# REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**APR1400 Design Certification**

**Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD**

**Docket No. 52-046**

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|---|---|
| **RAI No.:** | **432-8377** |
| **SRP Section:** | **SRP 19** |
| **Application Section:** | **19.1** |
| **Date of RAI Issue:** | **03/08/2016** |

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## Question No. 19-54

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

Section 19.1.4.2.1.2.1 of APR1400 design control document (DCD) Rev. 0 states the following:

> Detailed evaluation of phenomena that affect containment failure timing, fission product releases, or that may have an impact on downstream top events are treated through the use of decomposition event trees (DETs). The containment ultimate pressure capacity and severe accident phenomena analysis results are needed for quantification of the DETs. This CET/DET approach allows a relatively detailed treatment of the phenomena affecting containment performance while maintaining a relatively simple and easily understood CET.

APR1400 DCD Rev. 0 does not provide a description of DET analysis. Update the DCD providing a description of DETs.

## Response – (Rev. 1)

The decomposition events trees (DETs) are used for quantification of complex containment event trees (CETs). The DETs for the general CET and special CETs are described in Subsection 19.1.4.2.1.2.3 (revised to 19.1.4.2.1.2.5) in DCD 19.1. The description and figures of DETs are added in Subsection 19.1.4.2.1.2.5 in DCD 19.1 (See Attachment).

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### Impact on DCD

DCD Subsection 19.1.4.2.1.2.5 (revised from 19.1.4.2.1.2.3 by the response of RAI 432-8377, Question 19-55) is revised as shown in the Attachment.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on the Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

There is no impact on any Technical, Topical, or Environmental Report.

RAI 432-8377 - Question 19-54

RAI 432-8377 - Question 19-54_Rev. 1

19.1.4.2.1.2.~~4~~ 5

## ~~19.1.4.2.1.2.3~~ ~~CET~~ CET/DET Phenomenological Evaluations

The MAAP code was used to support many of the ~~CET~~ CET/DET phenomenological evaluations. MAAP evaluations included evaluations of core melt, RCS failure, containment pressurization, ex-vessel core-concrete interactions, and releases from the containment. Containment failure due to overpressurization was considered using the results of the containment ultimate capacity evaluation. Many other calculations were performed to support the CET. ~~Referring to the general CET presented in Figure 19.1-42, the following top events are described:~~

The general containment events tree (CET) has eight headings which are quantified using decomposition event trees (DETs). The DETs for the general CET are presented in Figure 19.1-46a through 19.1-46h. and the DETs for the special CETs are presented in Figure 19.1-46i through 19.1-46l.

~~a. RCSFAIL – Mode of RCS Failure Before Vessel Breach~~

~~b. MELTSTOP – In-Vessel Core Melt Arrest~~

~~c. DCF – Dynamic Containment Failure~~

~~d. ECF – Early Containment Failure~~

~~e. CSLATE – Late Containment Heat Removal Recovery Failure~~

~~f. DBCOOL – Ex-Vessel Debris Coolability~~

~~g. LCF – Late Containment Failure~~

~~h. BMT – Basemat Melt-Through~~

a. Decomposition Event Tree for RCSFAIL

~~1) RCSFAIL – Mode of RCS Failure Before Vessel Breach~~

The question posed in this DET is whether there is a severe accident-induced failure of the hot leg or steam generator tubes during severe accident progression. For high pressure core damage sequences, natural circulation of superheated gases can occur in the reactor coolant system after the core has uncovered. Natural circulation is a result of differences in gas density between the various regions of the reactor coolant system. Natural circulation of gases in the reactor coolant system during the severe accident is a significant phenomenon because it transports heat from the overheating core into the structure of the upper plenum, hot leg, surge line, and SG tubes. If the natural circulation flow of gases continues, it can cause failure of the hot leg, surge line, or SG tubes due to creep. However, if the SG tubes are cooled by water from the secondary side, the high temperature in the SG tubes will not occur. The induced SGTR event is possible only for dry and depressurized SG sequences.

The consequence of the induced primary system failures depends on the failure location. If the hot leg or surge line fails, the RCS is depressurized and many phenomena resulting from high RCS pressure at vessel breach, which threaten the containment integrity, are prevented. If the steam generator tubes fail, the direct release path of fission products from the RCS to the environment would be available. Note that these failure modes are mutually exclusive. Once failure occurs at any location, the resulting depressurization and reduction in stress on other components precludes subsequent failures. (That is, if the induced SGTR occurs, the induced hot leg or surge line failure will not occur. If the induced hot leg or surge line fails first, the induced STGR will not occur.) By considering the source term release consequences of each induced failure location, the induced SGTR is assumed to occur prior to hot leg or surge line failure in this analysis.

In terms of severe accident-induced SGTR, two unique induced tube rupture modes are possible during severe accident progression:

Pressure-induced SGTR (PI-SGTR): PI-SGTR results from a high differential pressure across the steam generator tubes occurring when RCS pressure is at the pressurizer relief valve (i.e., POSRV) setpoint and an SG is fully depressurized via a stuck-open ADV or MSSV. Note that core damage events that are expected to occur early in the sequence such as an MSLB or ATWS that involves induced SGTR are not included in this category. Such events were treated as bypass events previously in PDS analysis.

Thermally induced SGTR (TI-SGTR): TI-SGTR addresses the probability that high tube temperatures caused by the natural convection process after core damage, coupled with a significant RCS/SG pressure differential, will induce a rupture of SG tubes prior to hot leg and surge line failures.

~~2) MELTSTOP - In-Vessel Core Melt Arrest~~

b. Decomposition Event Tree for MELTSTOP

RV by cooling of the outer wall of the RV. The severe accident phenomena that occurred outside the RV and threatened containment integrity would be prevented.

When the vessel failure is prevented by effectively cooling the core by in-vessel injection or external RV cooling, the containment may eventually fail due to steam-induced overpressurization if containment heat removal is lost.

If core melt is arrested before vessel failure and containment heat removal is available, only limited hydrogen production would be expected and containment overpressurization would be limited. DCH would not be a threat. As a result, containment failure is extremely unlikely. Furthermore, radionuclide release from the debris would be limited and long-term revaporization of radionuclides deposited on RCS surfaces would be largely avoided. Hence, because the containment does not fail and the radionuclide release is limited, the environmental source terms for core damage sequences that are successfully terminated in-vessel are expected to be very small. The sequences of this type are very similar to the accident at Three Mile Island Unit 2 (TMI-2).

c. Decomposition Event Tree for DCF

~~3) DCF – Dynamic Containment Failure~~

This event determines whether the very energetic phenomenon only depending on the RCS pressure at vessel breach occurs and results in early containment failure at the time of vessel breach. This event can be included in the next event (Early Containment Failure). For convenience's sake, however, these phenomena are considered separately from Early Containment Failure. Two possibilities are considered:

a) No dynamic containment failure

b) Dynamic containment failure

In this top event, three energetic phenomena are considered:

a) In-vessel steam explosion ("Alpha-mode" containment failure)

Containment failure by direct liner attack: This event addresses the potential for a containment failure due to the direct impact of corium particles ejected from the RCS at high pressure. This potential containment challenge results from a high pressure RV discharge of energetic corium debris interacting with the containment shell (concrete and steel liner). Direct containment shell attack by high temperature core debris requires that the debris be relocated from the RCS to the containment. An ex-vessel distribution of the debris leading to direct contact with the containment shell is a minimal requirement for the occurrence of this postulated failure mechanism. Low pressure vessel failure events will lead to the deposition of core debris entirely within the reactor cavity. This would preclude direct contact with the containment shell liner. Thus, only high pressure vessel failure events need to be assessed for direct shell attack. A high pressure vessel failure can lead to debris dispersal and potential ejection of a portion of the debris from the reactor cavity into adjacent containment shells.

For the APR1400, this issue was found to be negligible because even if the RV were to fail at high RCS pressure, the containment geometry of the APR1400 strongly inhibits the possibility of debris entrainment to the containment shell. However, the analysis conservatively assigned a small (0.001) probability of containment failure in high pressure sequences (zero in medium and low pressure sequences).

d. Decomposition Event Tree for ECF

~~4) ECF – Early Containment Failure~~

This event determines whether a gross failure of containment occurs at or soon after RV failure. Four possibilities are considered:

a) No early containment failure without hydrogen burn

b) No early containment failure with hydrogen burn

c) Early containment failure (leak)

d) Early containment failure (rupture)

issue of rapid steam generation has been divided into two related containment threats. These are the ex-vessel steam explosion-induced containment failure and the quasi-static steam pressurization containment failure event. It should be noted that the rapid steam generation containment failure mode presumes that the cavity is water filled so that DCH loadings are insignificant and that the steam atmosphere is sufficient to inert post-VB hydrogen burns.

In the APR1400 Level 2 analysis, the probability of containment failure due to the above phenomena was found to be negligible. Despite the negligible potential for any of these challenges to fail the APR1400 containment, a small probability was conservatively assigned to each phenomenon. These small probabilities do not adversely skew the results, but allow for sensitivity evaluations, which are performed in the results section.

e. Decomposition Event Tree for CSLATE

~~5) CSLATE Late Containment Heat Removal Recovery Failure~~

This event determines if containment heat removal is available late (after vessel breach) in the accident sequence. It is assumed that late overpressurization can be avoided if containment heat removal is available. In this analysis, the containment spray system and the emergency containment spray backup system (ECSBS) are considered to function for containment spray. The branches for this event are:

a) No late containment spray available

b) Late containment spray available

For containment heat removal to be available after vessel breach, the containment heat removal function should be available early in the accident scenario and the function maintained after vessel failure, or the early failed containment heat removal should be recovered. Failure of equipment inside containment is considered to be 100 percent non-recoverable. For cases where the containment heat removal was unavailable because the operator had failed to initiate containment spray, containment spray would be initiated before containment failure given the available time and indications. In cases in which ac power was unavailable, containment spray would be operated if power is recovered prior to containment failure.

f. Decomposition Event Tree for DBCOOL

~~6) DBCOOL – Ex-Vessel Debris Coolability~~

This event determines whether the core debris relocated into the reactor cavity is rapidly quenched by an overlying water pool. Though the APR1400 has been designed with a large cavity area and the cavity flooding system, in this analysis, it is considered that the corium may not be well cooled by an overlying water pool. Three possibilities are considered:

a) Ex-vessel debris not cooled without an overlying water pool

b) Ex-vessel debris not cooled with an overlying water pool

c) Ex-vessel debris cooled

The debris in the reactor cavity can be submerged by water if safety injection is operating after vessel failure or the reactor cavity flooding system operates. If the debris is cooled, its only subsequent challenge to the containment is steam overpressurization due to the continued addition of decay heat to the cooling water and hence to the containment.

Physically, the debris is not cooled if the debris surfaces that are exposed to the heat-removing medium are not large enough with respect to the heat generating volume to prevent high temperatures from being attained. High surface-to-volume ratios indicate that the debris is being spread thinly over a large surface area. The geometry of the cavity (floor area) is an important factor.

g. Decomposition Event Tree for LCF

~~7) LCF – Late Containment Failure~~

This event determines whether a gross failure of containment due to overpressurization and/or overtemperature occurs late in the accident sequence. (“Late” is defined as being greater than after a few hours or a few days following RV failure.) This event is similar to the event for early containment failure, with the accident in progress for a significant amount of time as the obvious difference. Three possibilities are considered as follows:

a) No containment failure

b) Late containment failure (rupture)

c) Late containment failure (leak)

The phenomena that could potentially contribute to a late containment failure are:

a) Overpressurization caused by production of steam and/or non-condensable gases

b) Late hydrogen burn

c) Overtemperature failure of containment penetration sealants

The primary cause of failure of the containment is the steam overpressurization resulting from the loss of the containment heat removal. If the containment sprays (including ECSBS) are not available and the reactor cavity is flooded with water, the containment would finally fail due to steam overpressurization. The steam overpressurization process is slow and it takes a long time to reach the containment failure pressure. The containment pressurization may stop if a small leakage path exists.

The possibility of late containment failure due to a late hydrogen burn was evaluated with conservative assumptions that an ignition source is available when the maximum hydrogen concentration is reached. Pressure resulting from a late hydrogen burn through the adiabatic isochoric complete combustion (AICC) process was calculated using the MAAP code for various accident sequences. The probability of containment rupture, leak, or no containment failure was calculated based on the resultant pressure and the containment ultimate pressure capacity (UPC).

Overtemperature failure of containment seals is also considered, and was found to be negligible. However, the analysis conservatively assigned a small probability of containment failure in sequences with failed containment sprays and a dry cavity.

h. Decomposition Event Tree for BMT

~~8) BMT – Basemat Melt-Through~~

This event determines whether the containment can fail due to basemat melt-through. Two branches are considered:

a) No basemat melt-through

b) Basemat melt-through

The containment can fail due to basemat melt-through (even if the cavity is filled with water) if the molten debris is not coolable. Note that if the containment heat removal function is not available and the reactor cavity is wet, it is assumed that overpressure failure occurs and basemat melt-through is neglected since the offsite consequences of basemat melt-through would be small compared with those of overpressure failure.

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Successful cooling of the cavity debris bed implies that erosion-induced containment failure modes will not occur and that the radiological releases are attributable to either an alternate failure mode or containment leakage (assuming no other containment failure mode is identified).

19.1.4.2.1.3 Release Category Evaluations

The end points of the containment event tree (CET) represent the outcomes of possible accident progression sequences. These end points describe complete severe accident sequences from initiating event to release of radionuclides to the environment. The number of CET end points is large, and a detailed source term analysis for all of the end points is not feasible. In addition, such analyses for all accident sequences are not necessary because the amount and timing of the fission product release to the environment are similar for many of the accident sequences. Therefore, to reduce the source term evaluation effort, the CET accident sequences are grouped into a representative number of release categories that exhibit similar characteristics.

A particular release category consists of a group of CET end points that have similar source term governing characteristics. Once the release categories are determined, various accident sequences are allocated to each category. The APR1400-specific source terms are evaluated using the MAAP computer code for one sequence that best represents the

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i. Decomposition Event Tree for SGTR

The special CET for SGTR, which applies to PDS 01 and PDS 02, has one heading which is quantified using DET. This DET addresses the probabilities to reduce the releases of fission products when the SGTR event occurs and continuous feedwater is supplied into ruptured SGs. When the SGTR event occurs and the AF system continuously supplies the feedwater into the ruptured SG, the saturated water pool is available on the secondary side of ruptured SG. If the SG water level can be maintained at the normal level by AF system's operation, the SG water level is sufficient to submerge the SG tube break point. Following the EPRI TR-101869 (Reference 31), if the SGTR occurs and the AF system supplies the feedwater into the ruptured SG, the fission products from the core are assumed to enable to be scrubbed.

j. Decomposition Event Tree for ISLOCA

The special CET for ISLOCA, which applies to PDS 03 and PDS 04, has three headings which are quantified using DETs. When the ISLOCA event occurs and the SI system continuously supplies the water into the RCS, the IRWST water can be transported into the auxiliary building via the break point of interfacing system's piping. Potential paths for ISLOCA, which is the interface with the RCS, include the safety injection (SI) system, the shutdown cooling (SC) system, chemical and volume control (CV) system, and the sampling system. After screening process for potential paths is performed, the remaining six lines are considered as ISLOCA sources. They are four SI lines and two SC suction lines. In this analysis, it is conservatively assumed that ISLOCA occurs at the SC suction lines, because the piping size of SC line is greater than that of SI line.

It is assumed that the fission product can be scrubbed when ISLOCA occurs and the break point is the bottom level of auxiliary building. The probability that the ISLOCA occurs at relatively low elevation is evaluated by considering the piping length.

k. Decomposition Event Tree for Containment Isolation Failure

The special CET for Containment Isolation Failure, which applies to PDS 05 and PDS 06, has one heading which is quantified using DET. This DET addresses the probabilities to reduce the releases of fission products when the containment isolation fails and containment spray operates successfully.

Added

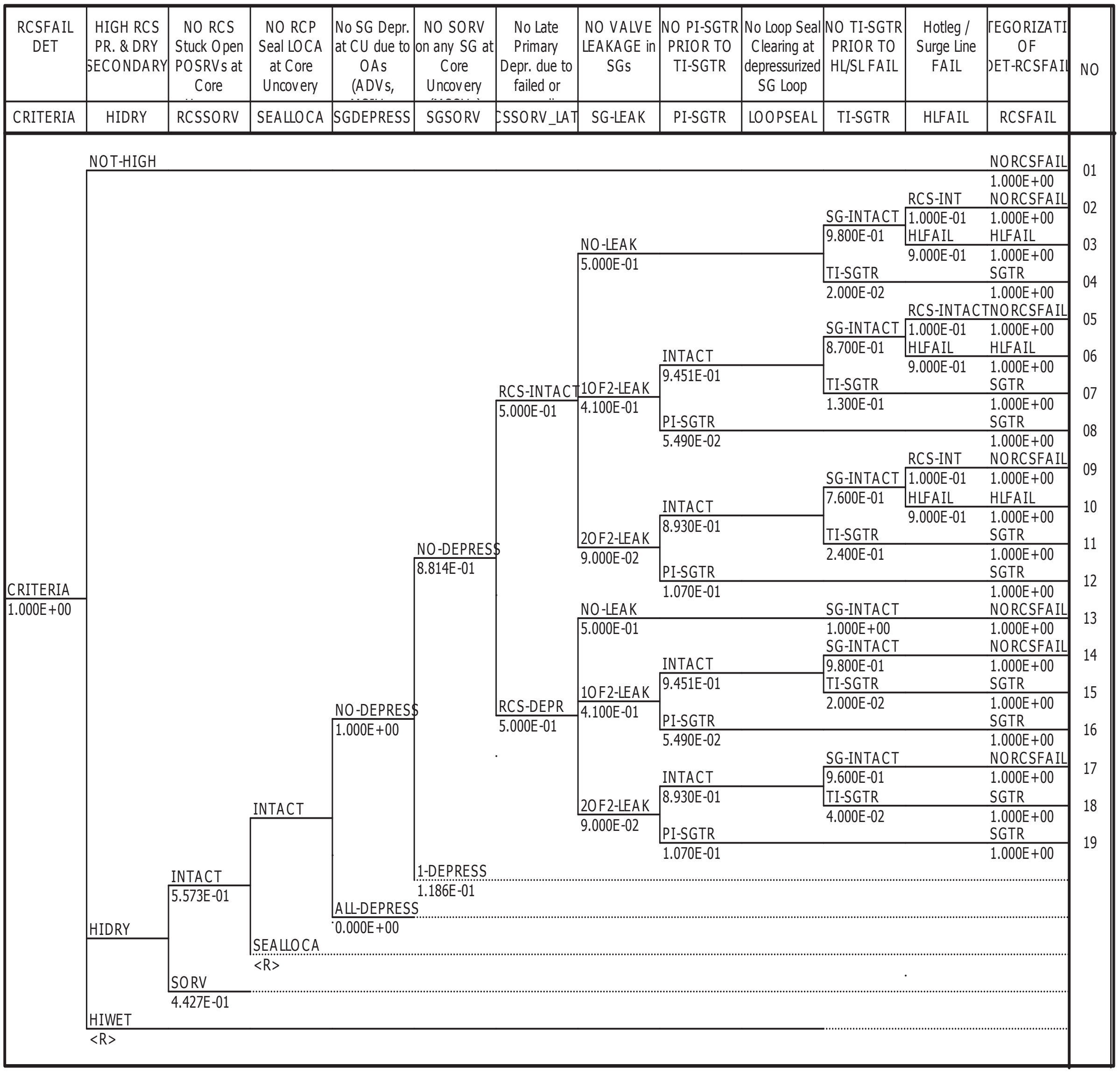


**Figure 19.1-46a RCSFAIL Decomposition Event Tree (1/6)**

Added

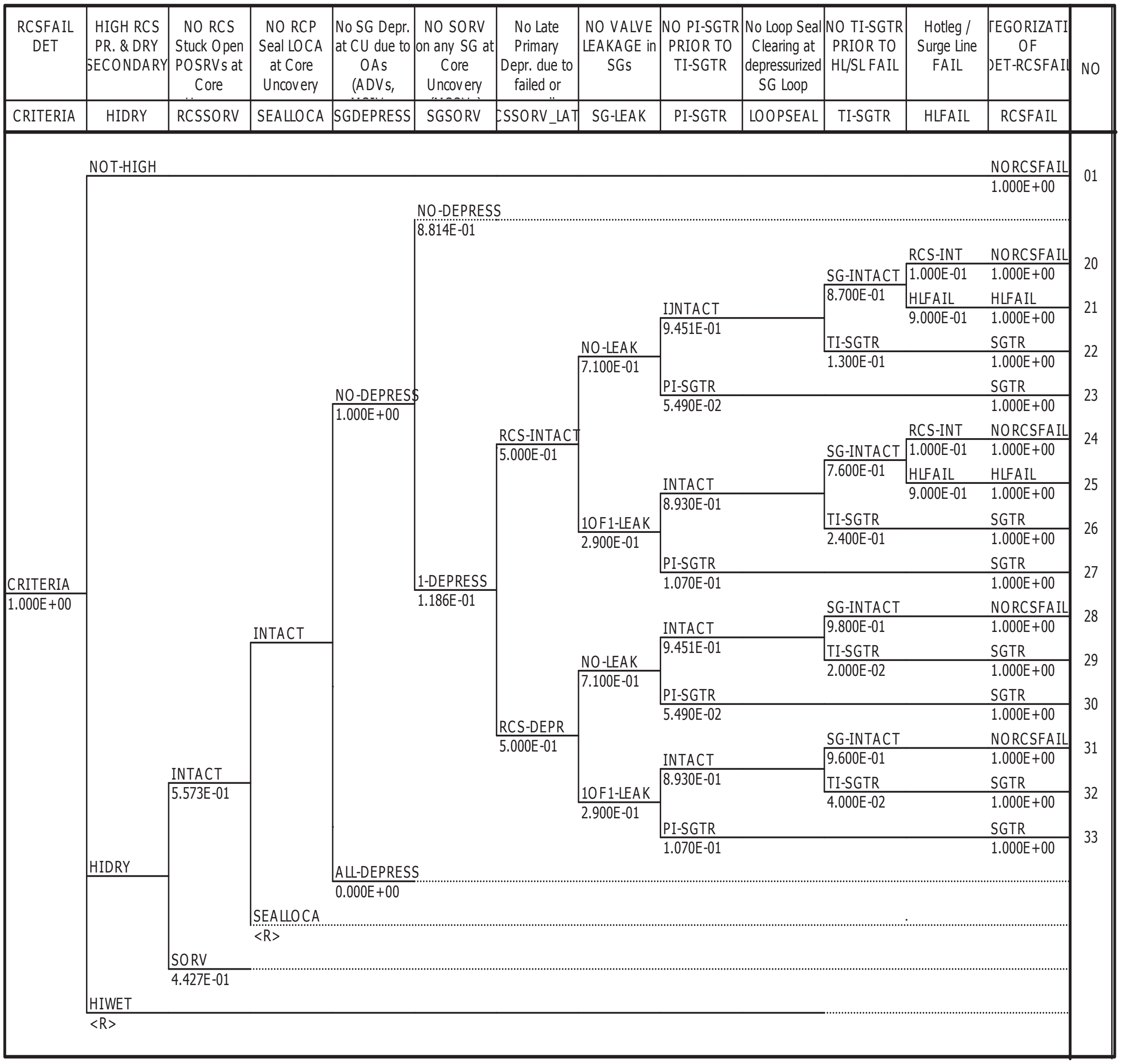


**Figure 19.1-46a RCSFAIL Decomposition Event Tree (2/6)**

Added

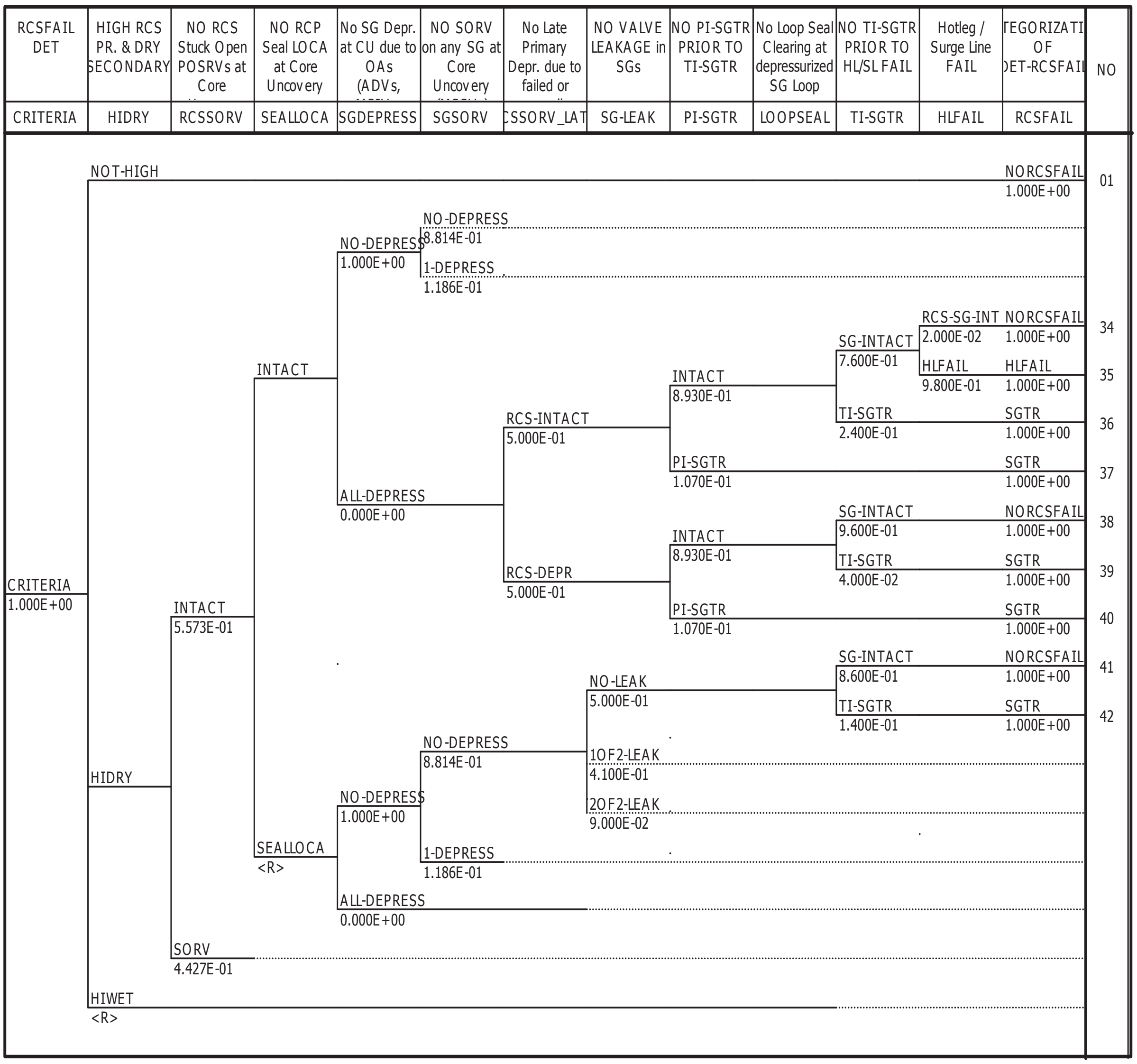


**Figure 19.1-46a RCSFAIL Decomposition Event Tree (3/6)**

Added

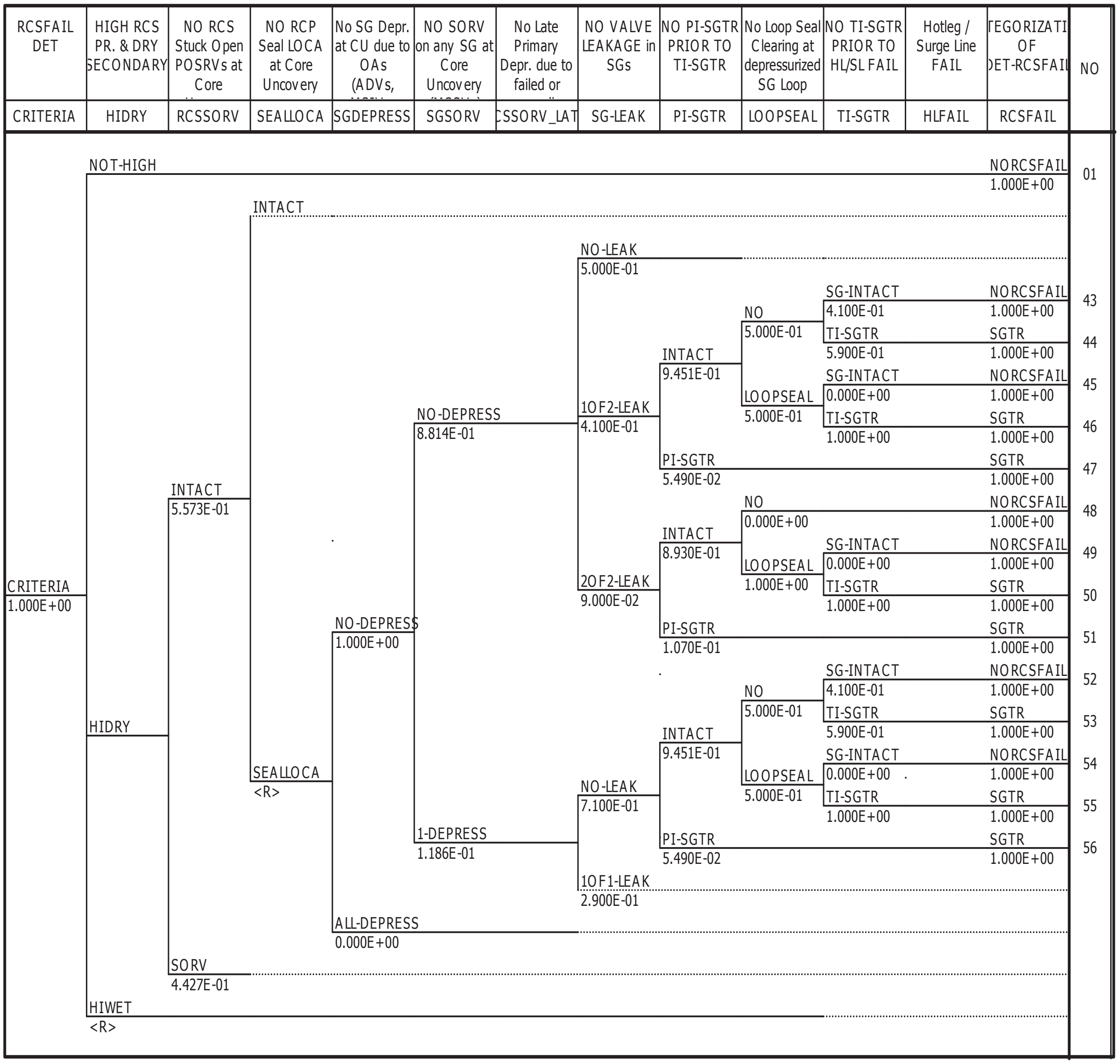


**Figure 19.1-46a RCSFAIL Decomposition Event Tree (4/6)**

Added

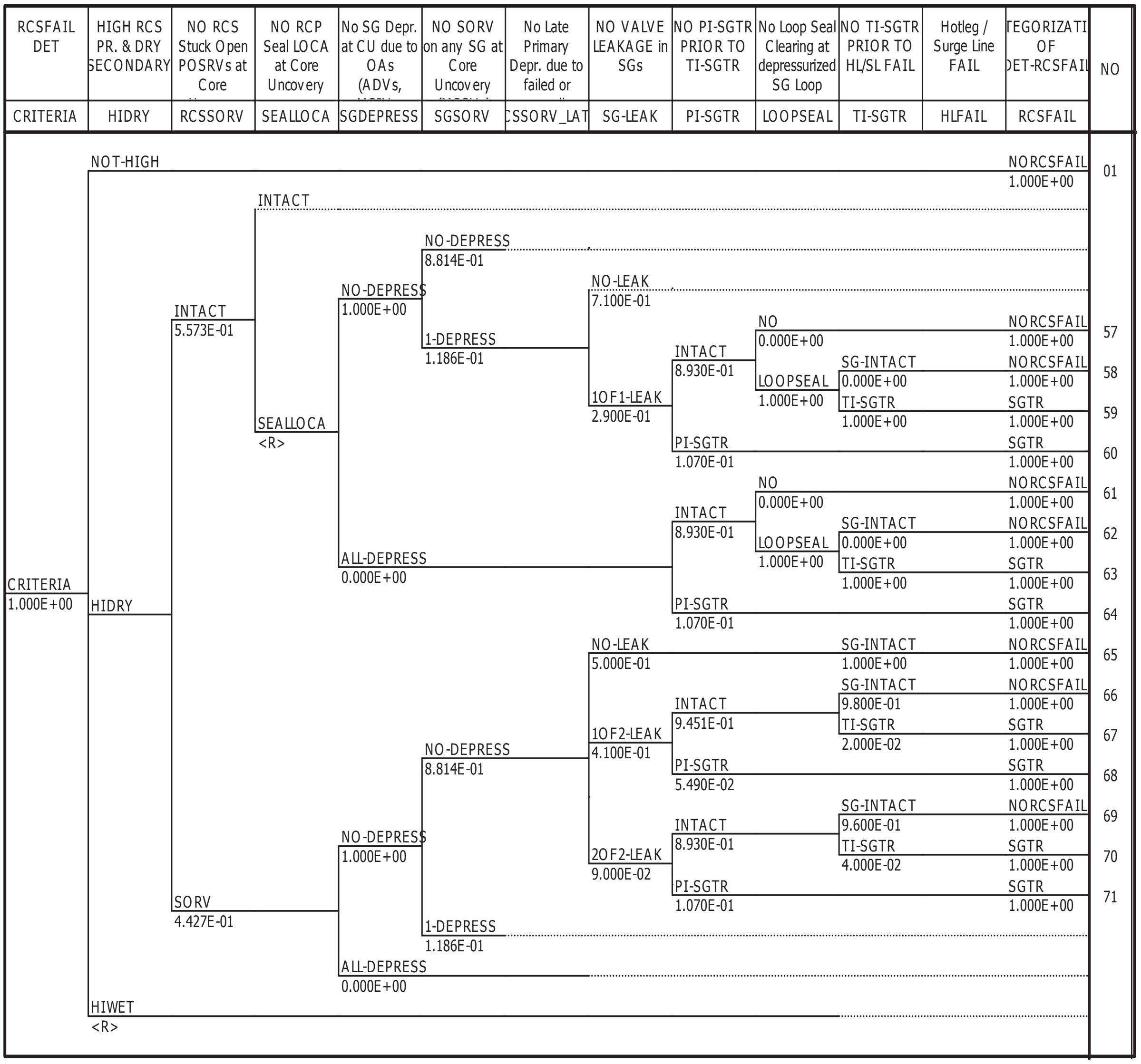


**Figure 19.1-46a RCSFAIL Decomposition Event Tree (5/6)**

Added

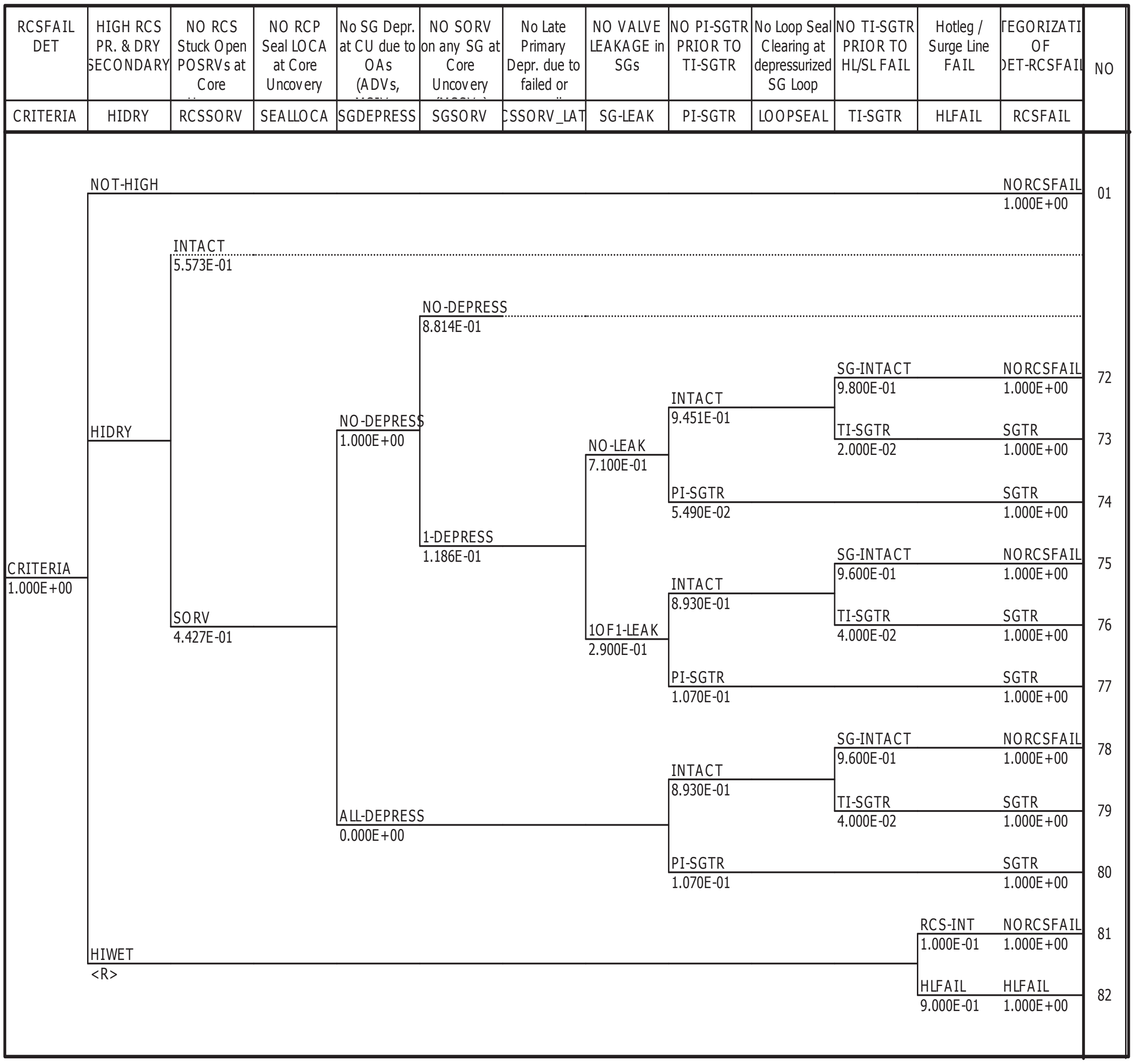


**Figure 19.1-46a RCSFAIL Decomposition Event Tree (6/6)**

Added

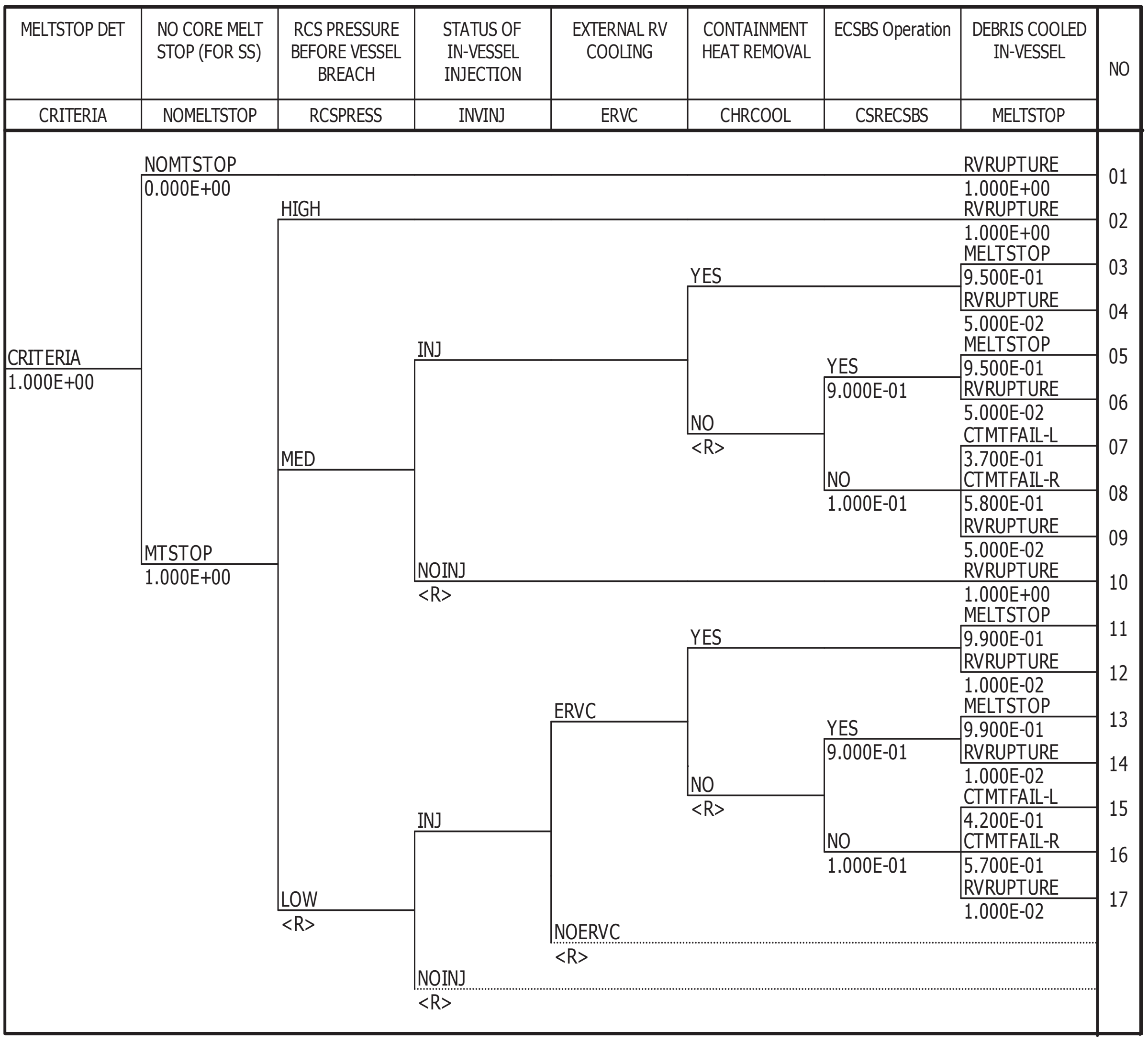


**Figure 19.1-46b MELTSTOP Decomposition Event Tree (1/2)**

Added

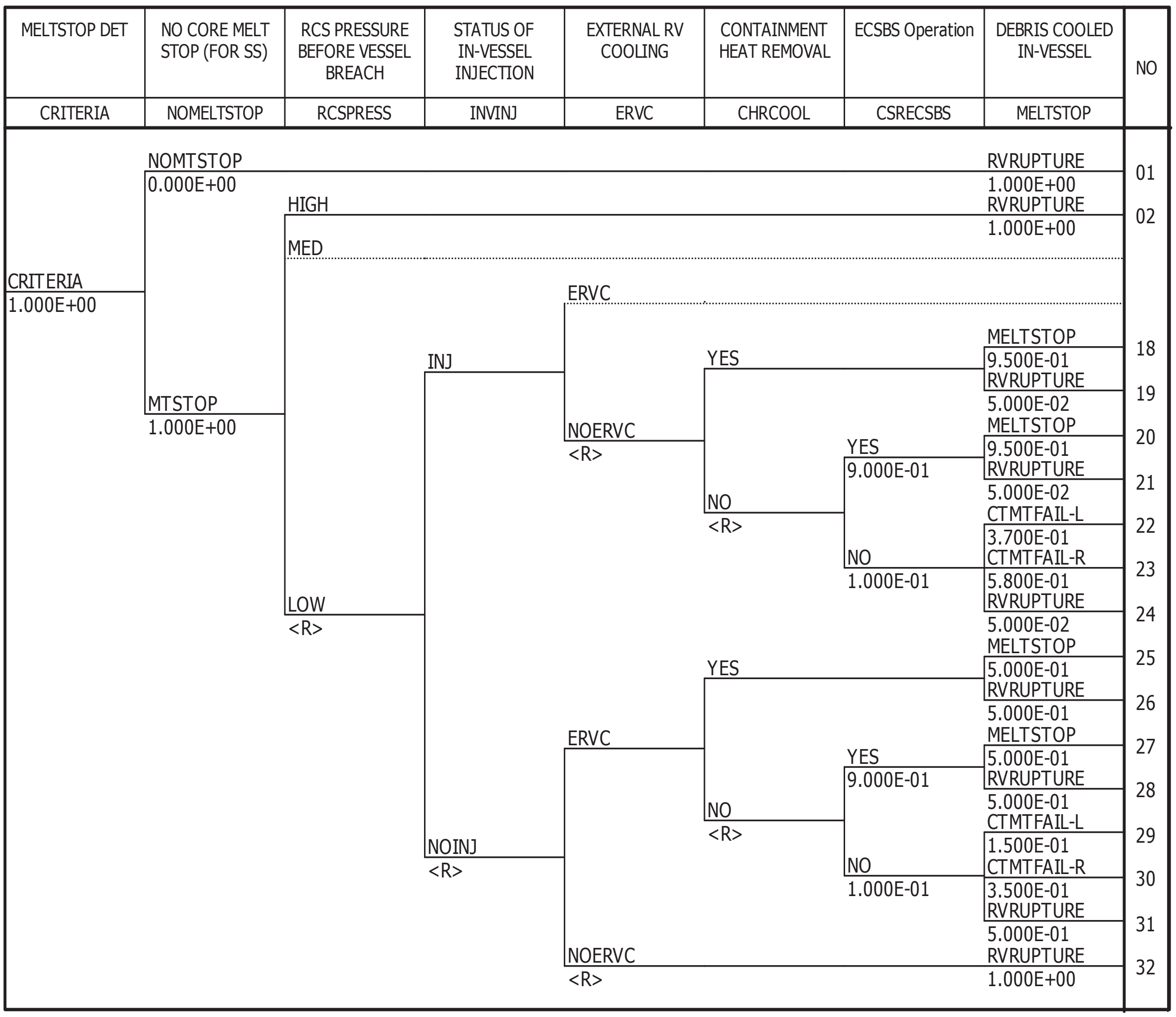


**Figure 19.1-46b MELTSTOP Decomposition Event Tree (2/2)**

Added

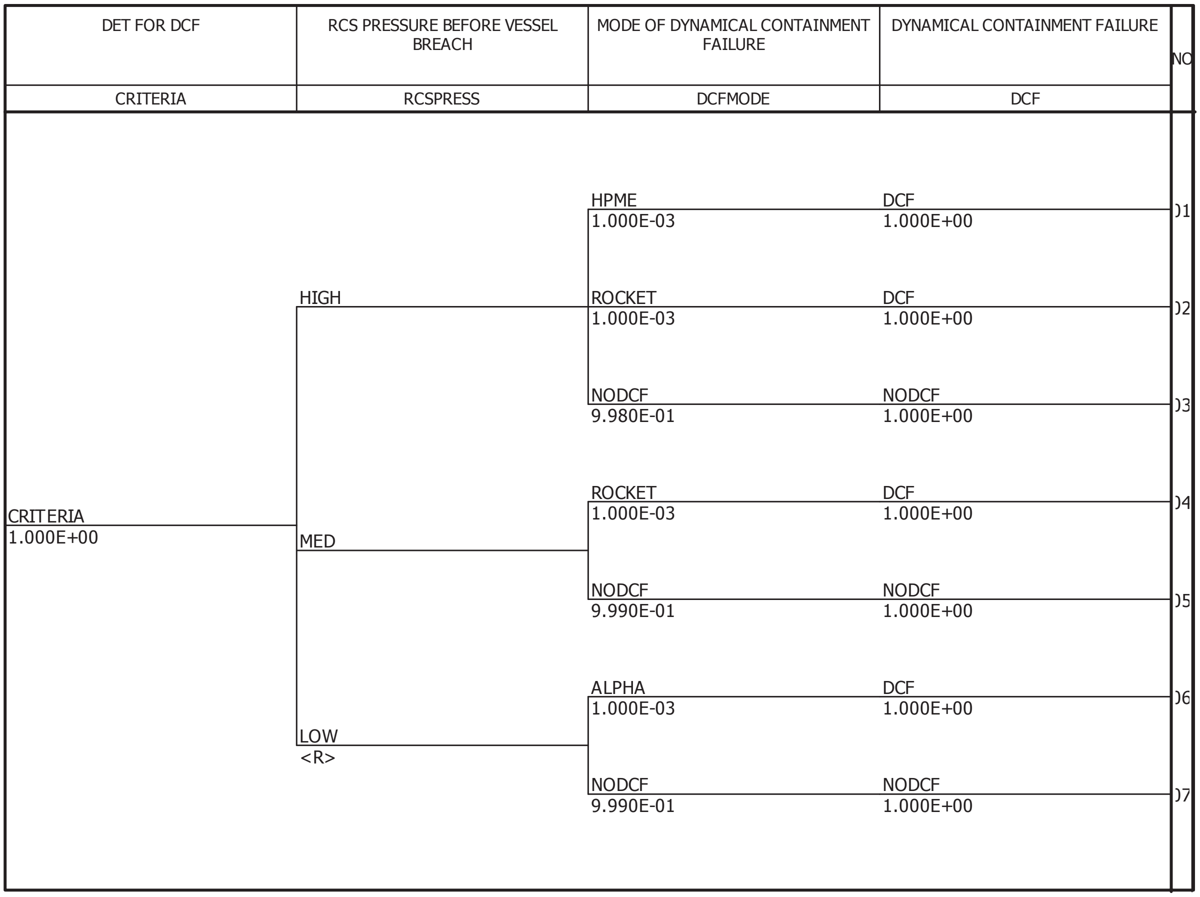


**Figure 19.1-46c DCF Decomposition Event Tree**

Added

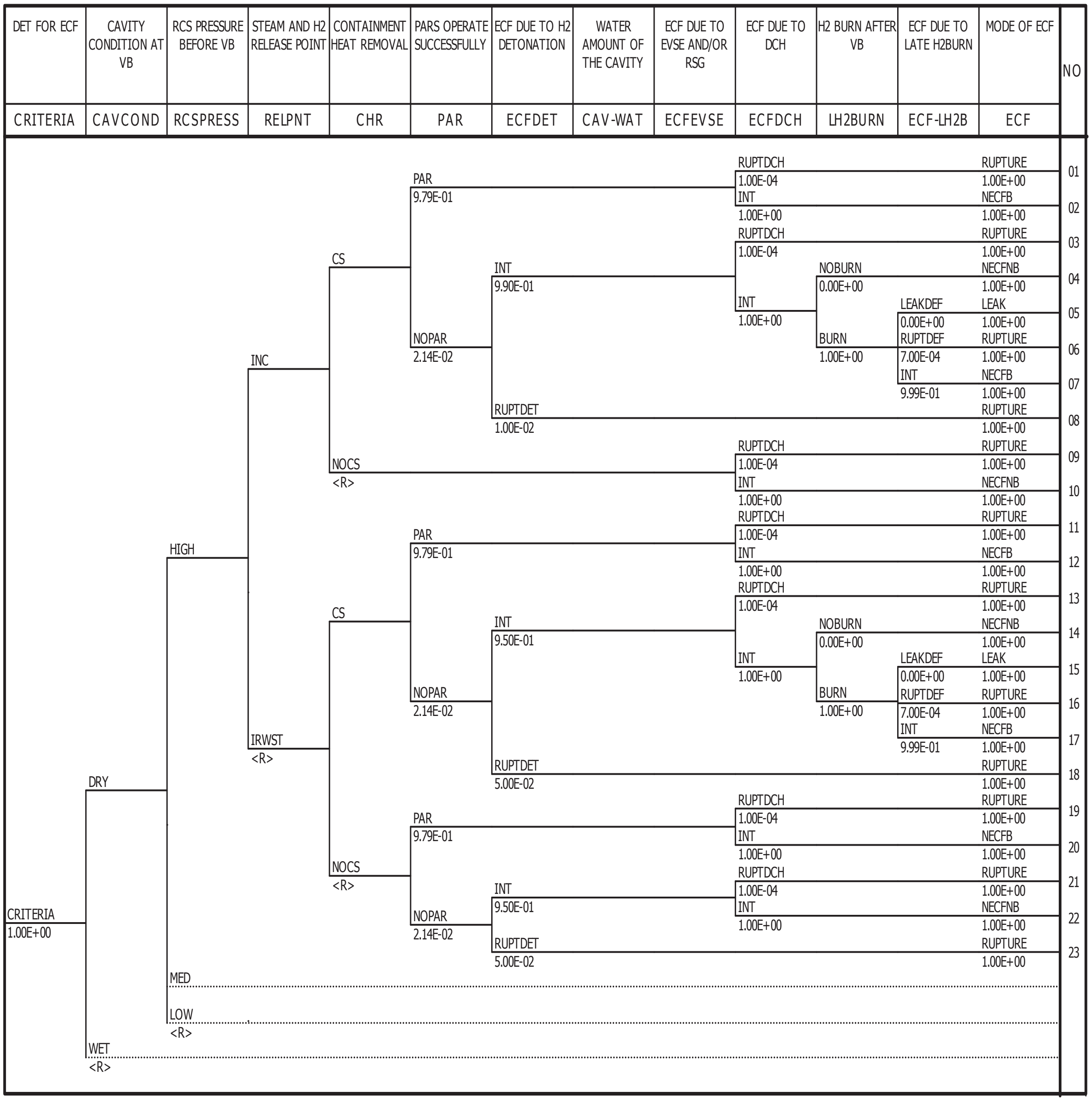


Figure 19.1-46d ECF Decomposition Event Tree (1/7)

Added

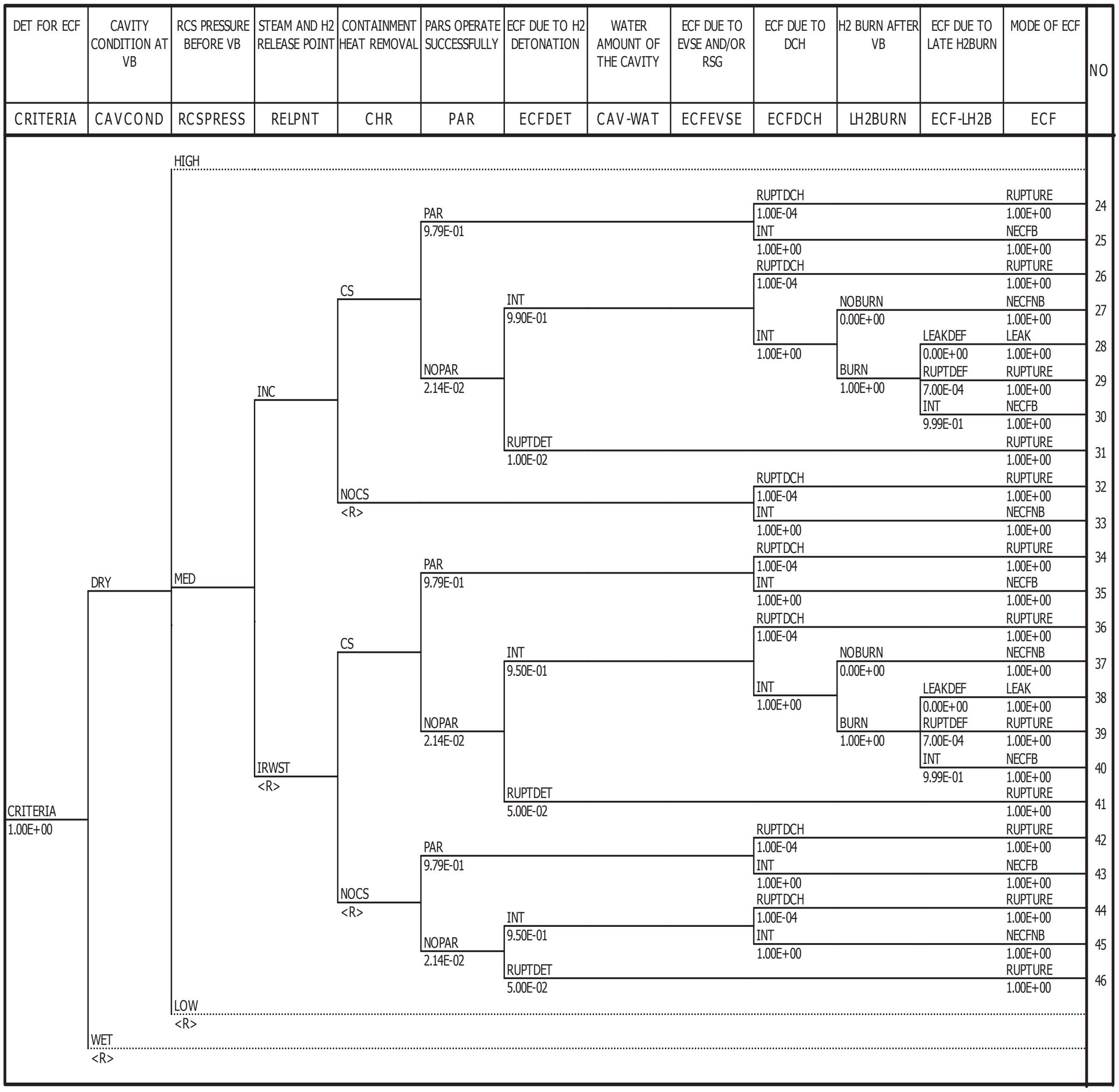


Figure 19.1-46d ECF Decomposition Event Tree (2/7)

Added

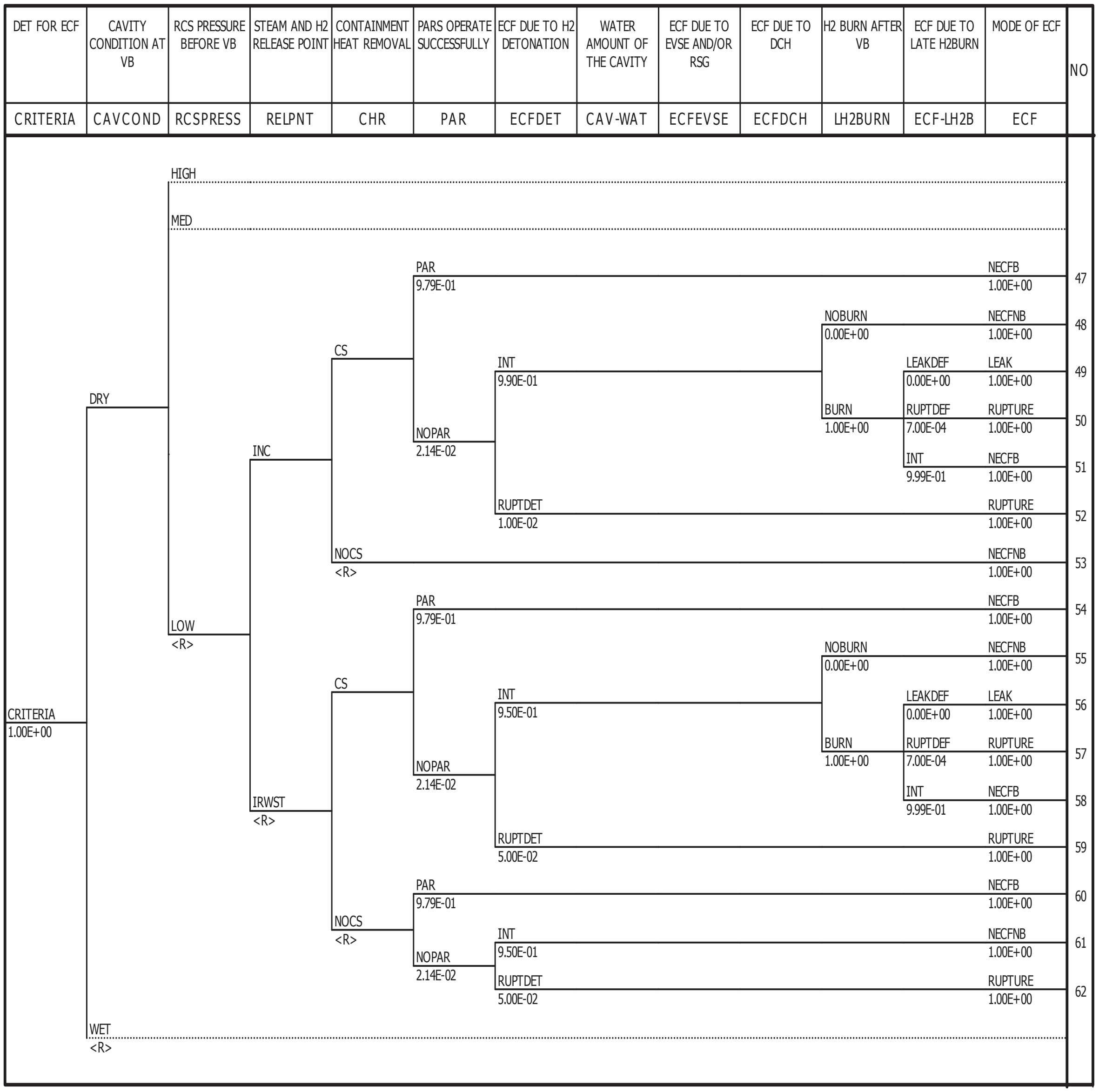


**Figure 19.1-46d ECF Decomposition Event Tree (3/7)**

Added

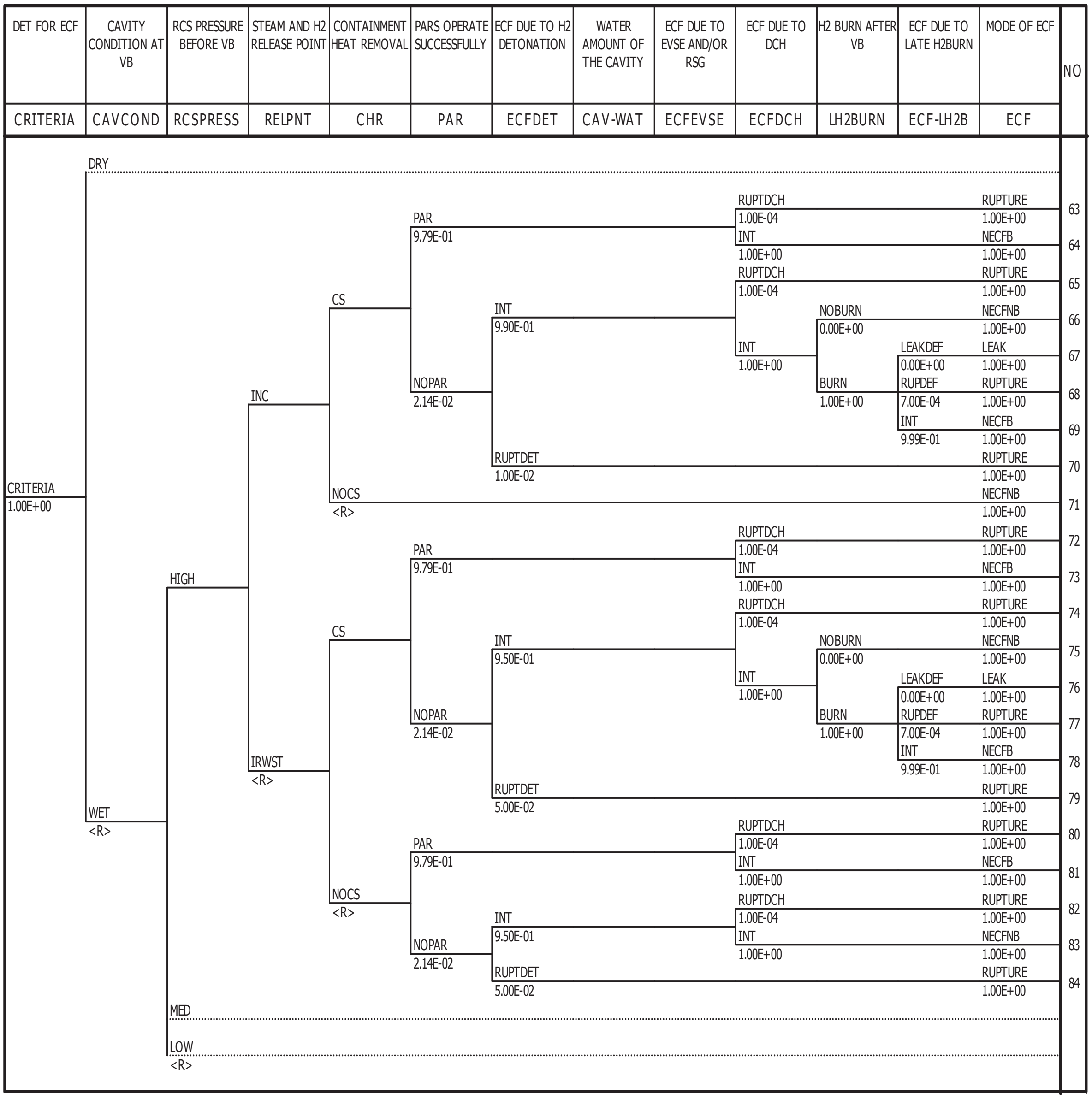


**Figure 19.1-46d ECF Decomposition Event Tree (4/7)**

Added

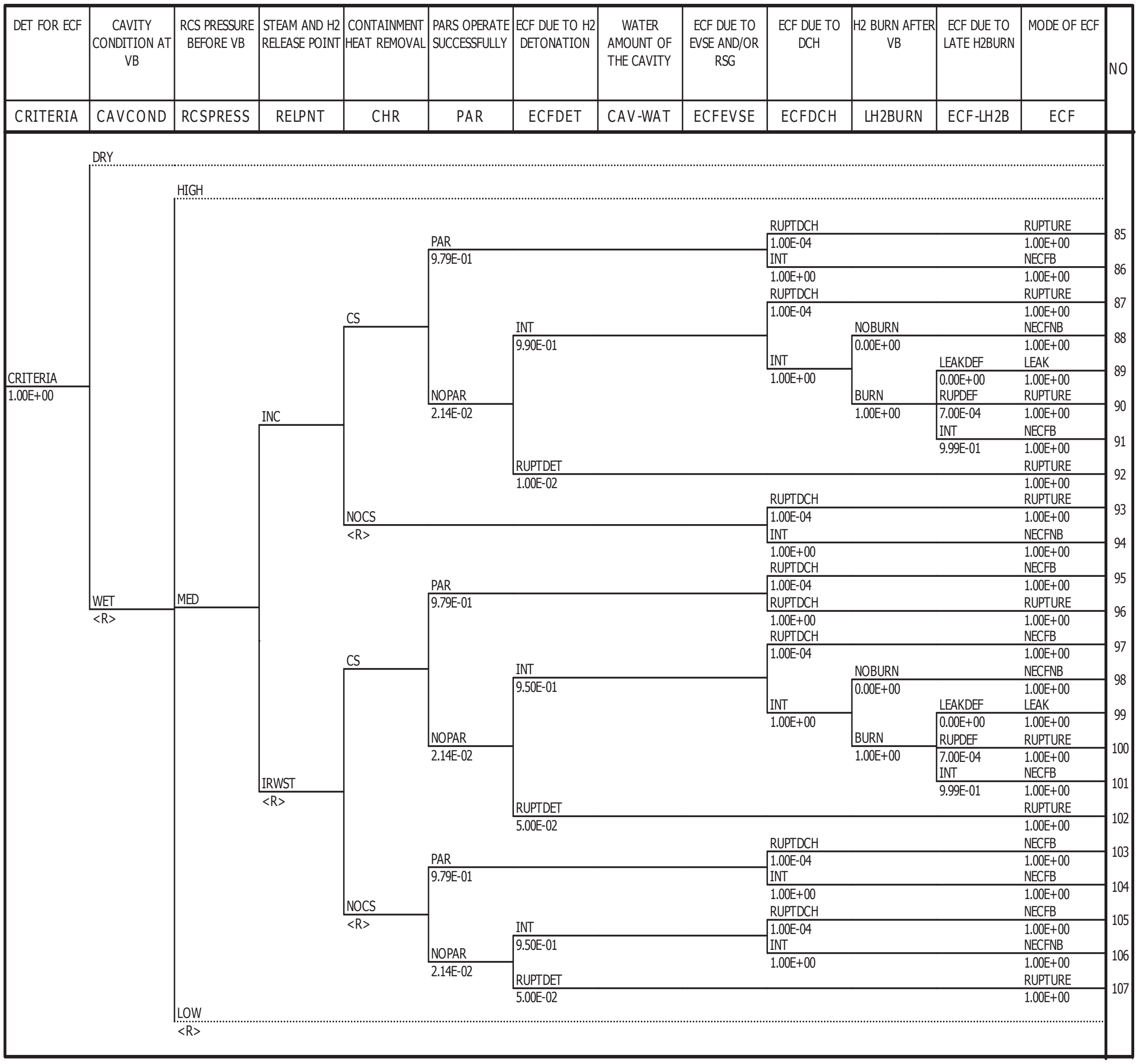


Figure 19.1-46d ECF Decomposition Event Tree (5/7)

Added

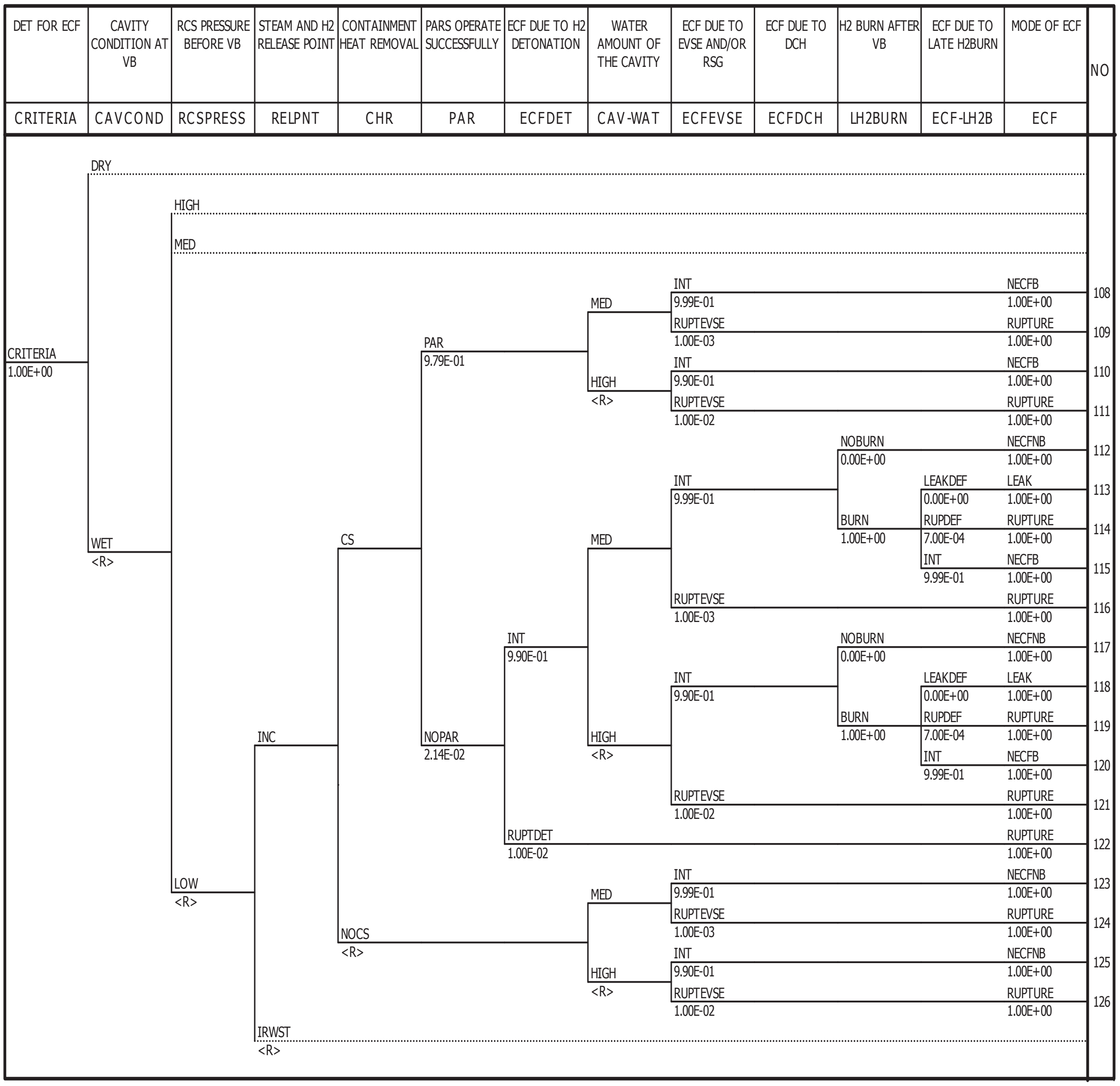


**Figure 19.1-46d ECF Decomposition Event Tree (6/7)**

Added

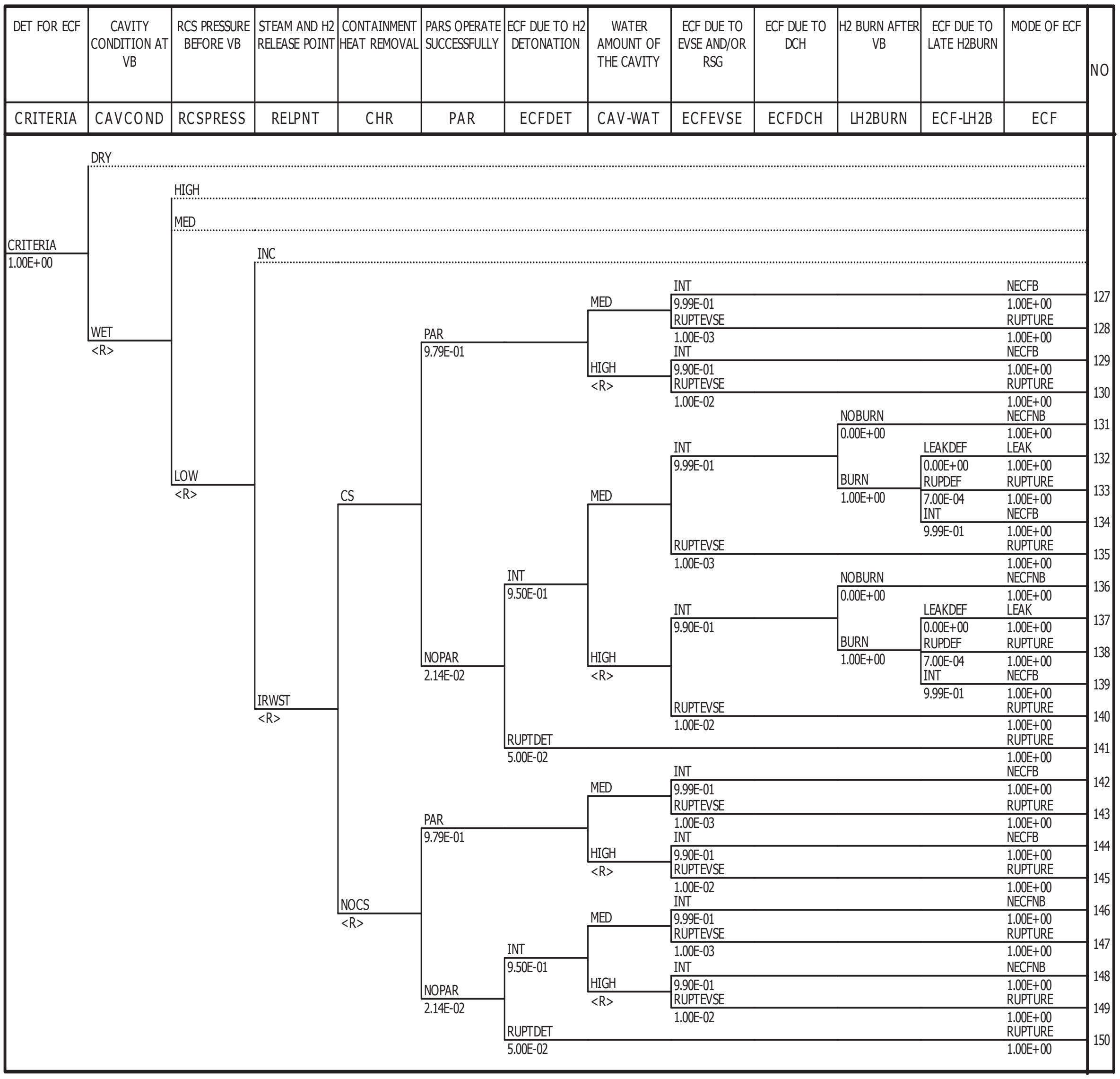


**Figure 19.1-46d ECF Decomposition Event Tree (7/7)**

Added

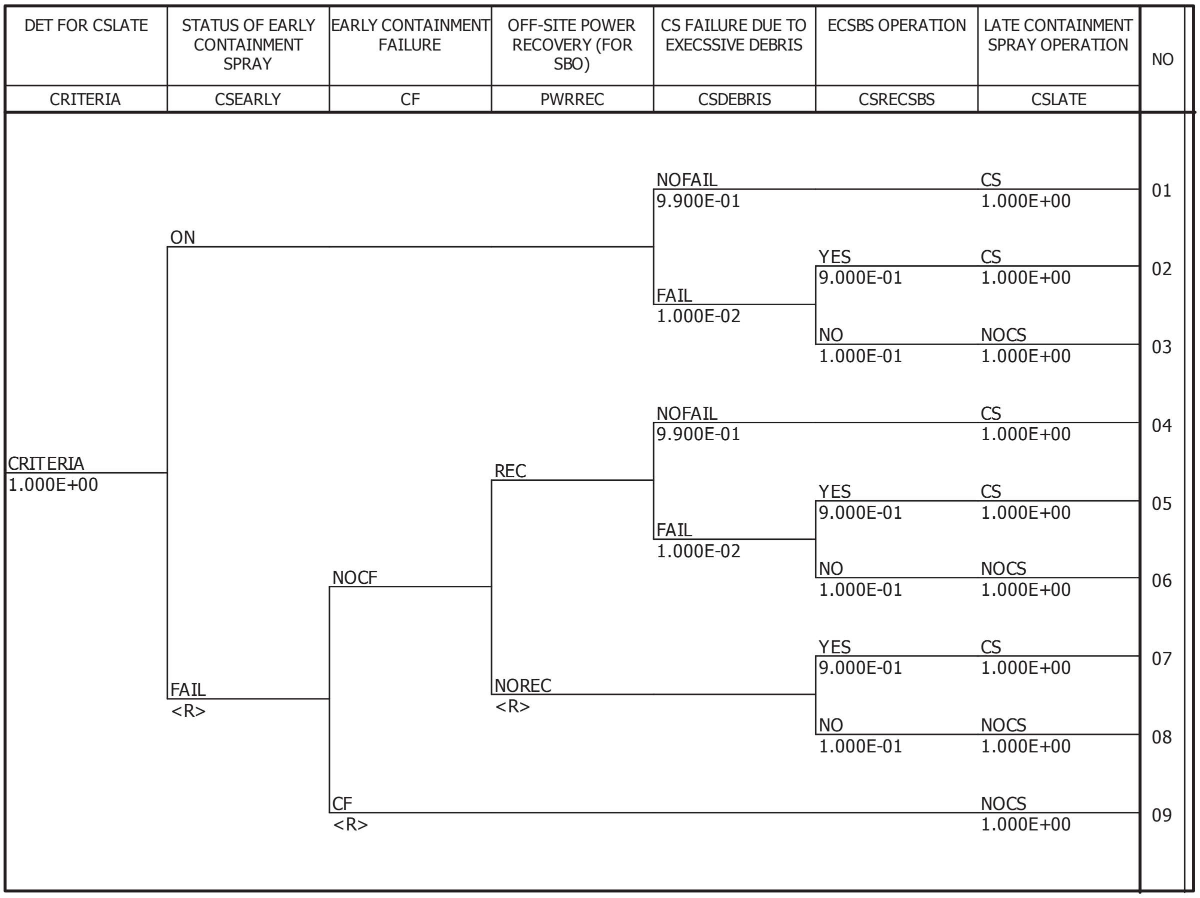


**Figure 19.1-46e CSLATE Decomposition Event Tree**

Added

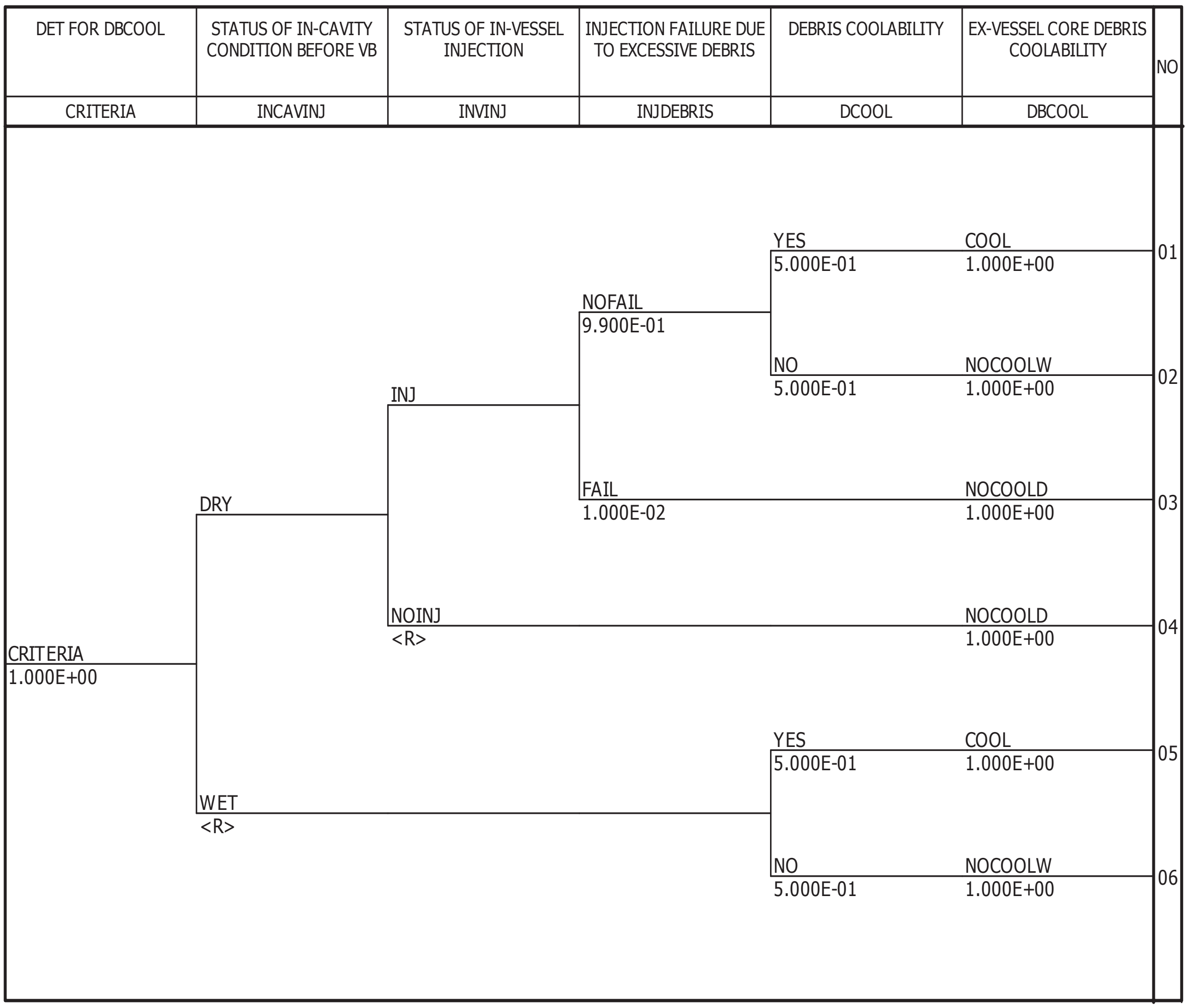


**Figure 19.1-46f DBCOOL Decomposition Event Tree**

Added

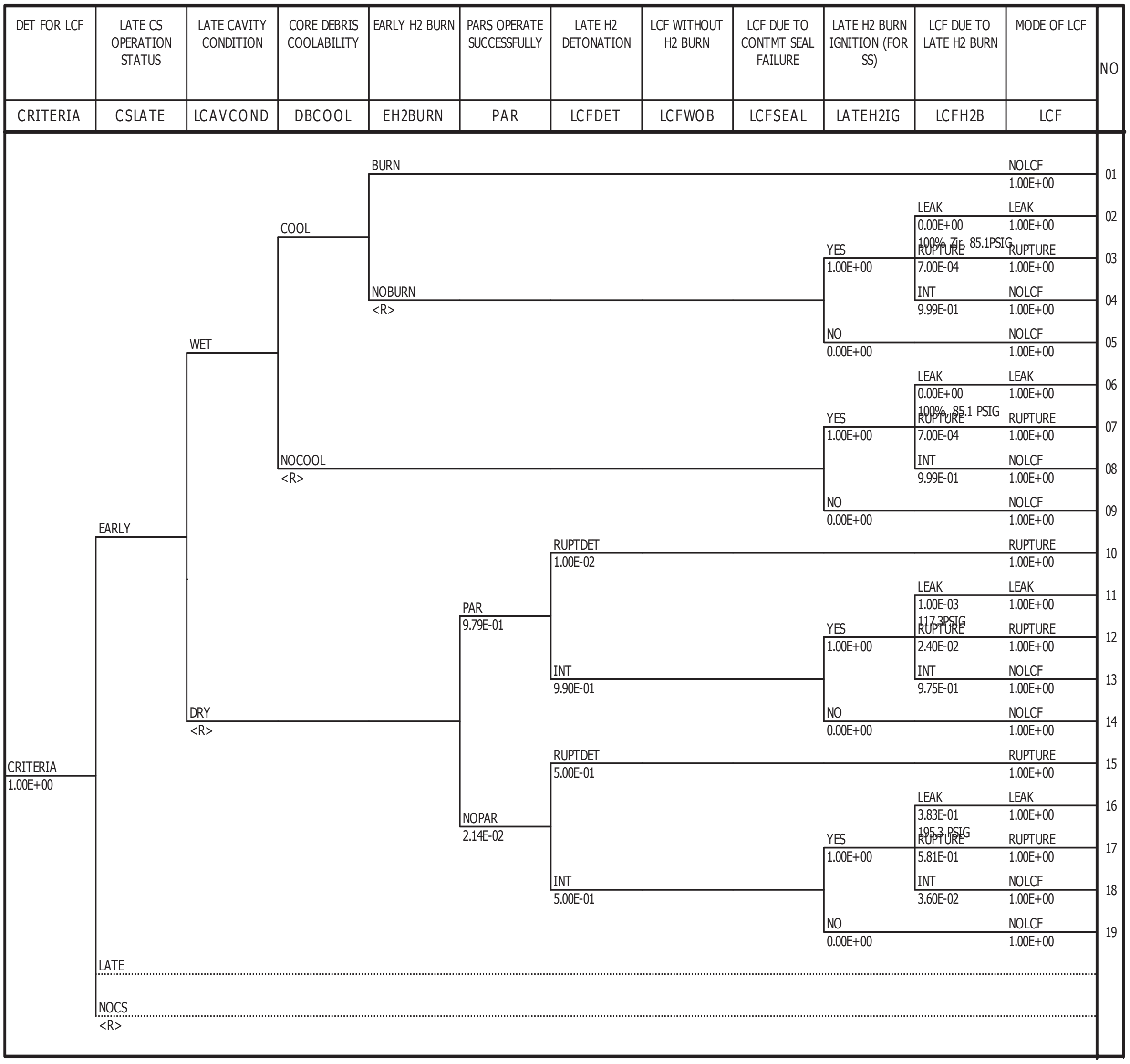


**Figure 19.1-46g LCF Decomposition Event Tree (1/3)**

Added

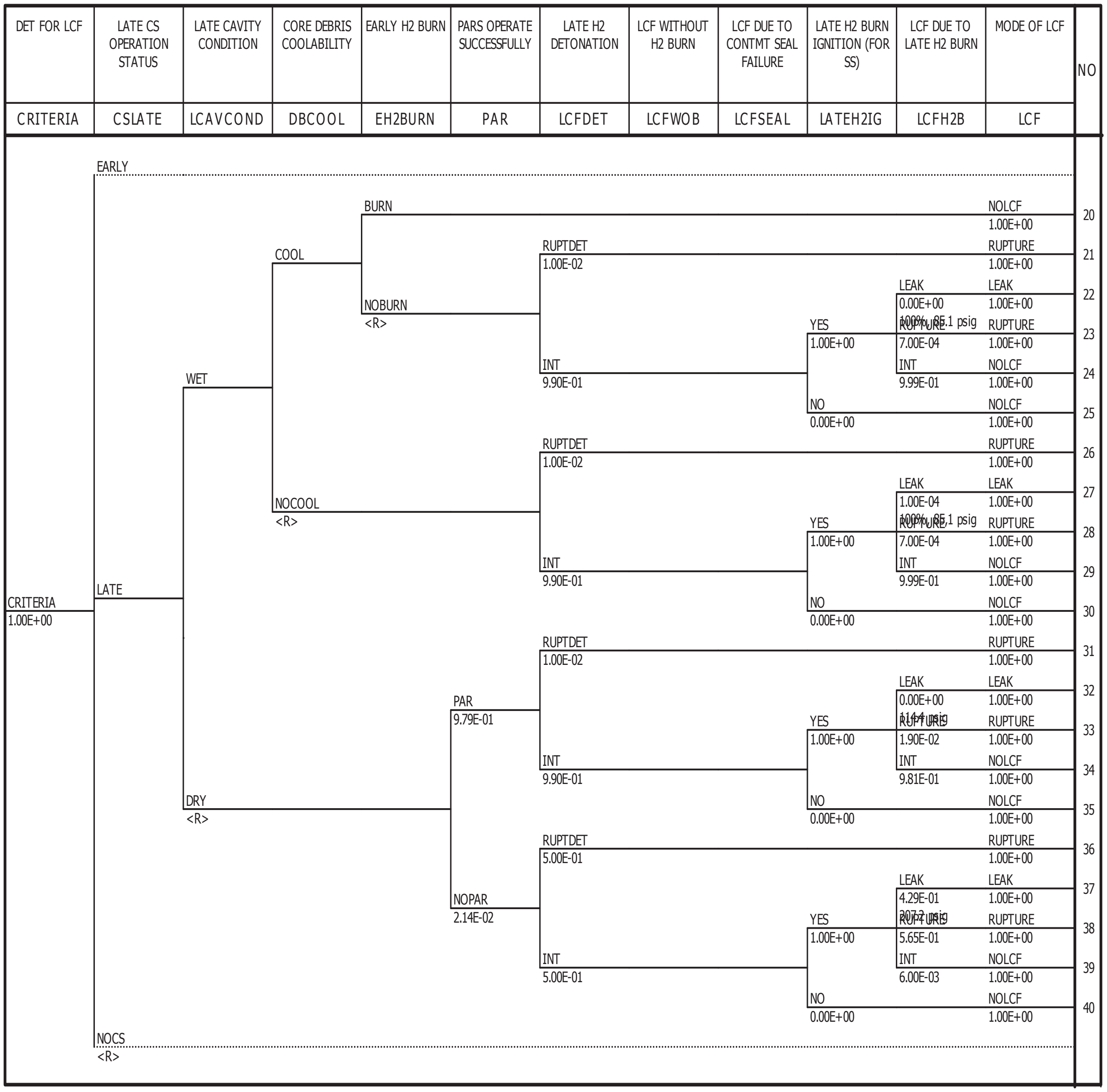


Figure 19.1-46g LCF Decomposition Event Tree (2/3)

Added

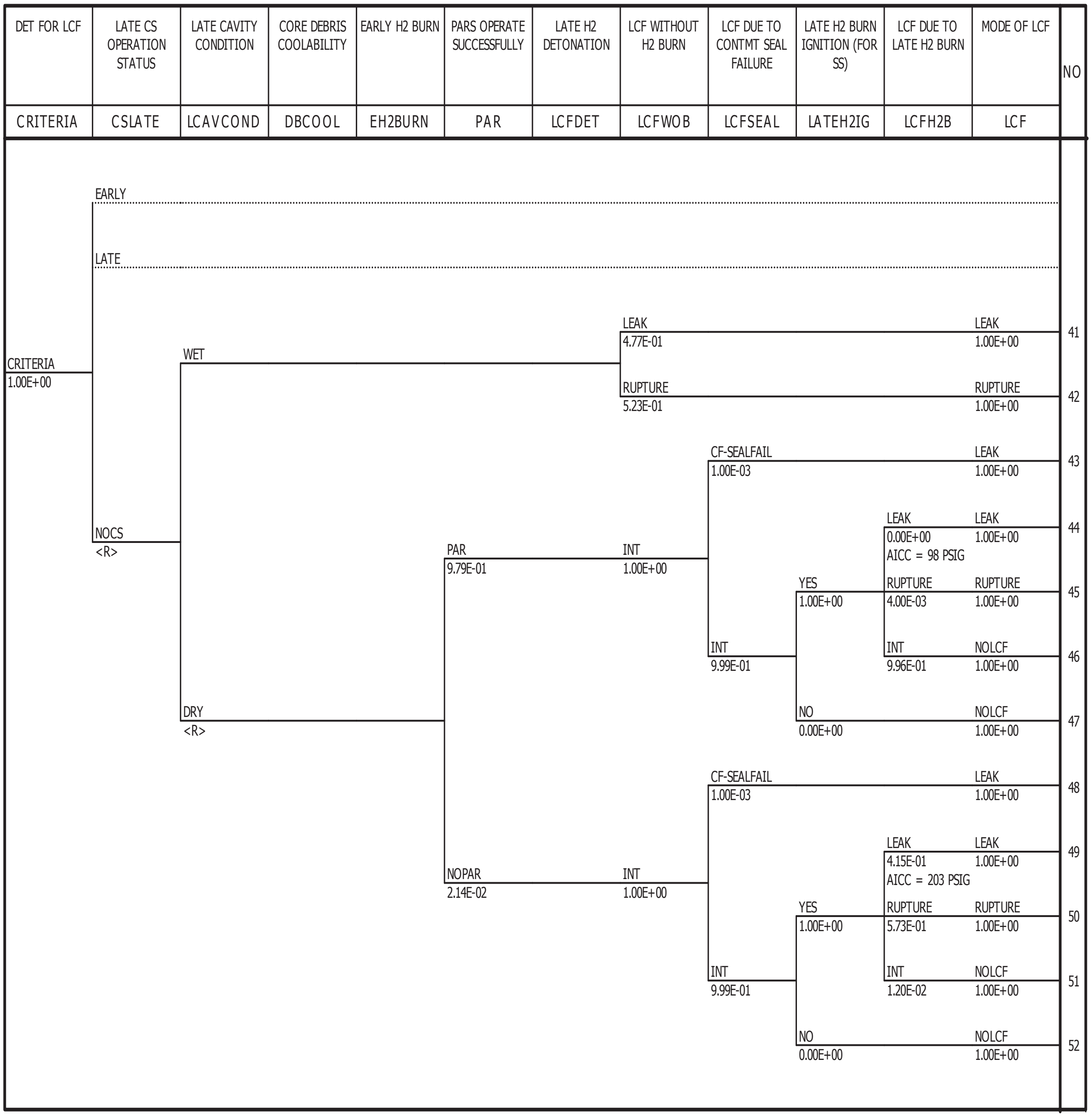


**Figure 19.1-46g LCF Decomposition Event Tree (3/3)**

Added

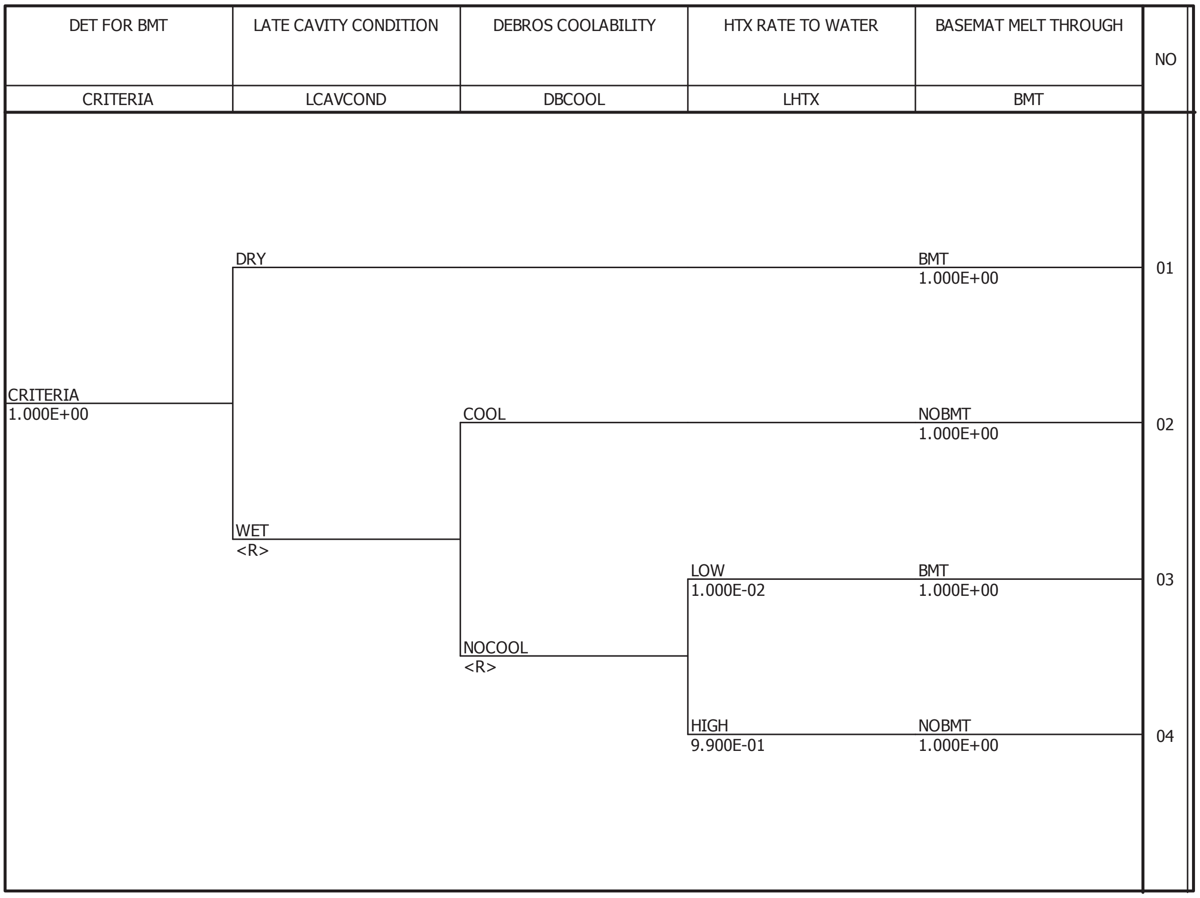


**Figure 19.1-46h BMT Decomposition Event Tree**

Added

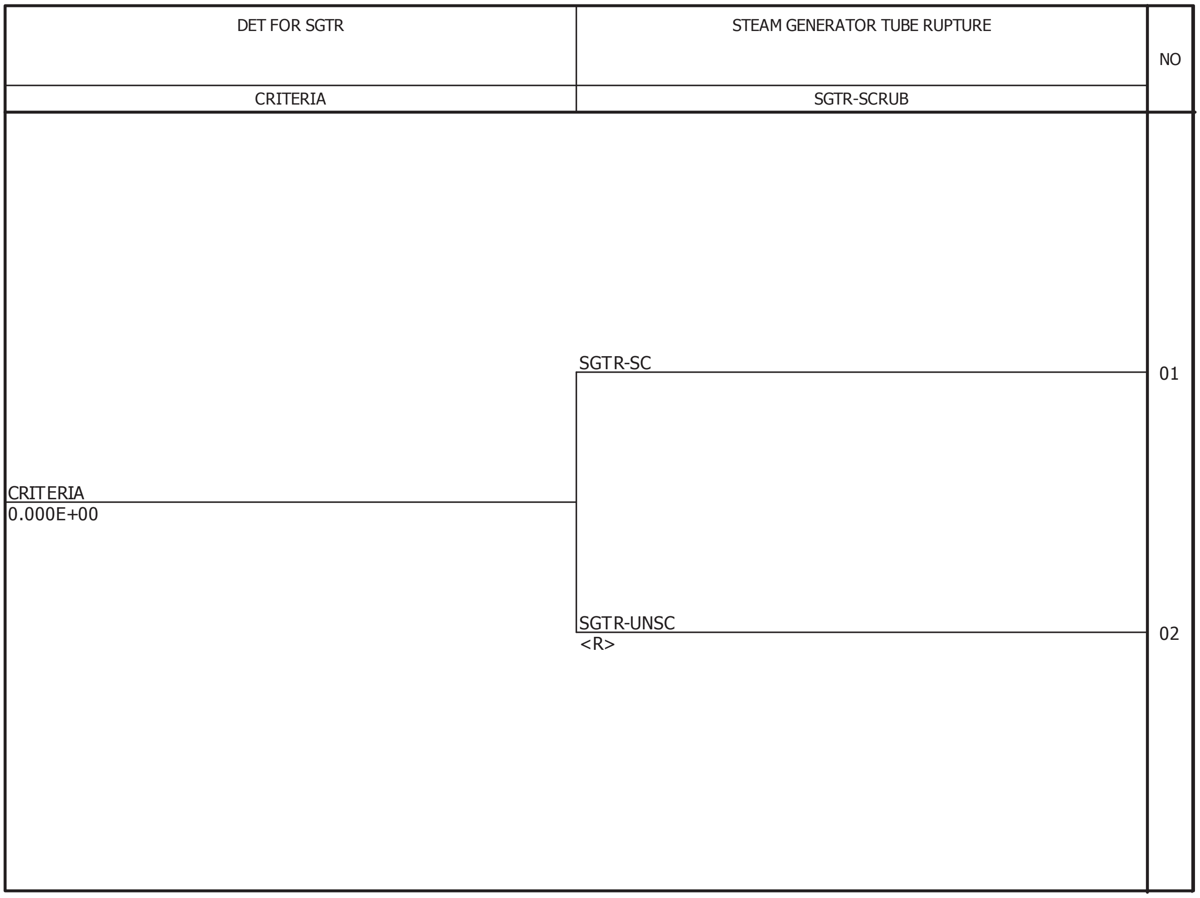


**Figure 19.1-46i SGTR Decomposition Event Tree**

Added

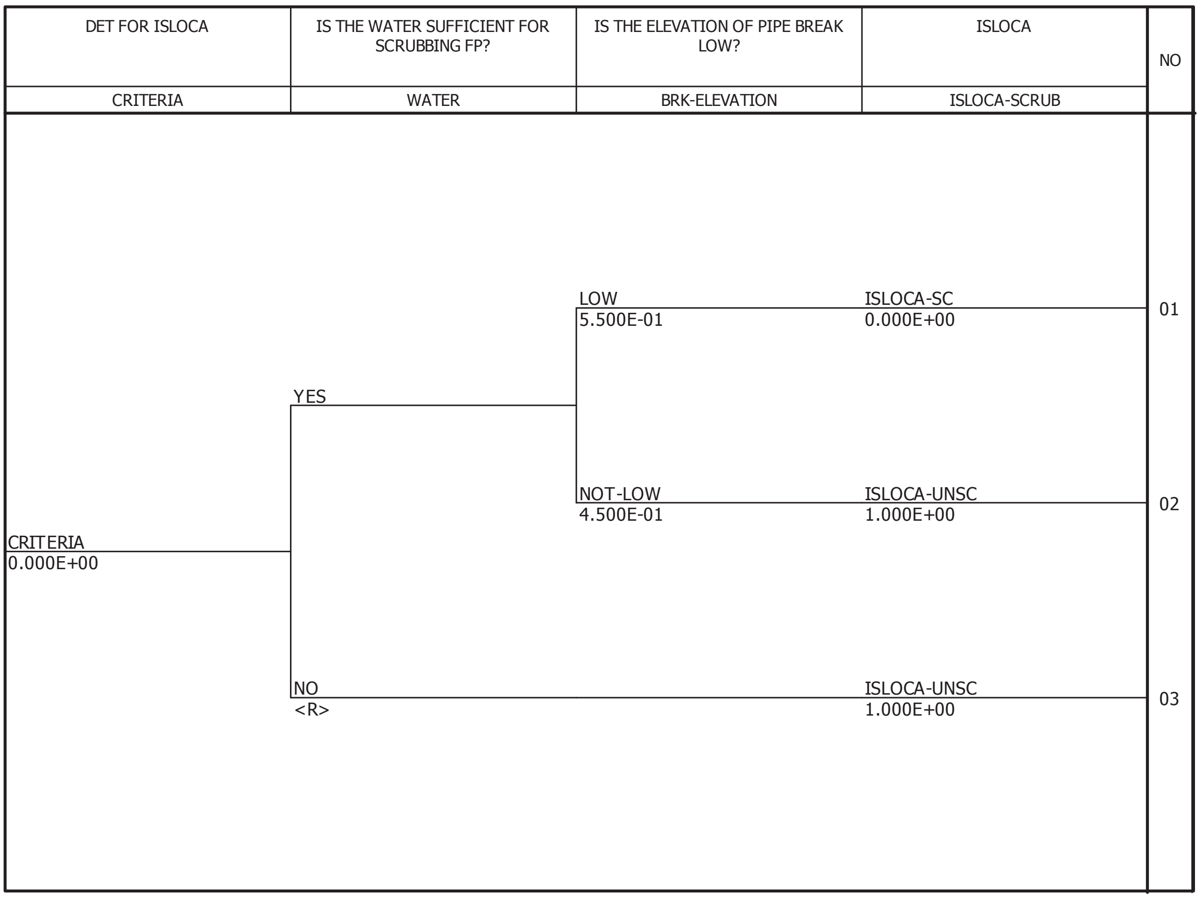


**Figure 19.1-46j ISLOCA Decomposition Event Tree**

Added

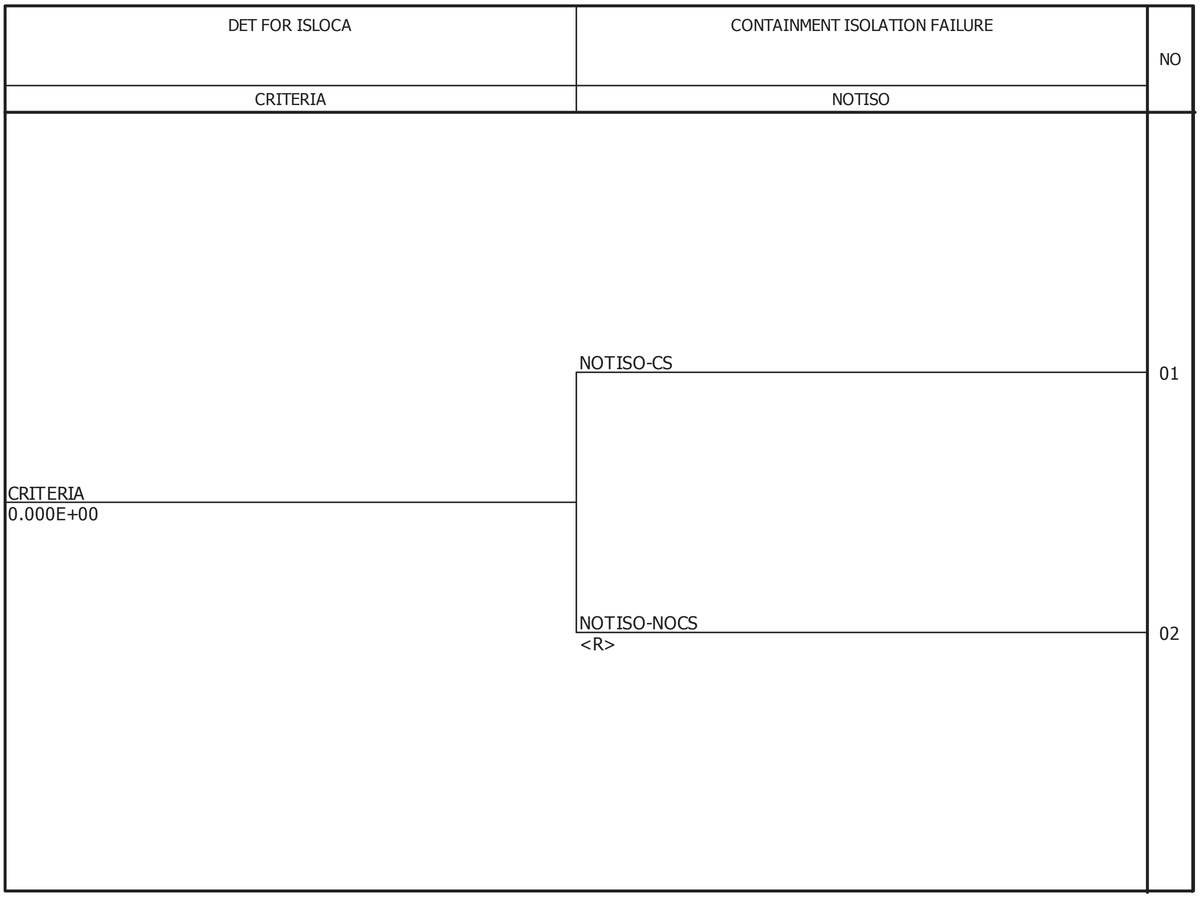


**Figure 19.1-46k ISLOCA Decomposition Event Tree**

Added

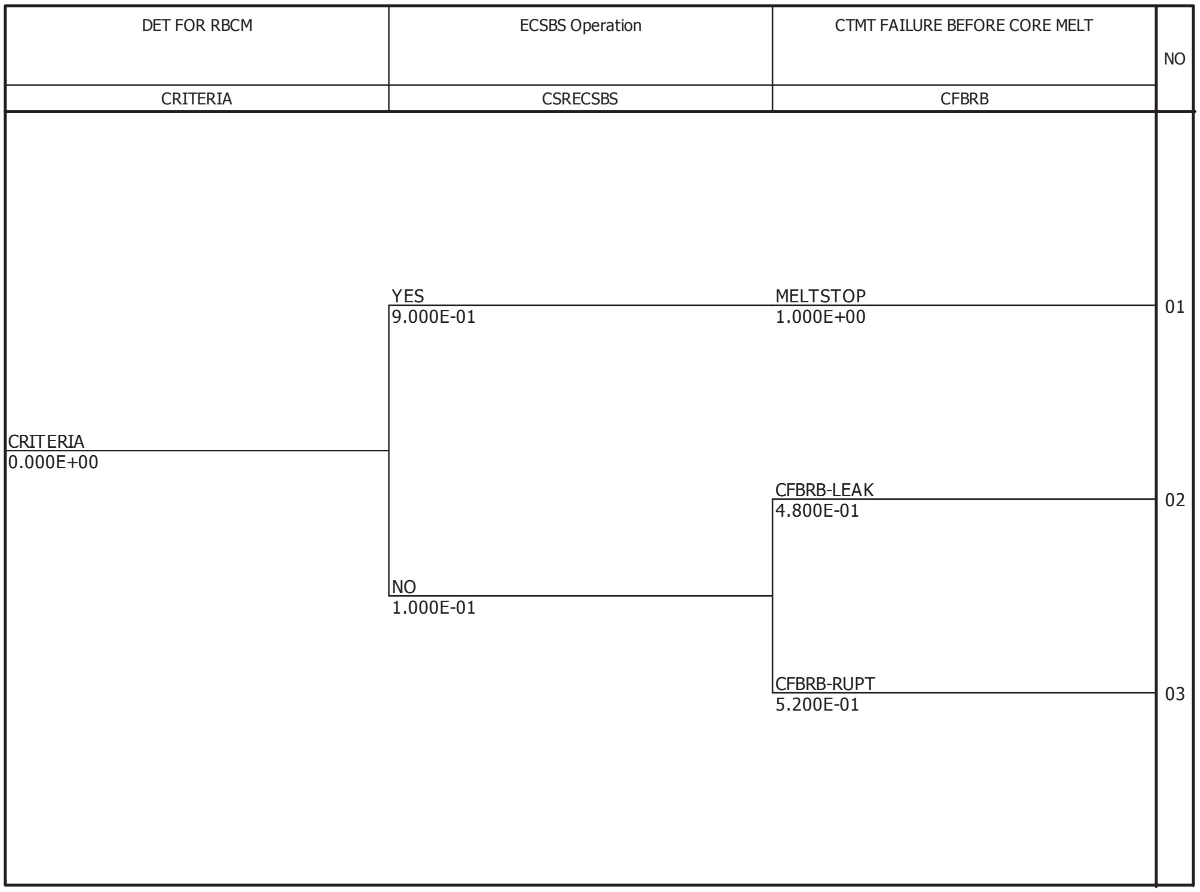


**Figure 19.1-46l RBCM Decomposition Event Tree**

# REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**APR1400 Design Certification**

**Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD**

**Docket No. 52-046**

| | |
|---|---|
| **RAI No.:** | **432-8377** |
| **SRP Section:** | **SRP 19** |
| **Application Section:** | **19.1** |
| **Date of RAI Issue:** | **03/08/2016** |

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## Question No. 19-55

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

APR1400 design control document (DCD) Rev. 0, Section 19.1.4.2.1.2.1, states that "The containment event trees [CETs] are shown in Figure 19.1-42 through Figure 19.1-46." Of these, the DCD does not describe Figures 19.1-43 through -46 showing CETs for SGTR, ISLOCA, Containment Isolation Failure, and Containment Failure before Vessel Breach, respectively, nor does it describe how the top events of these CETs were evaluated. Update the DCD describing these CETs and how their top events were evaluated.

## Response – (Rev. 1)

Each PDS end point represents a unique accident progression staring point with respect to the CET. In practice, there will be many commonalities for most accident sequences, except for those PDSs such as containment bypass and containment isolation failure. For example, core damage sequences in which containment is successfully isolated and not bypassed would have different modeling approaches with respect to containment challenges compared with those sequences in which containment bypassed initially. Therefore, to model containment responses for most accident sequences (i.e., PDS 8 through 108), a general CET is developed. Special CETs are developed for the containment bypass sequences (i.e., PDS 1 through 4), the containment isolation failure (i.e., PDS 5 through 6) and the containment failure before core melt (i.e., PDS 7).

The description of general CET and special CETs are added as described in Subsection 19.1.4.2.1.2.2 and 19.1.4.2.1.2.3 in DCD 19.1 (See Attachment).

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**Impact on DCD**

DCD Subsection 19.1.4.2.1.2.2 and 19.1.4.2.1.2.3 are added in  DCD 19.1.4.2.1.2 as shown in the Attachment.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on the Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

There is no impact on any Technical, Topical, or Environmental Report.

This issue is of primary concern to BWR plants because of the drywell design. For completeness of the PRA, this mode of failure was considered in the APR1400 PRA even though the pathways for debris transport out of the reactor cavity are to interior containment building compartments away from the containment wall. The probability of this failure mode was assigned a negligible value.

i. Failure of Containment Building Penetrations

Failure of containment building penetrations (electrical, fluid, equipment hatch, personnel hatch, etc.) was explicitly evaluated in the analysis of the containment overpressure capacity and found to be significantly less important than overpressure failure of the cylinder wall. Temperature-induced penetration failures were treated in the APR1400 PRA.

To model containment responses for most accident sequences, a general CET is developed. Special CETs were developed for the containment bypass, containment isolation failure, and containment failure before RV breach. These CETs properly considered all pertinent containment failure modes identified for the APR1400 containment. The important phenomena that can affect the containment failure modes and the source terms are also addressed in the CETs. The questions and important events that are used in significant references (e.g., NUREG-1150 and previous Level 2 PRAs for other plants) are reviewed and included in the APR1400 PRA CETs.

Insert A in next page

~~The containment event trees are shown in Figure 19.1-42 through Figure 19.1-46.~~

19.1.4.2.1.2.~~3~~ 4

~~19.1.4.2.1.2.2~~ Containment Ultimate Pressure Capacity Analysis

In order to evaluate the likelihood of containment failure for various accident progression phenomena, it is necessary to determine a realistic pressure at which the containment would fail. In nuclear power plants, the containment design failure pressure is 2 to 3 times less than the realistic, as-built failure pressure. Therefore, a best-estimate assessment of the APR1400 containment was performed. This section summarizes the evaluation and results.

19.1.4.2.1.2.2

~~19.1.4.2.1.2.3.1~~ General Containment Event Tree

a. Mode of RCS Failure Before Vessel Breach (RCSFAIL)

This event determines whether the elevated RCS temperature and pressure following core uncovery result in failure of the upper RCS pressure boundary prior to failure of the reactor vessel lower head. Three possibilities are considered:

1) The primary system remains intact prior to vessel breach
2) Severe Accident Induced Steam generator tube rupture
3) Induced Rupture of a hot leg or the pressurizer surge line

Each of the possible branch pathways for this event has an important impact on accident progression. Hot leg failures are likely to be of sufficient size to cause depressurization of the RCS prior to vessel failure. This will greatly reduce the possibility of energetic events at vessel failure (e.g., Direct Containment Heating : DCH), which may cause containment failure. Failure of one or more steam generator tubes can result in the bypass of the containment if a secondary relief/safety valve opens or if there is significant leakage past the main steam isolation valves (MSIVs). Large amount of fission products can be released through the direct path from the RCS to the environment without any scrubbing. This fission product release can affect consequences significantly.

b. In-Vessel Core Melt Arrest (MELTSTOP)

This question determines whether the damaged core can be cooled in-vessel, thereby terminating the accident progression before reactor vessel rupture. Four possibilities are considered:

1) The RPV lower head fails prior to the containment failure
2) Arrest of core melt progression before reactor vessel rupture

A(2/11)

3) Containment failure before vessel rupture (Leak)

4) Containment failure before vessel rupture (Rupture)

The core melt can be arrested and the damaged core can be safely and continually cooled, in the reactor vessel by the introduction of cooling water into the reactor vessel or the reactor cavity. There are two probable approaches for in-vessel core melt retention; injection of large amount of water (1) into the reactor vessel to completely submerge the damaged core or (2) into the reactor cavity to completely submerge the reactor lower head.

The safety injection system (SIS) can deliver sufficient water into the reactor vessel to cool the core when the intact core geometry is maintained or the core debris configuration is favorable for cooling. Once the core configuration becomes less favorable for cooling (e.g. after loss of original configuration and generation of obstacles in the core), substantially higher injection flow rates (several thousand gpm) may not be effective to cool the debris in-vessel because of low heat transfer rate from the core material.

There are two probable scenarios in which the core may be damaged in spite of the injection available. The first is a sequence in which the injection flow is insufficient to prevent core damage, as defined by success criteria in the Level 1 analysis for limiting peak core temperature to be less than 1800°F. The second is a sequence in which there is no coolant injection prior to core uncovery and incipient core damage, but some form of injection is recovered or initiated prior to vessel failure. In grouping the PDS ET sequences into PDSs, the safety injection flow is one of the grouping parameters.

If the reactor vessel lower head is submerged by water injected into the reactor cavity, the reactor vessel lower head can be maintained to be intact. The core can be cooled in the reactor vessel by cooling of the outer wall of the reactor vessel. All of the severe accident phenomena which occurred outside the reactor vessel and threaten the containment integrity, would be prevented. This type of external reactor vessel cooling strategy has been studied for the implementation to the APR1400 DC design.

A(3/11)

Currently, the system for injecting water into the reactor cavity is considered one train of SCS and a part of CVCS.

When the vessel failure is prevented by effectively cooling the core by in-vessel injection or external reactor vessel cooling, the containment may eventually fail due to steam caused over-pressurization if the containment heat removal is lost.

If core melt is arrested before vessel failure and the containment heat removal is available, only limited hydrogen production would be expected and containment over-pressurization would be limited. DCH would not be a possible threat. As a result, containment failure is extremely unlikely. Furthermore, radionuclide release from the debris would be limited and long term revaporization of radionuclides deposited on RCS surfaces would be largely avoided. Hence, because the containment does not fail and the radionuclide release is limited, the environmental source terms for core-damage sequences that are successfully terminated in-vessel are expected to be very small.

c. Dynamical Containment Failure (DCF)

This event determines whether the very energetic phenomena only depending on the RCS pressure at vessel breach occurs and results in the early containment failure at the time of vessel breach. This event can be included in the next event (Early Containment Failure). For convenience's sake, however, these phenomena are considered separately from Early Containment Failure. Two possibilities are considered:

1) No dynamical containment failure
2) Dynamical containment failure.

In this top event three energetic phenomena are considered:

<u>In-vessel steam explosion or Alpha mode containment failure</u>

A(4/11)

It refers to the scenario whereby an in-vessel steam explosion occurs and the thermal energy of molten corium is converted to the kinetic energy of the water-corium slug. The slug pushes the reactor vessel upper head. If the kinetic energy of the slug is sufficient to subsequently disassemble the reactor vessel upper head and propel the upper head (as a blunt missile) against the containment upper dome, the containment may fail. The consequences of an "alpha" mode containment failure will be a large area containment failure in the containment upper dome.

Rocket Induced Containment Failure

This phenomenon addresses the potential for containment failure due to an in-containment reactor vessel lift-off following the failure of the reactor vessel lower head.

High Pressure Melt Eject Induced Containment Failure by Liner Attack

This phenomenon addresses the potential for a containment failure due to the direct impact of corium particles ejected from the RCS at high pressure. The potential for this type of failure is highly dependent on the RCS pressure at the time of reactor vessel failure. In this analysis, the estimation of the probability values was conservatively based on the RCS pressure at the onset of core damage.

There is a substantial body of evidence to suggest that these energetic phenomena does not represent a credible threat of the containment failure at the vessel failure ( i. e., the probabilities of containment failure resulting from them are negligibly small ).

d. Early Containment Failure (ECF)

This event determines whether a gross failure of containment occurs at or soon after reactor vessel failure. Four possibilities are considered:

1) No early containment failure without hydrogen burn
2) No early containment failure with hydrogen burn

A(5/11)

3) Early Containment Failure (Leak)
4) Early Containment Failure (Rupture)

Early containment failure is defined as containment failure occurring within 2 hours after reactor vessel failure. For most transients this definition is inconsequential and arbitrary since containment failures will either occur prior to or immediately (within seconds to minutes) following vessel breach or very late in the accident sequence. However, this definition affects the fission product release following a post vessel breach hydrogen burn. The 2 hours criterion provides a demarcation between the potential hydrogen atmospheres following vessel breach and consequently their containment failure potential. For sequences defined as early, it is assumed that insufficient core concrete interaction can occur so that the hydrogen contribution due to CCI is small. These results in a maximum hydrogen production during the core melt progression equivalent to 100% oxidation of the active cladding. For late hydrogen burns in the containment, hydrogen produced due to core concrete interaction was assumed to be potentially available for combustion. These results in the potential for a late (> 2 hours following vessel breach) burn involving the hydrogen amount equivalent of up to 150% of the active core cladding. The value greater than 100% implies consumption of all the zircaloy in the core (cladding, grids and guide tubes) and limited amounts of steel.

The phenomena that could potentially contribute to early containment failure are:
1) Hydrogen burn before reactor vessel failure
2) Direct containment heating (DCH)
3) Hydrogen burn after reactor vessel failure
4) Rapid ex-vessel steam generation (RSG) and ex-vessel steam explosion (EVSE)

Two rupture sizes were considered possible for APR1400 DC large dry concrete containment:
1) Rupture (typical break size : 1.0 ft2)
2) Leak (typical break size : 0.1 ft2).

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The major difference between a rupture and a leak is that a rupture is capable of arresting a gradual pressure rise in the containment and depressurizing the containment in less than two hours. A leak would also arrest a gradual pressure buildup, but would not result in containment depressurization within two hours.

The containment structural analysis was performed to identify the ultimate containment strength and the likely failure modes for the APR1400 DC containment building. The result is showed in Section 19.1.4.2.1.2.3

e. Late Containment Heat Removal Recovery Failure (CSLATE)

This event determines if containment heat removal is available late (after vessel breach) in the accident sequence. It is assumed that late over-pressurization can be avoided if containment heat removal is available. In this analysis, the containment spray system and the emergency containment spray backup system (ECSBS) are considered to function for containment spray. The branches for this event are:

1) No late containment spray available
2) Late containment spray available.

For containment heat removal to be available after vessel breach, containment heat removal function should be available early in the accident scenario and the function should be maintained after vessel failure, or the early failed containment heat removal should be recovered. Failure of equipment inside containment is considered to be 100% non-recoverable. For cases where the containment heat removal was unavailable because the operator had failed to initiate containment spray, containment spray would be initiated before containment failure given the available time and indications. In cases that AC power was unavailable, containment spray would be operated if the power is recovered prior to containment failure.

f. Ex-Vessel Debris Coolability (DBCOOL)

A(7/11)

This event determines whether the core debris relocated into the reactor cavity is rapidly quenched by an overlying water pool. Though APR1400 DC has been designed with a large basemat sized and passive cavity flooding system, in this analysis, it is considered that the corium may not be well cooled by an overlying water pool. Three possibilities are considered:

1) Ex-vessel debris not cooled without overlying water pool
2) Ex-vessel debris not cooled with an overlying water pool
3) Ex-vessel debris cooled.

The debris in the reactor cavity can be submerged by water if safety injection is operating after vessel failure or the reactor cavity flooding system operates. If the debris is cooled, its only subsequent challenge to the containment is steam over-pressurization due to the continued addition of the decay heat to the cooling water and hence to the containment.

Physically, the debris is not cooled if the debris surfaces that are exposed to the heat-removing medium are not large enough with respect to the heat generating volume to prevent high temperatures being attained. High surface-to-volume ratios indicate that the debris is being spread thinly over a large surface area. The geometry of the cavity (floor area) is an important factor.

g. Late Containment Failure (LCF)

This event determines whether a gross failure of containment due to over-pressurization and/or over-temperature occurs late in the accident sequence. ("Late" is defined as being a few hours through a few days after reactor vessel failure.) This event is similar to the event for early containment failure, with the accident in progress for a significant amount of time as the obvious difference. Three possibilities are considered as late containment failure:

1) No containment failure
2) Late Containment Failure (Rupture)

A(8/11)

3) Late Containment Failure (Leak)

The phenomena that could potentially contribute to a late containment failure are:

1) Over-pressurization caused by production of steam and/or non-condensable gases
2) Late hydrogen burn
3) Over-temperature failure of containment penetration sealants

The primary cause of failure of the containment is the steam over-pressurization resulting from the loss of the containment heat removal. The steam over-pressurization process is slow and it takes long time to reach the containment failure pressure. The containment pressurization may stop if a small leakage path exists for this process. The possibility of late containment failure due to a late hydrogen burn or over-temperature failure of containment penetration sealants are also considered.

h. Basemat Melt-through (BMT)

This event determines whether or not the containment can fail due to basemat melt-through. Two branches are considered:

1) No basemat melt-through
2) Basemat melt-through.

The containment can fail due to basemat melt-through (even if the cavity is filled with water) if the molten debris is not coolable. Note that if the containment heat removal function is not available, we assume that overpressure failure will occur and we neglect basemat melt-through since the offsite consequences of basemat melt-through would be small compared with those of overpressure failure.

Successful cooling of the cavity debris bed implies that erosion induced containment failure modes will not occur and that the radiological releases are attributable to either an alternate failure mode or containment leakage (assuming no other containment failure mode is identified).

~~19.1.4.2.1.2.3.2~~ 19.1.4.2.1.2.3 Special Containment Event Tree

a. Special Containment Event Tree for SGTR

This CET represented in Figure 19.1-43, considers the steam generator tube rupture (SGTR) with stuck open secondary relief valves. Radiation releases resulting from the SGTR with a stuck open secondary relief valve are significantly influenced by the availability of liquid on the secondary side of the SG. Fission product releases below deep saturated liquid pools can reduce releases of soluble fission products. The availability of liquid on the secondary side of the ruptured SG depends on accident scenarios. Hence, two branches are considered;

1) SGTR with fission product scrubbing.
2) SGTR without fission product scrubbing

This CET applies to PDS 1 and PDS 2. PDS 1 includes the sequences that SGTR event occurs and auxiliary feedwater (AF) system supplies the feedwater into ruptured SG at the time of core damage. PDS 2 includes the sequences that SGTR event occurs and AF system doesn't supply the feedwater into ruptured SG at the time of core damage.

b. Special Containment Event Tree for ISLOCA

The CET for ISLOCA (interfacing system LOCA) PDS, is shown in Figure 19.1-44. The break location outside containment will be underwater by the time the radioactive releases commence. If the break in the interfacing system is located low in the primary auxiliary building, the RCS inventory and water injected into the RCS, flows through the break to the auxiliary building, it will form a pool which covers the break location of the system. Then, scrubbing effects are expected. Hence, two branches are considered;

1) ISLOCA with fission product scrubbing.
2) ISLOCA without fission product scrubbing

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This CET applies to PDS 3 and PDS 4. PDS 3 includes the sequences that ISLOCA event occurs and IRWST water is injected into RCS by SI system. PDS 4 includes the sequences that ISLOCA event occurs and IRWST water is not injected into RCS due to unavailable SI system.

c. Special Containment Event Tree for containment isolation failure

This CETs represented in Figures 19.1-45, consider two different loss of containment isolation scenarios, depending on the containment spray availability. The loss of containment isolation can arise from either mechanical, electrical faults and/or human error (e.g., failure of maintenance to cap off a containment penetration).

If the containment is unisolated and the containment spray operates successfully, it may result in lower radiation releases that that of the sequences which the containment is unisolated and the containment spray is not available. Hence, two branches are considered;

1) Containment isolation failure with containment spray.
2) Containment isolation failure without containment spray

This CET applies to PDS 5 and PDS 6. PDS 5 includes the sequences that the containment is not isolated and containment spray is available. PDS 6 includes the sequences that the containment is not isolated and containment spray is not available.

d. Special Containment Event Tree for Rupture Before Core Melt

This CET represented in Figure 19.1-46, considers that a gross failure of containment occurs before the onset of core damage. The type of accident sequences that lead to containment failure before the onset of core damage involves long term loss of containment heat removal. The accident sequences include large and medium LOCAs within containment with successful injection but failure of containment sprays or a transient with failure of secondary side heat removal followed by successful feed and bleed cooling but with failure to remove the energy from containment by cooling the IRWST. Analyses have shown that the time between loss of containment cooling and failure of containment is at least 40 hours. Following containment failure, fuel damage and fission product releases would occur if core cooling is lost. Core cooling would be lost if the SI pumps trip at the time of containment failure, and are not restarted or if the SI pumps are available but the IRWST inventory is not replenished before the existing inventory is lost through the

A(11/11)

containment breach. As previously mentioned in section 3.2, there are two containment failure modes which consist of leak mode and rupture mode. The typical leak size is evaluated to be of the order of 9.29E-3 $m^2$ (0.1 $ft^2$), and the typical rupture size is evaluated to be of the order of approximately 9.29E-2 $m^2$ (1.0 $ft^2$).

However, if the ECSBS operates under these situations, the containment over-pressurization would be prevented by sprays of ECSBS. And then, the core melt will not occur due to available core cooling. Hence, three branches are considered;

1) Containment failure (leak mode) before core damage.
2) Containment failure (rupture mode) before core damage.
3) Core melt arrest, and the containment over-pressurization is prevented by ECSBS

This CET applies to PDS 7. PDS 7 includes the sequences that the core cooling is maintained by SI injection or secondary heat removal, however, the containment spray is not available.

# REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**APR1400 Design Certification**

**Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD**

**Docket No. 52-046**

| | |
|---|---|
| **RAI No.:** | **432-8377** |
| **SRP Section:** | **SRP 19** |
| **Application Section:** | **19.1** |
| **Date of RAI Issue:** | **03/08/2016** |

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## Question No. 19-56

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

APR1400 design control document (DCD) Rev. 0, Tables 19.1-29 and -30, provide Summary of Source Term Evaluation and Source Term Category Frequencies and Contributions to large release frequency (LRF) for internal events. However, similar information is not provided in the DCD for internal fire and internal flooding. Update the DCD providing similar tables for internal fire and internal flooding.

## Response – (Rev. 1)

The Table 19.1-29 shows the summary of source term evaluation not only for internal events, but also for internal fire & flooding events. In the APR1400 At-power Level 2 PRA, the accident scenarios for internal fire events and internal flooding events are assigned to the same STC (i.e., source term category) grouping logic which is used for internal events. Per the definition of source term category, a particular release category includes the accident scenarios which have similar source term governing characteristics. The source term is the result of the MAAP analysis and presents the release fraction of the initial core inventory which is released to the environment as a function of time. To characterize the source term associated with each release category, a single representative sequence was chosen for each release category by using MAAP code. Therefore, the source term for each STC represent various sequences assigned to each STC resulting from the internal events as well as internal fire & flooding events.

New tables regarding the summary of STC frequencies for At-power Internal Fire Events and Internal Flooding Events (i.e., similar information with Table 19.1-30) will be included in APR1400 DCD 19.1 as shown in the Attachment.

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**Impact on DCD**

The DCD will be revised to reflect the response of this RAI as shown in the Attachment.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on the Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

There is no impact on any Technical, Topical, or Environment Report.

due to steam overpressurization. In this category, there is no significant fission product release to the environment due to a wet cavity. However, the releases are not scrubbed by the containment sprays.

The summary of the MAAP results (release magnitude and timing) and release categorization (i.e., large release, large early release, or not large release) is presented in Table 19.1-29 ~~and Table 19.1-30~~. through 19.1-30b

19.1.4.2.2 Results from Level 2 Internal Events PRA for Operations at Power

It should be noted that units for CDF and LRF are expressed in terms of "reactor calendar year" (shortened to "/year" when displayed in the text in this section).

19.1.4.2.2.1 Risk Metrics

Total LRF from internal events is $1.1 \times 10^{-7}$/year. This is well below the NRC goal for LRF below $1 \times 10^{-6}$/year. Mean value and associated uncertainty distribution can be found in Subsection 19.1.4.2.2.7.

The conditional containment failure probability (CCFP) from all internal events (at power) in large release sequences is $8.4 \times 10^{-2}$. This meets the NRC goal of no more than approximately 0.1 for CCFP. This CCFP is the conditional probability of a large release (CPLR) for operations at power.

19.1.4.2.2.2 Internal Events Core Damage Release Category Results

The relative contributions of the release categories to the total STC frequency are shown in Figure 19.1-49. Figure 19.1-50 groups the categories further into no contailment failure, large release, and small release.

Approximately 49 percent of the LRF for internal events is from STC 1, which are unmitigated, unisolated SGTR releases (both SGTR initiating event and induced SGTR). The next-highest frequency STC is a late rupture with no containment sprays (27 percent), followed by containment failure (rupture) prior to core damage (12 percent), and containment failure (leak) prior to core damage (10 percent). Early containment rupture

Table 19.1-30 (1 of 2)

Source Term Category Frequencies and Contributions to LRF for Internal Events

| Source Term Category | Description | LRF, LERF or non-LRF | Frequency | % of total STC freq | % of total LRF |
|---|---|---|---|---|---|
| STC 1 | SGTR w/o scrubbing | LRF / LERF | 5.33E-08 | 4.1 | 48.5 |
| STC 21 | Late containment failure with a rupture failure size | LRF | 2.96E-08 | 2.3 | 26.9 |
| STC 8 | CFBRB with a rupture failure size | LRF / LERF | 1.30E-08 | 1.0 | 11.8 |
| STC 7 | CFBRB with a leak failure size | LRF | 1.14E-08 | 0.9 | 10.4 |
| STC 13 | Early containment failure with a rupture failure size | LRF / LERF | 1.79E-09 | 0.1 | 1.6 |
| STC 6 | Not isolation w/o CS | LRF / LERF | 1.23E-09 | 0.1 | 1.1 |
| STC 4 | ISLOCA with scrubbing | LRF / LERF | 6.49E-11 | 0.0 | 5.9E-02 |
| STC 3 | ISLOCA w/o scrubbing | LRF / LERF | 5.31E-11 | 0.0 | 4.8E-02 |
| STC 20 | Late containment failure with a rupture failure size | LRF | 1.19E-11 | 0.0 | 1.1E-02 |
| STC 2 | SGTR with scrubbing | Non-LRF | 2.41E-08 | 1.8 | |
| STC 5 | Not isolation with CS | Non-LRF | 2.46E-09 | 0.2 | |
| STC 9 | Intact containment w/o RPV breach | Non-LRF | 3.67E-07 | 28 | |
| STC 10 | Intact containment with RPV breach | Non-LRF | 7.64E-07 | 58.2 | |
| STC 11 | Basemat Melt-through | Non-LRF | 1.33E-08 | 1.0 | |
| STC 12 | Early containment failure with a leak failure size | Non-LRF | 0.00 | 0.0 | |
| STC 14 | Late containment failure with a leak failure size | Non-LRF | 4.28E-11 | 0.0 | |

Table 19.1-30 (2 of 2)

| Source Term Category | Description | LRF, LERF or non-LRF | Frequency | % of total STC freq | % of total LRF |
|---|---|---|---|---|---|
| STC 15 | Late containment failure with a leak failure size | Non-LRF | 0.00 | 0.0 | |
| STC 16 | Late containment failure with a leak failure size | Non-LRF | 7.30E-12 | 0.0 | |
| STC 17 | Late containment failure with a leak failure size | Non-LRF | 2.70E-08 | 2.1 | |
| STC 18 | Late containment failure with a rupture failure size | Non-LRF | 4.19E-10 | 0.0 | |
| STC 19 | Late containment failure with a rupture failure size | Non-LRF | 4.01E-09 | 0.3 | |
| Total frequency of all STCs | | | 1.31E-06 | | |
| Total frequency of the Large Release STCs | | | 1.10E-07 | | |

Table Added
(19.1-30a, 19.1-30b)

Table 19.1-30a

Source Term Category Frequencies and Contributions to LRF for Internal Fire Events

| Source Term Category[1)] | Description | LRF, LERF or non-LRF | Frequency | % of total STC freq | % of total LRF |
|---|---|---|---|---|---|
| STC 1 | SGTR w/o scrubbing | LRF / LERF | 8.31E-08 | 4.0 | 49.4 |
| STC 21 | Late containment failure with a rupture failure size | LRF | 4.38E-08 | 2.1 | 26.1 |
| STC 6 | Not isolation w/o CS | LRF / LERF | 2.60E-08 | 1.3 | 15.5 |
| STC 8 | CFBRB with a rupture failure size | LRF / LERF | 6.16E-09 | 0.3 | 3.7 |
| STC 7 | CFBRB with a leak failure size | LRF | 4.36E-09 | 0.2 | 2.6 |
| STC 13 | Early containment failure with a rupture failure size | LRF / LERF | 3.71E-09 | 0.2 | 2.2 |
| STC 20 | Late containment failure with a rupture failure size | LRF | 9.34E-10 | 0.0 | 0.6 |
| STC 5 | Not isolation with CS | Non-LRF | 2.38E-08 | 1.2 | - |
| STC 9 | Intact containment w/o RPV breach | Non-LRF | 2.83E-07 | 13.7 | - |
| STC 10 | Intact containment with RPV breach | Non-LRF | 9.62E-07 | 46.6 | - |
| STC 11 | Basemat Melt-through | Non-LRF | 5.56E-07 | 26.9 | - |
| STC 14 | Late containment failure with a leak failure size | Non-LRF | 2.39E-09 | 0.1 | - |
| STC 16 | Late containment failure with a leak failure size | Non-LRF | 5.71E-10 | 0.0 | - |
| STC 17 | Late containment failure with a leak failure size | Non-LRF | 3.99E-08 | 1.9 | - |
| STC 18 | Late containment failure with a rupture failure size | Non-LRF | 2.34E-08 | 1.1 | - |
| STC 19 | Late containment failure with a rupture failure size | Non-LRF | 6.51E-09 | 0.3 | - |
| Total frequency of all STCs | | | 2.07E-06 | | |
| Total frequency of the Large Release STCs | | | 1.68E-07 | | |

Note 1: There are no sequences assigned to the STC 2, STC 3, STC 4, STC 12 and STC 15 in the internal fire events.

Table 19.1-30b

Source Term Category Frequencies and Contributions to LRF for Internal Flooding Events

| Source Term Category[1)] | Description | LRF, LERF or non-LRF | Frequency | % of total STC freq | % of total LRF |
|---|---|---|---|---|---|
| STC 21 | Late containment failure with a rupture failure size | LRF | 8.70E-09 | 3.7 | 51.2 |
| STC 1 | SGTR w/o scrubbing | LRF / LERF | 3.42E-09 | 1.5 | 20.2 |
| STC 6 | Not isolation w/o CS | LRF / LERF | 3.29E-09 | 1.4 | 19.4 |
| STC 8 | CFBRB with a rupture failure size | LRF / LERF | 5.94E-10 | 0.3 | 3.5 |
| STC 7 | CFBRB with a leak failure size | LRF | 5.38E-10 | 0.2 | 3.2 |
| STC 13 | Early containment failure with a rupture failure size | LRF / LERF | 4.33E-10 | 0.2 | 2.5 |
| STC 20 | Late containment failure with a rupture failure size | LRF | 3.37E-12 | 0.0 | 2.0E-04 |
| STC 5 | Not isolation with CS | Non-LRF | 3.86E-10 | 0.2 | - |
| STC 9 | Intact containment w/o RPV breach | Non-LRF | 2.39E-08 | 10.1 | - |
| STC 10 | Intact containment with RPV breach | Non-LRF | 1.82E-07 | 77.2 | - |
| STC 11 | Basemat Melt-through | Non-LRF | 2.92E-09 | 1.2 | - |
| STC 14 | Late containment failure with a leak failure size | Non-LRF | 8.73E-12 | 0.0 | - |
| STC 16 | Late containment failure with a leak failure size | Non-LRF | 2.06E-12 | 0.0 | - |
| STC 17 | Late containment failure with a leak failure size | Non-LRF | 7.93E-09 | 3.4 | - |
| STC 18 | Late containment failure with a rupture failure size | Non-LRF | 8.55E-11 | 0.0 | - |
| STC 19 | Late containment failure with a rupture failure size | Non-LRF | 1.53E-09 | 0.6 | - |
| Total frequency of all STCs | | | 2.35E-07 | | |
| Total frequency of the Large Release STCs | | | 1.70E-08 | | |

Note 1: There are no sequences assigned to the STC 2, STC 3, STC 4, STC 12 and STC 15 in the internal flooding events.