to-secondary heat transfer period is evidenced by an early steep drop in level. The S-RELAP5 result shows an initial low primary-to-secondary heat transfer period prior to the fast cooldown. Even during the fast cooldown, the liquid level predicted by S-RELAP5 does not decline at the same rate as in the MAAP4 simulation. Since S-RELAP5 predicts low primary-to secondary heat transfer, water from the EFW contributes to this slower response through increased coolant inventory and lower coolant temperatures.

The difference in the steam generator liquid level behavior is attributed to the physical models for natural circulation within the RCS loops. In the case of S-RELAP5, this phenomenon takes much longer to fully develop than in the MAAP4 calculation. Improved primary-to-secondary heat transfer in the MAAP4 simulation contributes to the lower core temperature excursion observed in that simulation relative to the S-RELAP5 result.

A follow-on sensitivity study was performed to validate the inconsistency in primary-to-secondary heat transfer between MAAP4 and S-RELAP5. To compensate for this apparent inconsistency, S-RELAP5 was rerun, delaying Loop 3 RCP from trip for 20 minutes (1200 s). In doing this, convective heat transfer to the secondary continues in this loop.

Leaving the one RCP running for 20 minutes in the S-RELAP5 simulation improves the magnitude and trend of the steam generator liquid levels as shown in Figure 19-246-16. Consequently, relative magnitude and trending of the fuel temperatures are also improved. The PCT comparison shown in Figure 19-246-17 shows a difference of 100°F.

Figure 19-246-12—RCS Pressure (Best Estimate)

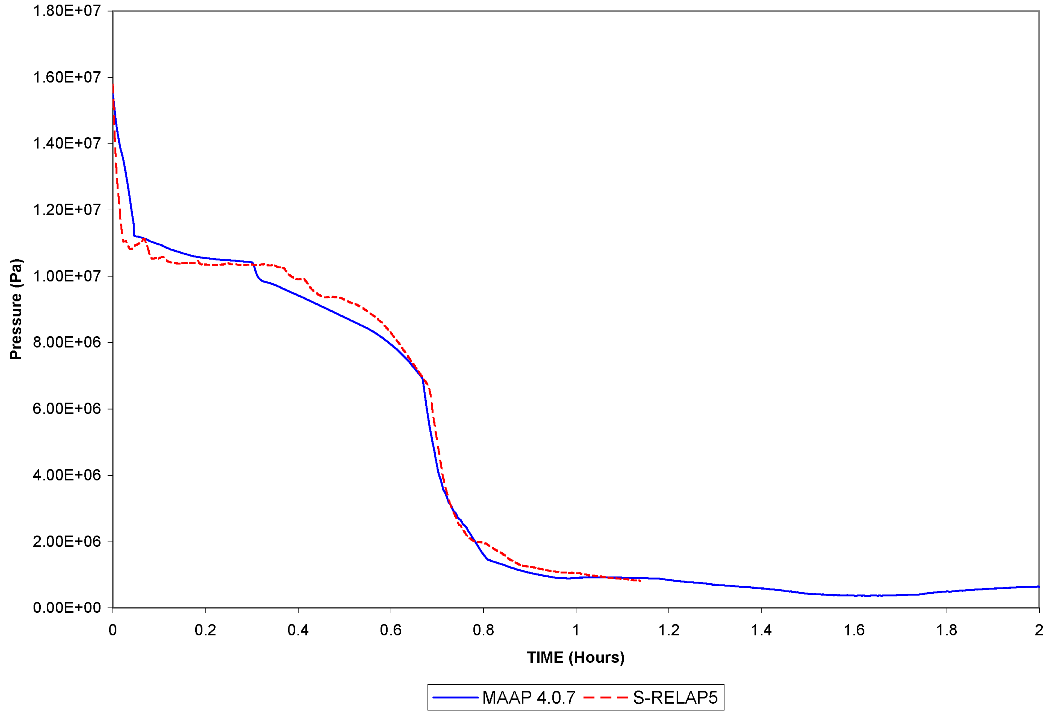


Figure 19-246-13—Break Flow (Best Estimate)

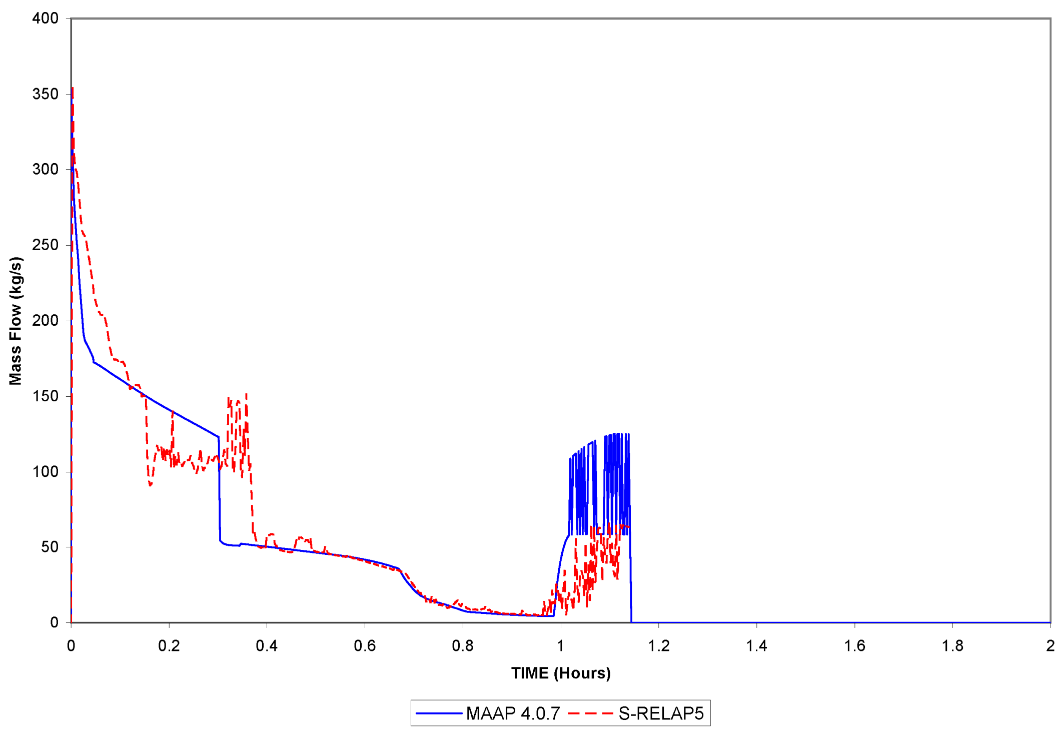


Figure 19-246-14—MAAP4 vs. S-RELAP5 Core Temperature (Best Estimate)

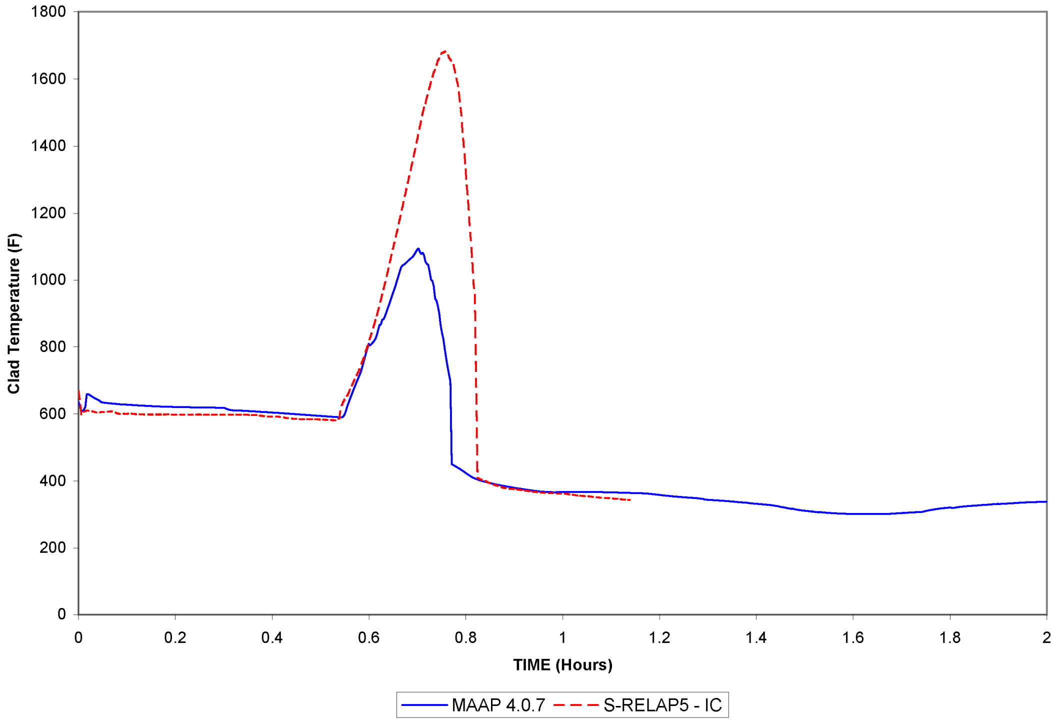


Figure 19-246-15—Steam Generator Water Level (Best Estimate)

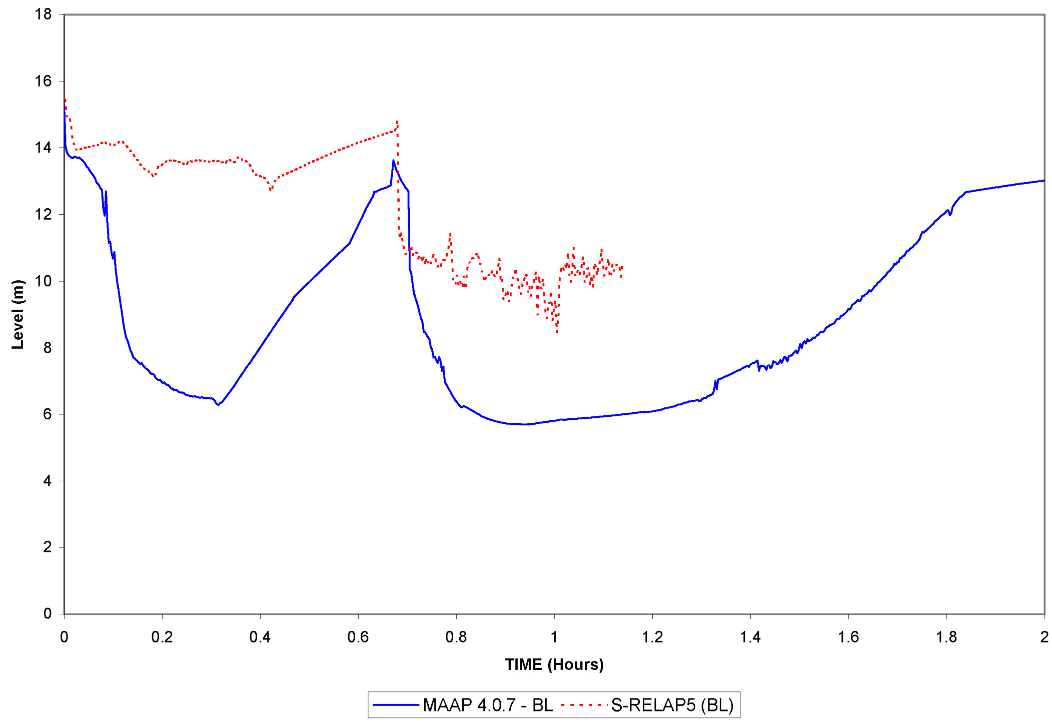


Figure 19-246-16—S/G Water Level (Tuned)

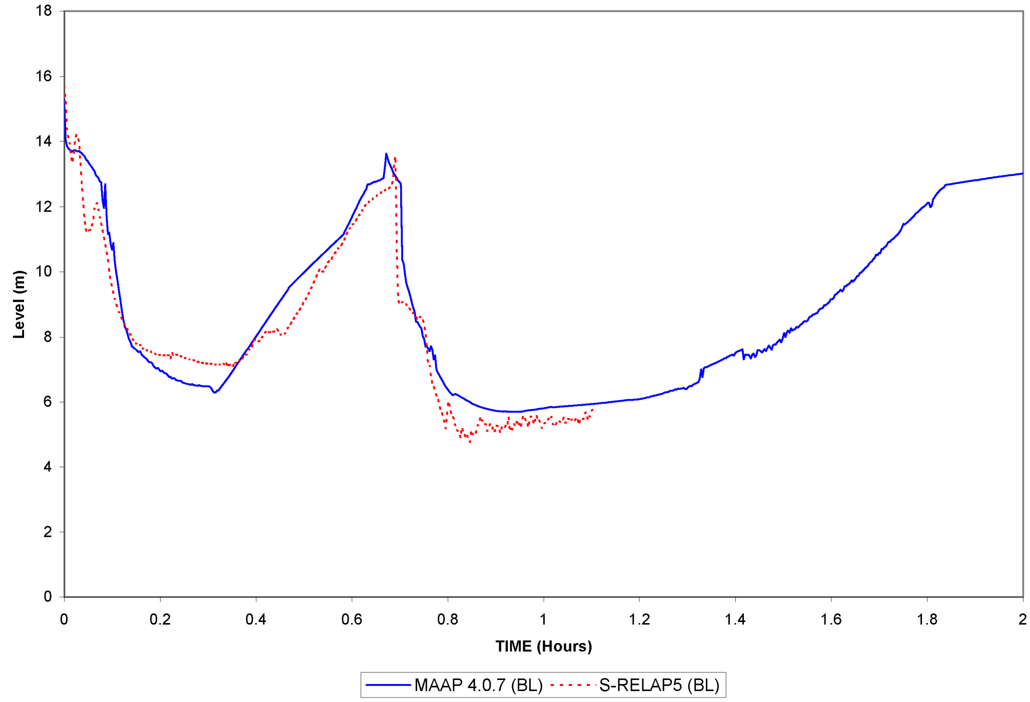
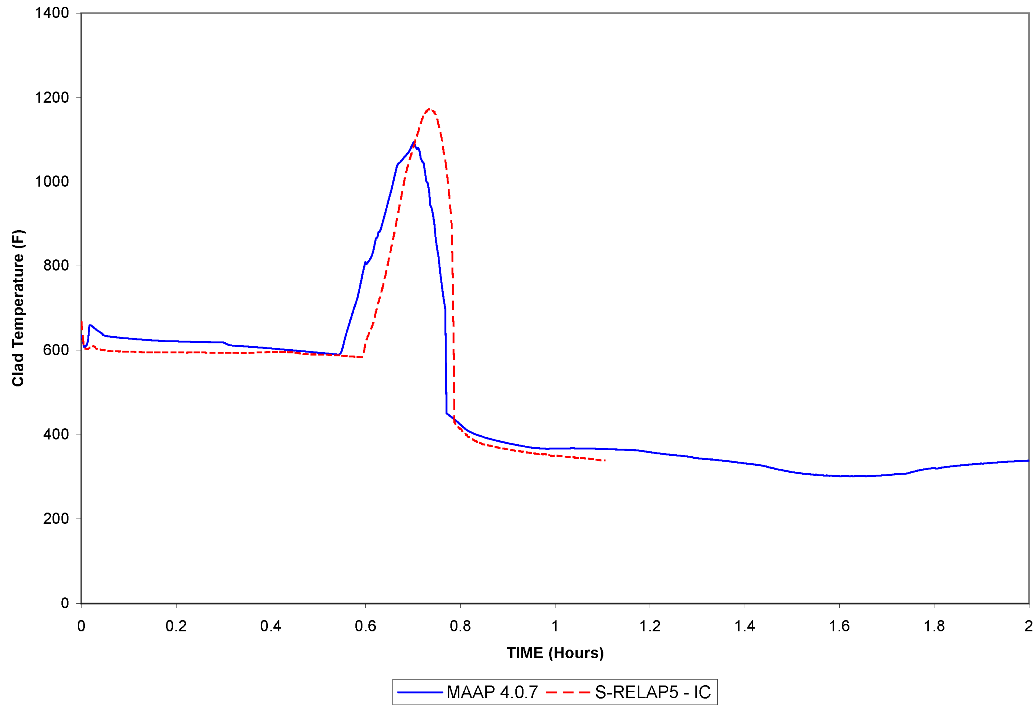


Figure 19-246-17—MAAP4 vs. S-RELAP5 Core Temperature (Tuned)



6” LOCA Case 16a1

The immediate consequence of the initiating event, a 6 inch cold leg break loss of coolant accident, is a rapid decrease in primary system pressure. Normally, a reactor trip would occur on low system pressure; however, in the MAAP4 and S-RELAP5 calculations, reactor trip is assumed coincident with the initiating event along with RCP trip and MSIV closure.

The performance of the MAAP4 code closely follows the S-RELAP5 code, as measured by both magnitudes and trends of key parameters. With the EFW, MSRV, and MSSV not functioning, primary-to-secondary heat transfer is insignificant and, unlike the other benchmarks, this is not a source of error in this benchmark. The relative difference between the MAAP4 core temperature and the S-RELAP5 hot rod is approximately 400°F, as shown in Figure 19-246-18 and Figure 19-246-19. The resulting difference in the core temperature is attributed to the fact that MAAP4 core temperature is for a core region, while the core temperature for S-RELAP5 is for a modeled hot rod.

Figure 19-246-18—MAAP4 vs. S-RELAP5 Core Temperature (Best Estimate)

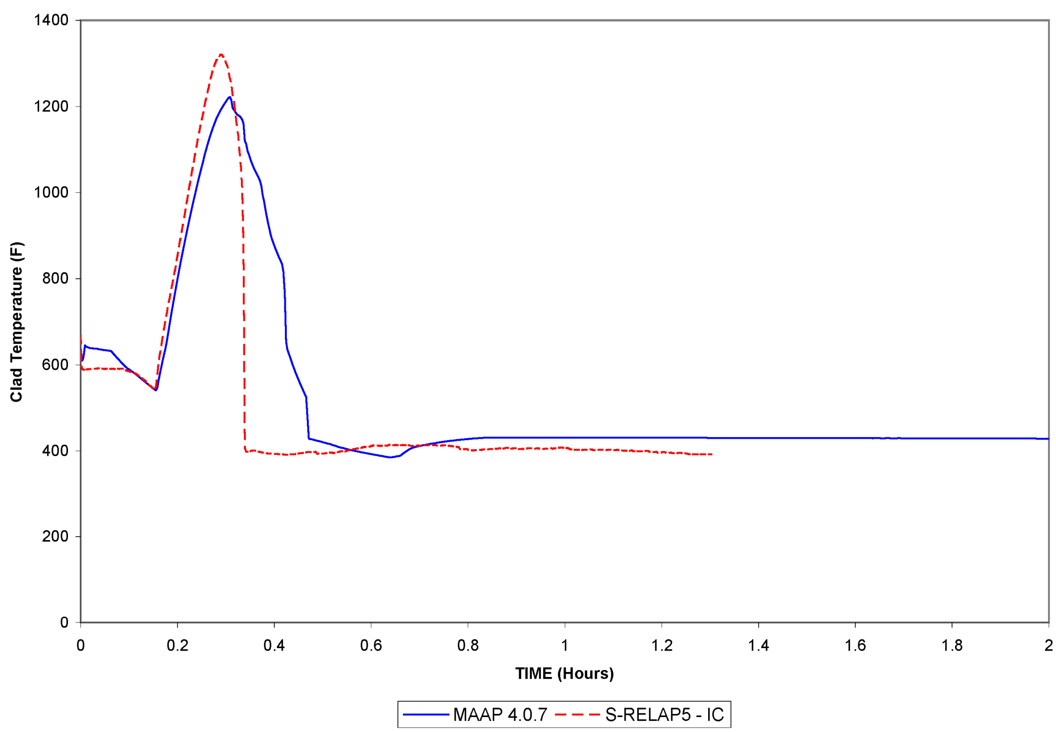
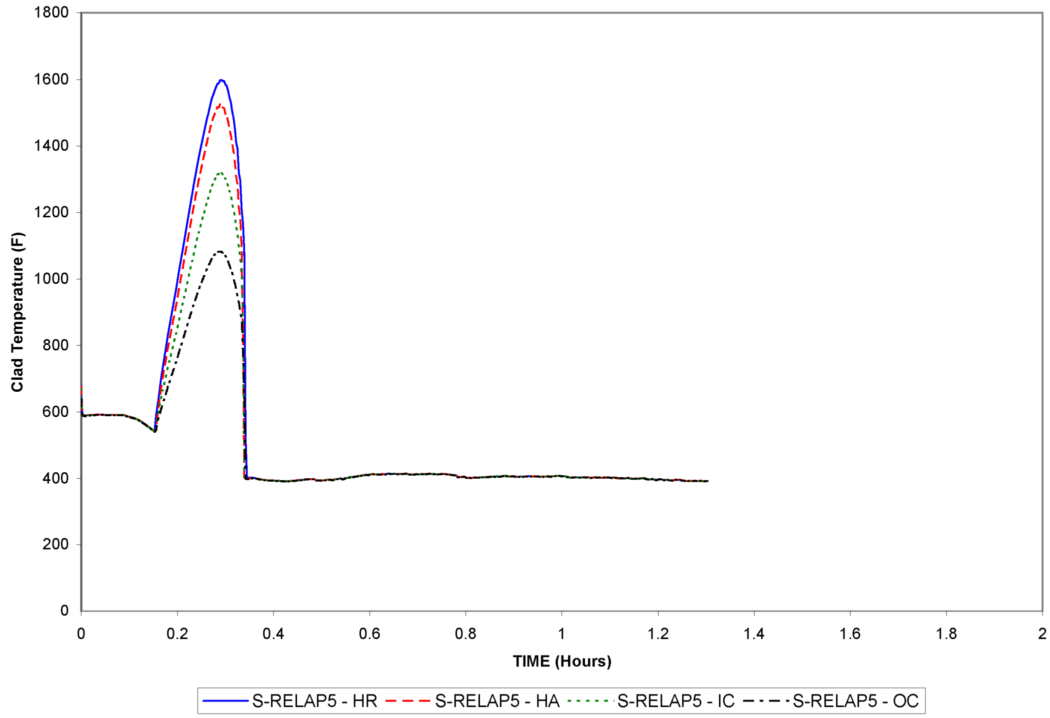


Figure 19-246-19—S-RELAP5 Cladding Temperatures (Best Estimate)



Overall, the MAAP4 results agree with the S-RELAP5 results as summarized in Table 19-246-2.

Table 19-246-2—Comparison of S-RELAP and MAAP4 Results

Case	Initiating Event	MAAP Peak Region Temp.	S-RELAP5 Equivalent / Hot Rod PCT (Best Estimate)	S-RELAP5 Equivalent / Hot Rod PCT Sensitivity Study
3c	LOFW	680°F	670°F / 670°F	670°F / 670°F
4g	LOFW	1820°F	670°F / 670°F	750°F / 700°F
4i	LOFW	>2200°F	>2200°F	N/A
4cc2	LOFW	1250°F	800°F / 850°F	N/A
12a	LOCA	680°F	670°F / 670°F	670°F / 670°F
13d	LOCA	1100°F	1700°F / 1950°F	1150°F / 1300°F
16a1	LOCA	1210°F	1300°F / 1600°F	N/A

2. The onset of a severe accident requires multiple failures that ultimately lead to a loss of core cooling and core uncovery. This is true regardless of the assumed initiating event and subsequent additional failures. The principle phenomena leading to any core melt scenario are core heat transfer, primary-to-secondary heat transfer, critical flow (e.g., from either a break or the pressurizer safety or relief valves), decay heat, and metal-water reaction. Uncertainties associated with decay heat and metal-water reaction are small; therefore, the applicability of analytical models needs only to be determined for core and primary-to-secondary heat transfer models and critical flow.

   Scenarios from the loss of coolant-accident (LOCA) and loss of feedwater (LOFW) events were selected as being representative of the two distinct mechanisms for losing core cooling: a breach of the primary system or a loss of heat sink on the secondary. With a breach of the primary system, the primary system pressure will be low at the time of core melt, low enough to signal safety injection. This is the expected condition for the full spectrum of LOCAs, including steam generator tube ruptures. With the loss of heat sink, the primary system pressure will be high at the time of core melt, preventing safety injection. This is the expected condition for certain loss of offsite-power, LOFW, and steamline break scenarios. Unisolateable steamline breaks that lead to a blowdown of one or more steam generators and other reactivity-dominant scenarios are not addressed by these benchmarks. Rather, a separate acceptance criteria applies that acknowledges the impact on core reactivity (i.e., power > 130 percent). Collectively, the phenomena assessed in the benchmark study appear in all the defined relevant scenario categories which address over 90 percent of the expected scenarios considered in defining the core damage frequency (Refer to U.S. EPR FSAR Tier 2, Section 19.2.4.2.2 and the response to RAI 6, Question 19-99).

   While there are certain scenarios for which MAAP4 is not applicable, such as the anticipated transients without scram and the large-break LOCA (at break sizes beyond the area of the largest attached pipe), events well characterized by mass and energy transport can be appropriately simulated, as demonstrated by the benchmark study. Determining the degree to which the MAAP4 results are applicable for the more likely initiating event leading to a severe accident was the principle objective of this study.

3. The following analysis assumptions were applied in either the model development activity or in processing the individual calculations:

   1) In many instances, conservative parameter values applied in safety analysis models are replaced with realistic (nominal) values (e.g, setpoints).

   2) An application-specific version of S-RELAP5 was used for the benchmark study. This version is unique in that it enables all of the calculational features available in the small-break LOCA, non-LOCA, and realistic large-break LOCA S-RELAP5 methodologies (References 1 – 3).

   3) The turbine control valve position was manually set to 0.803 based on a preliminary steady-state run. This value was not confirmed in subsequent revisions to the steady-state model; however, the thermal-hydraulic phenomena of interest in this study are not expected to be sensitive to this value.

   4) All S-RELAP5 transient calculations applied the film boiling code biases derived for the realistic large-break LOCA methodology. These biases where derived and validated against various reflooding tests for the purpose of performing best-estimate

analyses. S-RELAP5's phenomenological description of film boiling heat transfer is independent of the initiating event.

5) LOCA calculations model the break in the cold leg within the loop containing the pressurizer. This is the assumed worst break location using the realistic large-break LOCA methodology. Break location is not necessarily an important consideration in this benchmark; however, consistency between the models is necessary.

6) Safety injection in most cases is limited to just one MHSI or one LHSI. It is assumed that the functioning SI pump supplies Loop 1, an unbroken loop that is not cross connected with the broken loop. This is in contrast to the cross-connect design of the U.S. EPR that delivers safety injection to two loops from two separate pumped injection trains.

7) The choked flow model option applied at relevant junctions in S-RELAP5 applies the Homogeneous Equilibrium Model (HEM), the best-estimate model identified in the realistic large-break LOCA methodology. Consistent with the U.S. EPR MAAP4 model, a multiplier of 0.75 is used at the break in the S-RELAP5 model. For relief and safety valves off the pressurizer and steam generators, a multiplier of 1.0 is used. This is different from the MAAP4 model, which uses 0.75 at those junctions as well; however, comparison of code results show better agreement with this configuration. The critical flow model is not necessarily an important consideration in this benchmark; however, consistency between the models is necessary.

8) In all S-RELAP5 transient calculations, junctions describing the spray lines and nozzle gaps are not modeled to improve S-RELAP5 calculational robustness. The contribution of these features is not expected to be significant for the events analyzed.

9) No pressurizer spray, emergency boration system, or control volume control system were modeled in the MAAP4 and S-RELAP5 calculations with one exception: MAAP4 Case 4g did include one control volume control system; however, it is not expected to have a significant impact on the progression of the event.

10) Pressurizer relief valves are assumed to operator normally in all S-RELAP5 calculations.

4. While the results from the benchmark study showed a difference between MAAP4 and S-RELAP5 predictions of primary-to-secondary heat transfer, MAAP4 shows a good simulation of transient trends. Nonetheless, the MAAP4 prediction of cladding temperature magnitude was not sufficiently accurate to accept without compensation. For the high pressure LOFW scenarios, MAAP4 did consistently over-predict peak cladding temperature (PCT). However, this is not true for the low pressure LOCA scenarios. The revised acceptance criteria was derived based on the LOCA results.

   The imposed 1800°F limit for MAAP4 analyses was established by the estimated PCT difference between the S-RELAP5 equivalent hot rod and S-RELAP5 hot rod. In cases 13d and 16a, the difference was 250°F and 300°F, respectively. Based on those results, 400°F was judged to be acceptable. With regard to the secondary test to determine whether the scenario should be further evaluated, the imposed 1400°F limit for MAAP4 analyses was established by the estimated PCT difference between the MAAP4 result and the S-RELAP5 hot rod result. In cases 13d and 16a1, the differences were 390°F and 850°F. Based on those results, 800°F was judged to be acceptable.

5. The purpose of the benchmark was to justify the application of MAAP4 for PRA Level 1 mission success criteria. The benchmark specifically examined the principle phenomena leading to a core melt situation: core heat transfer, primary-to-secondary heat transfer, and critical flow. The results of the benchmark between MAAP4 and S-RELAP5 provided insight into the applicability of MAAP4 to simulate plant thermal-hydraulic behavior prior to core damage. Overall, the MAAP4 results show good agreement to the S-RELAP5 results. However, two significant differences were identified.

   This first difference is related to the simulation of primary-to-secondary heat transfer. This difference is associated with the onset of fully developed natural circulation in the RCS coolant loops. MAAP4 consistently predicts this occurring shortly after the RCPs have tripped, while S-RELAP5 typically showed the development of this loop flow within 20 minutes following RCP trip. The onset of natural circulation has been identified as a major uncertainty associated with using the MAAP4 code. This only impacts those events in which RCPs are tripped early in an event simulation. Consideration of the impact of this uncertainty has been incorporated into the revised mission success acceptance criteria.

   The second difference appeared in one of the LOFW scenarios in which the RCPs were allowed to remain operating for the duration of the simulation. Because of the lack of explicit interfacial drag and other code limitations, MAAP4 results exhibited unphysical phase separation. As a result, when a significant amount of RCS coolant is lost, the heat transfer benefits of entrained liquid are absent and in the situation in which the core becomes uncovered, a subsequent cladding temperature excursion occurs earlier and lasts longer than that predicted by the best-estimate S-RELAP5 code. In those scenarios MAAP4 results are considered conservative relative to a best-estimate representation. Consideration of the impact of this uncertainty has been incorporated into the revised mission success acceptance criteria.

   Other conclusions:

   - With critical flow discharge coefficients for MAAP4 derived from comparisons to S-RELAP5, break and pressurizer flows are well predicted.
   - For configurations in which the pressurizer PORV or PDS valves are opened as a RCS bleed function, the MAAP4 code will not capture the correct core heat transfer during RCS blowdown. This is a conservatism in MAAP4.
   - For LOCAs (and other low RCS pressure situations), when the uncertainty in primary-to-second heat transfer is removed, the relative difference between MAAP4 core temperature and the S-RELAP5 hot rod is approximately 400°F.
   - For LOCAs (and other low RCS pressure situations), the disabling of MHSI and accumulators along with the disabling of the emergency feedwater system (EFWS) and the MSRV and MSSV in three of the four steam generators will lead to core damage unless timely action by the operator initiates a fast cooldown.
   - For LOCAs with a break size of 2 inches and smaller and LOFW events, the combined use of one functioning MHSI, one EFWS, and one MSRV is sufficient to mitigate these events; however, overpressurization of steam generators with non-functioning MSRVs and MSSVs is possible.

- With normal primary-to-secondary heat transfer characterized by fully functioning RCP operation and core power levels at the maximum post-scram levels, steam generator liquid inventory lasts about 1 hour.

Based on the results from the benchmark study, MAAP4 is considered a good simulator of nuclear plant transient trends. However, MAAP4 prediction of cladding temperature magnitude is not sufficiently accurate to accept without compensation. The following bases for Level 1 PRA mission success criteria are to be applied when using MAAP4 for the U.S. EPR:

- MAAP4 cases resulting in a PCT of 1400°F or less are considered a success.
- MAAP4 cases resulting in a PCT of 1800°F or greater are considered a failure.
- MAAP4 cases resulting in a PCT greater than 1400°F and less than 1800°F are examined in detail, possibly with a corresponding S-RELAP5 calculation.

Reference 4 defines the original set of acceptance criteria for precluding core damage. The conclusions from the benchmark study refine the peak cladding temperature acceptance criteria. The other acceptance criteria are:

- For overpressure events, the reactor coolant system (RCS) pressure must be less than 130 percent of design pressure. The design pressure is 176 bar(abs) (2550 psia).
- For low power and shutdown events, the core must remain covered (i.e., the two phase-level in the reactor vessel is above the elevation of the top of the core).
- For all events, a 24-hour mission time is required. Therefore, EFWS should be able to inject for this period and all 4 EFW tanks should not become empty within 24 hours after event initiation.

6. The success criteria results obtained with MAAP 4.0.7 are summarized in Table 19-246-3 for the initiating events considered in the U.S. EPR PRA. For each initiating event, the table identifies the required safety functions, and the different combination of mitigating systems required to achieve these functions. The systems credited in the PRA for each safety function are as follows:

   - Reactivity control: the chemical and volume control system (CVCS) and the extra boration system (EBS).
   - Secondary cooling: the steam generator (SG) inventories, the startup and shutdown system (SSS), the main feedwater system (MFW), the emergency feedwater system (EFWS), the main steam bypass (MSB), the MSRV, and the MSSV.
   - RCS integrity and pressure control: the pressurizer safety relief valves (PSRV) and severe accident depressurization valves (SADV).
   - Inventory control: the LHSI, MHSI, and accumulators.
   - Containment cooling: the LHSI and severe accident heat removal system (SAHRS).

   Additional information on the success criteria for the fast cooldown and feed and bleed functions can be found in the responses to RAI 7, Question 19-060 and Question 19-077.

Table 19-246-3—Summary of U.S. EPR PRA Success Criteria from MAAP 4.07 (6 Sheets)

Initiator	Reactivity Control	Secondary Cooling (SC)	RCS Integrity and Pressure Control	Inventory Control	Containment (IRWST Cooling)
Reactor Trip (RT)	Guarantee Success (a)	1/4 MFW to 1SG + MSB or 1/4 MFW to 1SG + 1/4 MSRV or 1/4 MFW to 1SG + 1/8 MSSV or SSS to 1SG + MSB or SSS to 1SG + 1/4 MSRV or SSS to 1SG + 1/8 MSSV or 1/4 EFW (c) + 1/4 MSRV or 2/4 EFW (c) + 1/8 MSSV	Guarantee Success (d)(f)	Guarantee Success (e)	Not Required (NR)
		Failed (SG relief 1/3 paths/SG is required)	Operator Action (OP) + 3/3 PSV or OP + 1/2 SADV	1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR
Turbine Trip (TT)	79 of 89 rods insert (b) Or See ATWS	Same as RT	Same as RT	Same as RT	Same as RT
Loss of Main Feedwater (LOMFW)	79 of 89 rods insert (b) Or See ATWS	<u>Same as RT except the following impacts:</u> 1. MFW is unavailable 2. SSS model includes conditional failure probability	Same as RT (f)	Same as RT	Same as RT

Table 19-246-3—Summary of U.S. EPR PRA Success Criteria from MAAP 4.07 (6 Sheets)

Initiator	Reactivity Control	Secondary Cooling (SC)	RCS Integrity and Pressure Control	Inventory Control	Containment (IRWST Cooling)
Loss of Condenser (LOC)	79 of 89 rods insert (b) Or See ATWS	Same as RT except the following impacts: 1. MSB is unavailable 2. MFW depends on DWS 3. SSS depends on DWS	Same as RT (f)	Same as RT	Same as RT
Spurious MSIV Closure	79 of 89 rods insert (b) Or See ATWS	Same as LOC plus secondary pressure relief (EFW and SSS models) may require operators to open MSIV bypass to share SG relief and safety valves	Same as RT (f)	Same as RT	Same as RT
Loss of Balance of Plant (LBOP)	79 of 89 rods insert (b) Or See ATWS	Same as RT except the following impacts: 1. MFW is unavailable 2. MSB is unavailable 3. SSS is unavailable	Same as RT (f)	Same as RT	Same as RT
Small Break LOCA (SBLOCA)	(g)	**Partial Cooldown (PCD)** 1/4 MFW to 1SG + MSB or 1/4 MFW to 1SG +1/4 MSRV or SSS to 1SG + MSB or SSS to 1SG + 1/4 MSRV or 1/4 EFW + 1/4 MSRV	NR (PCD)	1/4 MHSI	1/4 LHSI or OP + SAHR

Table 19-246-3—Summary of U.S. EPR PRA Success Criteria from MAAP 4.07 (6 Sheets)

Initiator	Reactivity Control	Secondary Cooling (SC)	RCS Integrity and Pressure Control	Inventory Control	Containment (IRWST Cooling)
		Fast Cooldown (FCD) OP + 1/4 MFW to 1SG + 1/4 MSRV or OP + SSS to 2SG + 2/4 MSRV or OP + 2/4 EFW + 2/4 MSRV	NR (FCD)	1/4 ACC + 1/4 LHSI	NR (LHSI satisfies)
		Failed (SG relief 1/3 paths/SG is required)	OP + 3/3 PSV or OP + 1/2 SADV	2/4 ACC + 1/4 LHSI	NR (LHSI satisfies)
				1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR
Medium Break LOCA (MBLOCA)	(g)	**PCD** MSB + Inventory of 4/4 SG or 1/4 SG MSRV + Inventory of 4/4 SG (No EFW required if 4/4 SG inventory available)	NR (PCD)	1/3 MHSI	1/4 LHSI or OP + SAHR
		FCD OP + 4 SG MSRV (No EFW required if 4/4 SG inventory available)	NR (FCD)	1/3 ACC + 1/3 LHSI	NR (LHSI Satisfies)
		Failed (1 MSSV required)	OP + 3/3 PSV or OP + 1/2 SADV	1/3 MHSI + 1/3 ACC	1/4 LHSI or OP + SAHR
				2/3 ACC + 1/3 LHSI	NR (LHSI Satisfies)

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Initiator	Reactivity Control	Secondary Cooling (SC)	RCS Integrity and Pressure Control	Inventory Control	Containment (IRWST Cooling)
Large Break LOCA (LBLOCA)	(g)	**NR**	NR	1/2 MHSI + 1/3 ACC + 1/2 LHSI or 2/3 ACC + 1/2 LHSI	NR (LHSI satisfies)
Steam Generator Tube Rupture (SGTR)	(g)	Success Criteria requires Secondary Cooling (1 of 3 SG with MSRV) with operator actions to isolate faulted SG and conduct a cool down to stop loss of inventory Primary Feed & Bleed Cooling provides an alternative success if secondary cooling fails			
Steam Line Break Outside Containment (SLBO)	2/4 SG ISO or 1/2 EBS	4/4 SG ISOL + 1/4 EFW/MSRV	NR	NR	NR
		4/4 SG ISOL + SC Fails	OP + 3/3 PSRV or OP + 1/2 SADV	1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR
		3/4 SG ISOL + 1/3 EFW/MSRV	NR	NR	NR
		3/4 SG ISOL + SC Fails	OP + 3/3 PSRV or OP + 1/2 SADV	1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR
		2/4 SG ISOL + OP + 1/2 EFW/MSRV + 1/4 RHR	NR	NR	NR
		2/4 SG ISOL + OP + SC Fails	OP + 3/3 PSRV or OP + 1/2 SADV	1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR
		2/4 SG ISOL + OP + 1/2 EFW/MSRV + 1/4 RHR	NR	OP + 1/2 EBS	NR
		2/4 SG ISOL + OP + SC Fails	OP + 3/3 PSRV or OP + 1/2 SADV	1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR

Table 19-246-3—Summary of U.S. EPR PRA Success Criteria from MAAP 4.07 (6 Sheets)

Initiator	Reactivity Control	Secondary Cooling (SC)	RCS Integrity and Pressure Control	Inventory Control	Containment (IRWST Cooling)
		0 or 1/4 SG ISOL + SC assumed to fail	OP + 3/3 PSRV or OP + 1/2 SADV	1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR
		0 or 1/4 SG ISOL + SC assumed to fail	OP + 3/3 PSRV or OP + 1/2 SADV	1/4 MHSI + 1/4 ACC	1/4 LHSI or OP + SAHR
Anticipated Transient without SCRAM (ATWS)	2/2 CVCS from IRWST or 1/2 EBS	SSS + MSB or SSS + 2/4 MSRV or 2/4 EFW +2/4 MSRV	1/3 PSV + Trip 2/4 RCP + 3/3 PSV Close or 2/3 PSV + Trip 1/4 RCP + 3/3 PSV Close	NR	NR
	1/2 CVCS from IRWST and 2/2 EBS	**<u>PCD (PSV failure to reclose)</u>** SSS + MSB or SSS + 2/4 MSRV or 2/4 EFW +2/4 MSRV	1/3 PSV + Trip 2/4 RCP. Or 2/3 PSV + Trip 1/4 RCP	2/4 MHSI (PCD)	1/4 LHSI or OP+SAHR
Loss of Offsite Power (LOOP)	79 of 89 rods insert (b) Or See ATWS	Success Criteria require coping for 2 hours (1/4 I&C Bus + All 4 SG Inventory Available with Steam Relief are required or core damage end state) Success with offsite power recovery (2 hours) is allowed if there is no seal LOCA or 1 hour if there is a seal LOCA and coping for 2 hours is successful			
Loss of Component Cooling Water (LOCCW)	79 of 89 rods insert (b) Or See ATWS	There are several CCWS initiating events with varying impacts on frontline systems dependent on CCWS train failures. This initiator also represents one of the more likely ways to initiate a RCP seal LOCA. The success criteria are very similar to transients and SBLOCA (RCP seal LOCA) above.			
Loss of 1 Emergency Power Division (31BDA)	79 of 89 rods insert (b) Or See ATWS	Same as RT except for impacts associated with 31BDA failure			

Notes:

a. Plant spurious scrams that were not real protection system challenges (spurious success of reactivity control caused trip).

b. The reactor control system of the U.S. EPR is equipped with 89 control rods. Accident analyses of the KONVOI NPP have shown that the function of reactor trip will be fulfilled if less than 10 neighboring control rods fail. The same success criteria is used here since the core design of the U.S. EPR and KONVOI are similar and the U.S. EPR core has lower specific power and the KONVOI core is equipped with a smaller number of control rods (i.e., 61). Reactor trip also requires input signals and reactor protection system logic and instrumentation and control (I&C) for success.

c. The model does not credit aligning residual heat removal system (RHR) shutdown cooling as an alternative to requiring EFW for 24 hours. 1/4 EFW only required via MSSV if RCPs are tripped (e.g., LOOP conditions).

d. Primary safety valves are not challenged (with or without pressurizer spray) unless MSRVs fail and or several trains of EFW fail. Therefore, the probability of challenging and sticking open a pressurizer safety valve (PSV) is bounded by SBLOCA. The probability of an RCP seal LOCA is low and is analyzed for dominating initiating events such as LOOP and LOCCW.

e. No inventory makeup is required unless secondary cooling fails and primary feed and bleed cooling is required (see secondary cooling function).

f. It is assumed that the frequency of events that open a PSV (e.g., with only 1/8 MSSV) is unlikely when compared to small LOCA initiating event sequences.

g. Reactivity control and its mitigation (i.e., ATWS) are modeled for all transients. The frequency of non-transients is lower and mitigation success criteria is usually relaxed for non-transient ATWS events (e.g., safety injection ensures borates water sources are injected). Thus, the risks from these events are low and bounded by transient events.

References for 19-246:

1. EMF-2103 (P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” April 2003.
2. EMF-2328 (P)(A), “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” March 2001.
3. EMF-2310 (P)(A), “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” May 2004.
4. ANP-10274NP, “Probabilistic Risk Assessment Methods Report," December 15, 2006.

FSAR Impact:

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