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.:	**416-8358**
SRP Section:	**SRP 19**
Application Section:	**19.1**
Date of RAI Issued:	**02/23/2016**

Question No. 19-35

Item 11 of Section II, "Acceptance Criteria," of the (Draft) Revision 3 SRP, states, "The PRAs that meet the applicable supporting requirements for Capability Category I and meet the high-level requirements as defined in the ASME PRA Standard (ASME/ANS RA-S-2008 and addenda ASME/ANS RA-Sa–2009) should generally be acceptable for DC and COL applications. Alternatively, the applicant may identify, and justify the acceptability of, alternative measures for addressing PRA quality and technical adequacy. The staff should specifically review the acceptability of these alternative measures in the context of the specific uses and applications of the PRA."

The staff reviewed the APR1400 design control document (DCD) Section 19.1.4.1.1, "Description of Level 1 Internal Events PRA for Operations at Power," and found insufficient information describing the accident sequence analysis performed. Specifically, the applicant did not identify and discuss a rationale for accident sequences that were not sequentially ordered according to the timing of the event (ASME/ANS PRA Standard supporting requirements – AS-A6). Therefore, in order for the staff to reach an assurance finding on the conformance to SRP Chapter 19.0 regarding PRA technical adequacy, please revise the DCD with a description of accident sequences not sequentially ordered.

Response

There were no accident sequences that were not ordered sequentially. The comment in the DCD is a general statement of fact that the order of events may be changed to simplify the event tree structure as long as the proper functional relationships are maintained. Section 19.1.4.1.1.2 is revised as shown in Attachment.

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 Technical/Topical/Environmental Report.

(left to right)

System and operator functional responses are ordered in the event trees sequentially based on the timing of the accident scenarios as they develop. ~~In selected cases, events may be ordered differently to simplify the event tree structure while retaining the proper functional relationships.~~

Internal flooding, while considered to be an internal event, is described separately in Subsection 19.1.5.3.

Each Level 1 event tree sequence is assigned to an end state. The possible end states (for Level 1 analysis) are:

a. OK – The key safety functions have been performed successfully so that core damage is prevented during the mission time or a safe stable state is reached at a time beyond the 24-hour mission time;

b. CD – One or more key safety functions have failed in such a way that core damage will occur; or

c. TR – The accident progression has resulted in a transfer to another event tree that will define additional success requirements (e.g., a transient that results in a failure to scram the reactor will be transferred to the ATWS event tree).

Further classification of the core damage end states into specific plant damage states (PDSs) is performed in the Level 2 PRA analysis (see Subsection 19.1.4.2).

19.1.4.1.1.3 Success Criteria Analysis

The approach used in this success criteria analysis is based on the ASME/ANS PRA Standard requirements. The technical portions of the success criteria determination are based on the following:

a. The definition of core damage

Core damage is defined as the uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage involving a large fraction of the core is anticipated.

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Question No. 19-36

Item 11 of Section II, “Acceptance Criteria,” of the (Draft) Revision 3 SRP, states, “The PRAs that meet the applicable supporting requirements for Capability Category I and meet the high-level requirements as defined in the ASME PRA Standard (ASME/ANS RA-S-2008 and addenda ASME/ANS RA-Sa–2009) should generally be acceptable for DC and COL applications. Alternatively, the applicant may identify, and justify the acceptability of, alternative measures for addressing PRA quality and technical adequacy. The staff should specifically review the acceptability of these alternative measures in the context of the specific uses and applications of the PRA.”

The staff reviewed the APR1400 design control document (DCD) Section 19.1.4.1.1, “Description of Level 1 Internal Events PRA for Operations at Power,” and found insufficient information describing the accident sequence analysis performed. Specifically,

1 The applicant did not describe the method used to implement event tree transfers

2 The transfers shown in the event trees to not identify the trees they transfer to

3 The applicant did not justify accident sequences with reactor trip failure that ended in core damage (ASME/ANS PRA Standard supporting requirements – AS-A11).

Therefore, in order for the staff to reach an assurance finding on the conformance to SRP Chapter 19.0 regarding PRA technical adequacy, please revise the DCD accordingly with the information needed.

Response

1. Event tree transfers are analyzed internally within the SAREX code. In the development of the event trees, the first node for event trees which are transferred to defines which sequences from all other event trees are being transferred in.

For example, in the development of the GRID-LOOP event tree, the first node is defined as a transfer (vs. a normal initiating event frequency), and all of the event tree sequences which transfer to it are defined within the node logic (e.g., GTRN sequence 6, TLOCCW sequence 4, etc.). During quantification, the SAREX code incorporates the transferred sequences, including any and all initiating events, and previous successes and/or failures from the originating event trees, and then continues with the logical quantification of the rest of the GRID-LOOP event tree. Quantification in this manner preserves the dependencies that are part of the transferred sequence. The DCD is revised to reflect this (Attachment 1).

2. The pictures of the event trees provided in the DCD with transfer sequences will be revised to identify the event tree to which it is being transferred. Attachment 2 contains the revised event tree pictures.

   The revised pictures include an identifier to indicate to which event tree the sequence is transferred. Obviously, transfer sequences which include failure of reactor trip transfer to the ATWS event tree (TR-ATWS), transfer sequences which include stuck open POSRVs transfer to the stuck-open POSRV event tree (TR-PR-SL), transfer sequences with include consequential LOOP transfer to the consequential LOOP event tree (TR-GRID-LOOP), LOOP sequences with failure of all DGs transfer to the SBO event tree (TR-SBO) , and finally GRID-LOOP sequences with failure of all DGs transfer to the GRID-SBO event tree (TR-GRID-SBO).

3. In cases where there is no technical justification for a safe stable state, core damage was assumed. An anticipated transient without scram (ATWS) is an anticipated operational occurrence (AOO) as defined in Appendix A of 10 CFR Part 50 followed by the failure of the reactor trip portion of the protection system as specified in General Design Criteria (GDC) 20. There is no analysis of non-AOO events (e.g., LOCAs, secondary side breaks, etc.) with subsequent failure of reactor trip. Since these are unanalyzed conditions, they are assumed to lead directly to core damage. Due to a combination of generally low initiating event frequencies of non-AOO events and the reliability of the reactor trip system, the CDF for each of these sequences is less than 3E-09 with the sum total of all of these sequences ~8E-09 which is < 1% than the total internal events CDF of ~1E-06. The DCD is revised to reflect this (Attachment 1).

Impact on DCD

The DCD will be revised as shown in the Attachment 1 and 2.

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 Technical/Topical/Environmental Report.

System and operator functional responses are ordered in the event trees sequentially based on the timing of the accident scenarios as they develop. In selected cases, events may be ordered differently to simplify the event tree structure while retaining the proper functional relationships.

Internal flooding, while considered to be an internal event, is described separately in Subsection 19.1.5.3.

Each Level 1 event tree sequence is assigned to an end state. The possible end states (for Level 1 analysis) are:

if there is no technical justification for a safe stable state; or

a. OK – The key safety functions have been performed successfully so that core damage is prevented during the mission time or a safe stable state is reached at a time beyond the 24-hour mission time;

b. CD – One or more key safety functions have failed in such a way that core damage will occur; or

c. TR – The accident progression has resulted in a transfer to another event tree that will define additional success requirements (e.g., a transient that results in a failure to scram the reactor will be transferred to the ATWS event tree).

Further classification of the core damage end states into specific plant damage states (PDSs) is performed in the Level 2 PRA analysis (see Subsection 19.1.4.2).

Insert A in next page

19.1.4.1.1.3 Success Criteria Analysis

The approach used in this success criteria analysis is based on the ASME/ANS PRA Standard requirements. The technical portions of the success criteria determination are based on the following:

a. The definition of core damage

Core damage is defined as the uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage involving a large fraction of the core is anticipated.

A

In reference to item b. above, an anticipated transient without scram (ATWS) is an anticipated operational occurrence (AOO) as defined in Appendix A of 10 CFR Part 50 followed by the failure of the reactor trip portion of the protection system as specified in General Design Criteria (GDC) 20. There is no analysis of non-AOO events (e.g., LOCAs, secondary side breaks, etc.) with subsequent failure of reactor trip. Since these are unanalyzed conditions, they are assumed to lead directly to core damage (e.g., MLOCA sequence 4 - Figure 19.1-16).

In reference to item c. above, event tree transfers are analyzed internally within the SAREX code. In the development of the event trees, the first node for event trees which are transferred to defines which sequences from all other event trees are being transferred in. For example, in the development of the GRID-LOOP event tree (Figure 19.1-32), the first node (GRID-LOOP) is defined as a transfer (vs. a normal initiating event frequency), and all of the event tree sequences which transfer to it are defined within the node logic (e.g., GTRN sequence 6 - Figure 19.1-22, TLOCCW sequence 4 - Figure 19.1-36, etc.). During quantification, the SAREX code incorporates the transferred sequences, including any and all initiating events, and previous successes and/or failures from the originating event trees, and then continues with the logical quantification of the rest of the GRID-LOOP event tree. Quantification in this manner preserves the dependencies that are part of the transferred sequence.

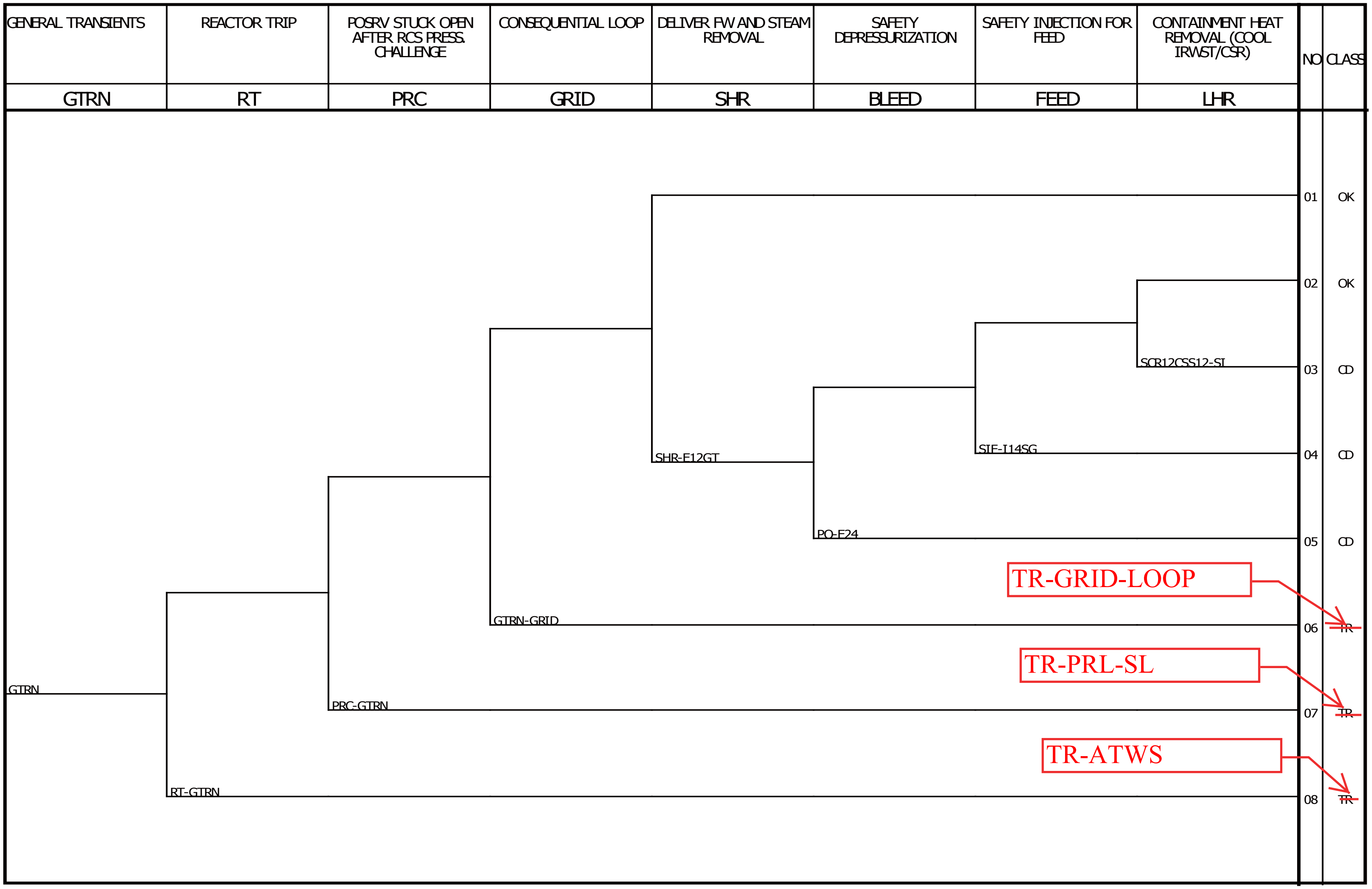

Figure 19.1-22 Level 1 Event Tree - General Transient (GTRN)

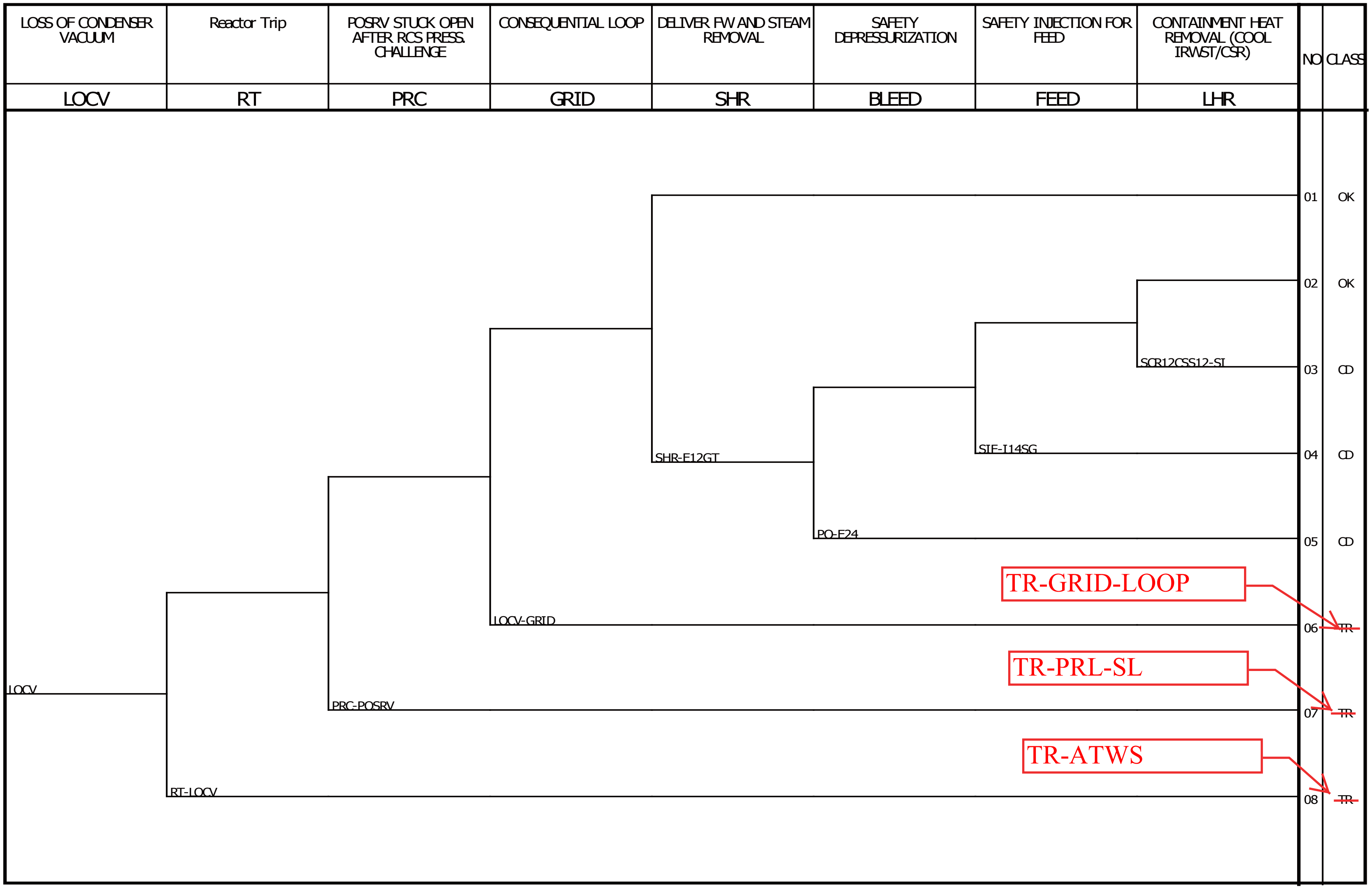

Figure 19.1-23 Level 1 Event Tree - Loss of Condenser Vacuum (LOCV)

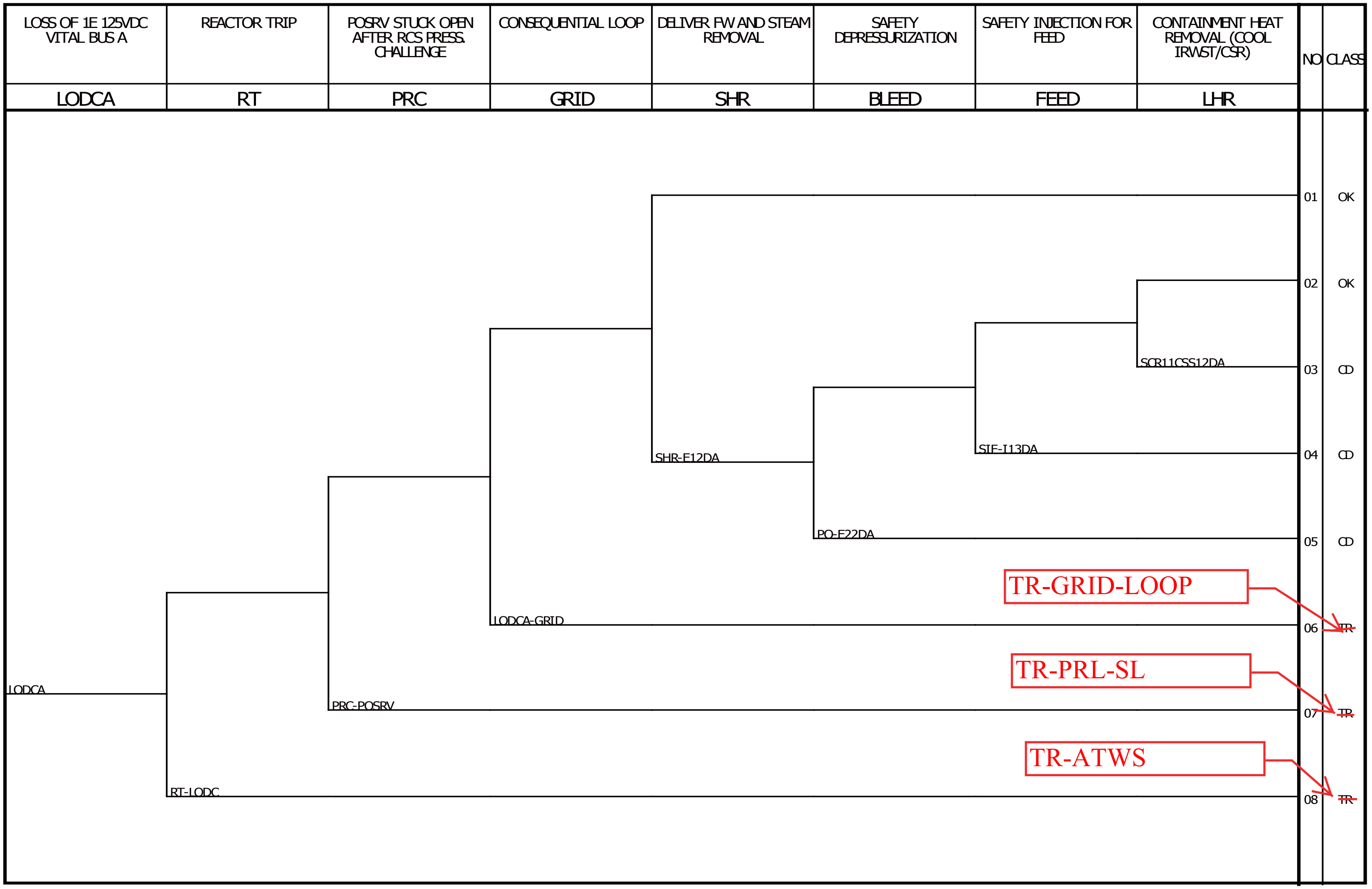

Figure 19.1-24 Level 1 Event Tree - Loss of 125 Vdc - Bus A (LODCA)

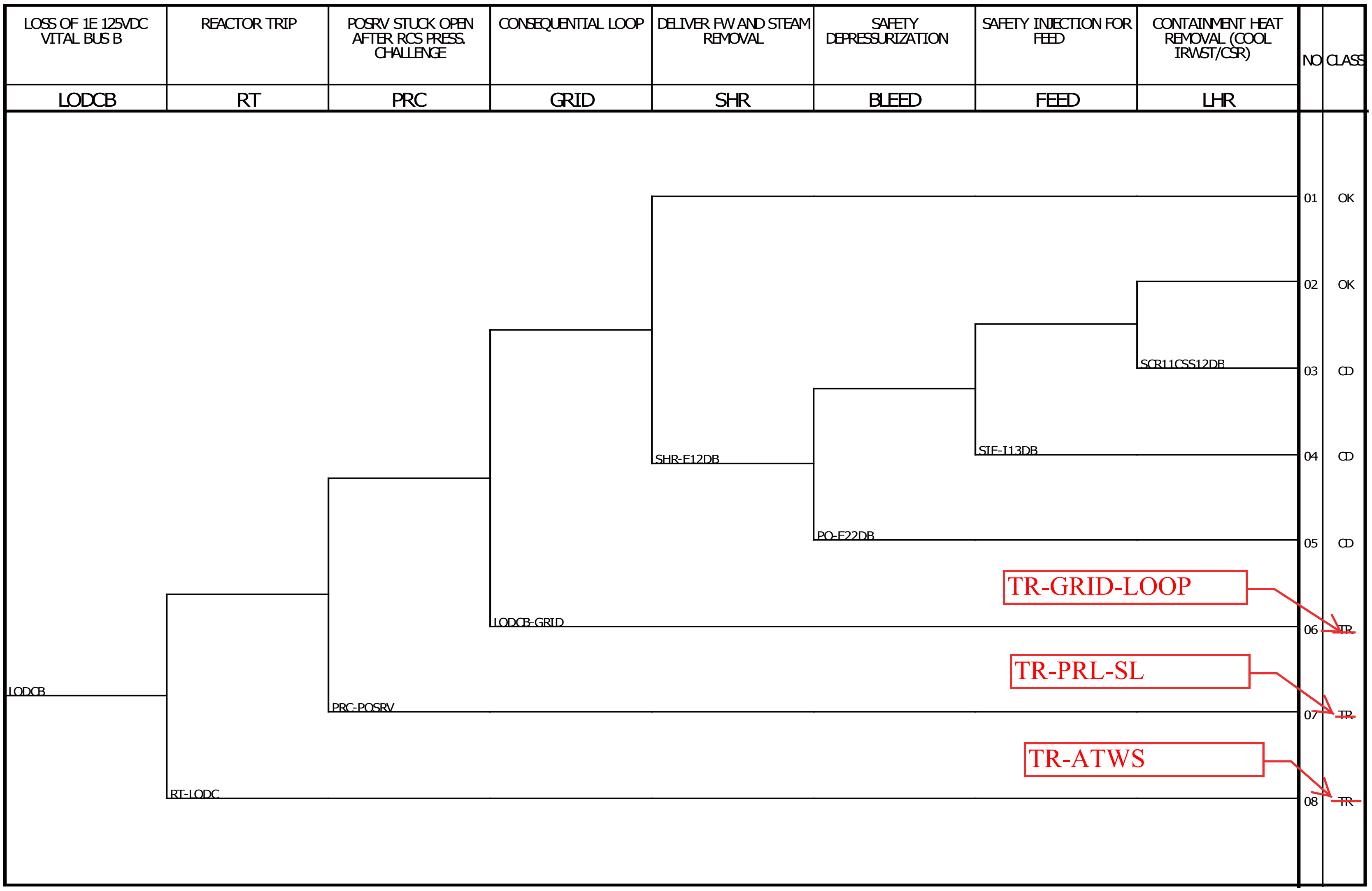

Figure 19.1-25 Level 1 Event Tree - Loss of 125 Vdc - Bus B (LODCB)

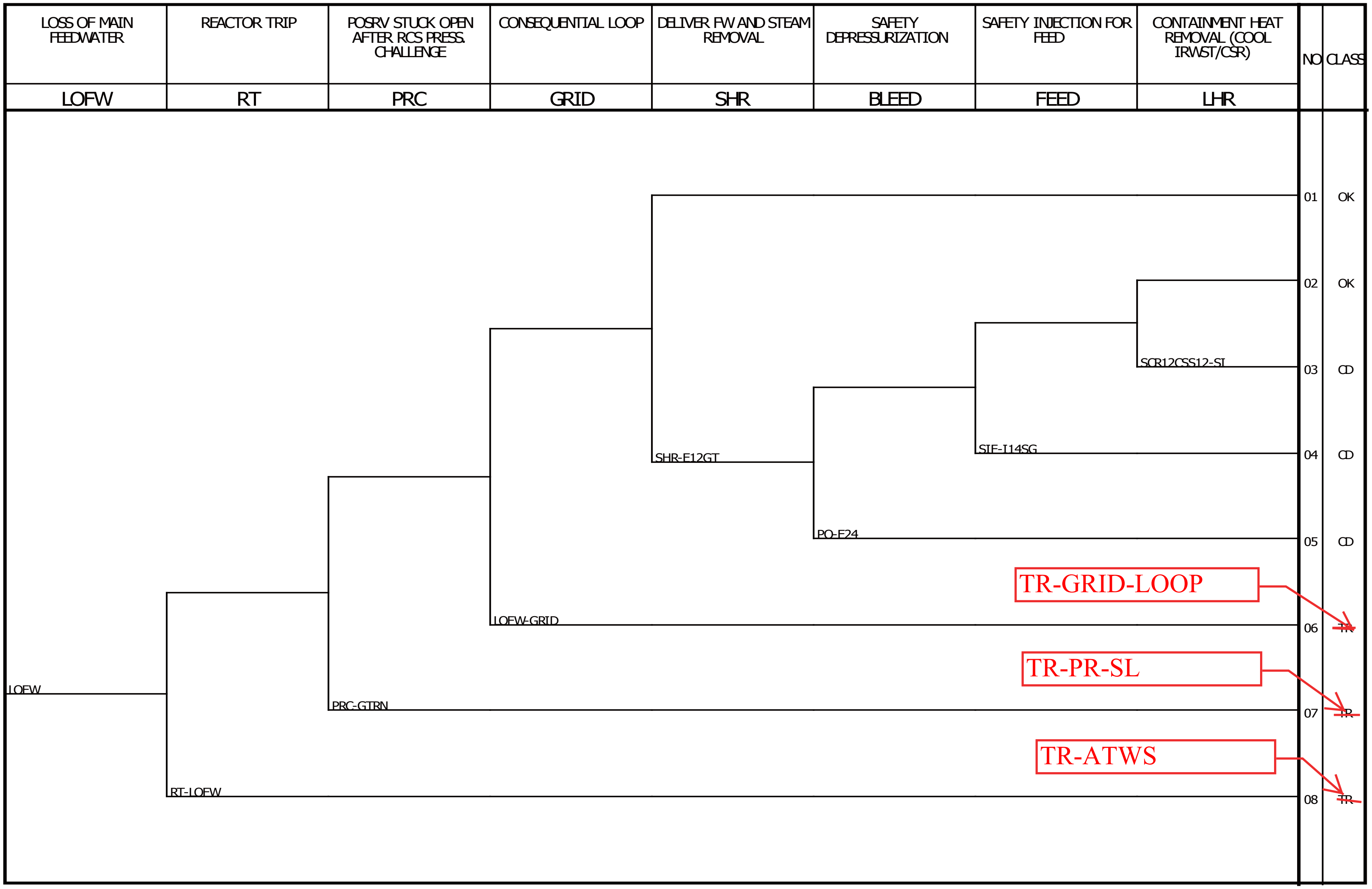

Figure 19.1-26 Level 1 Event Tree - Loss of Feedwater (LOFW)

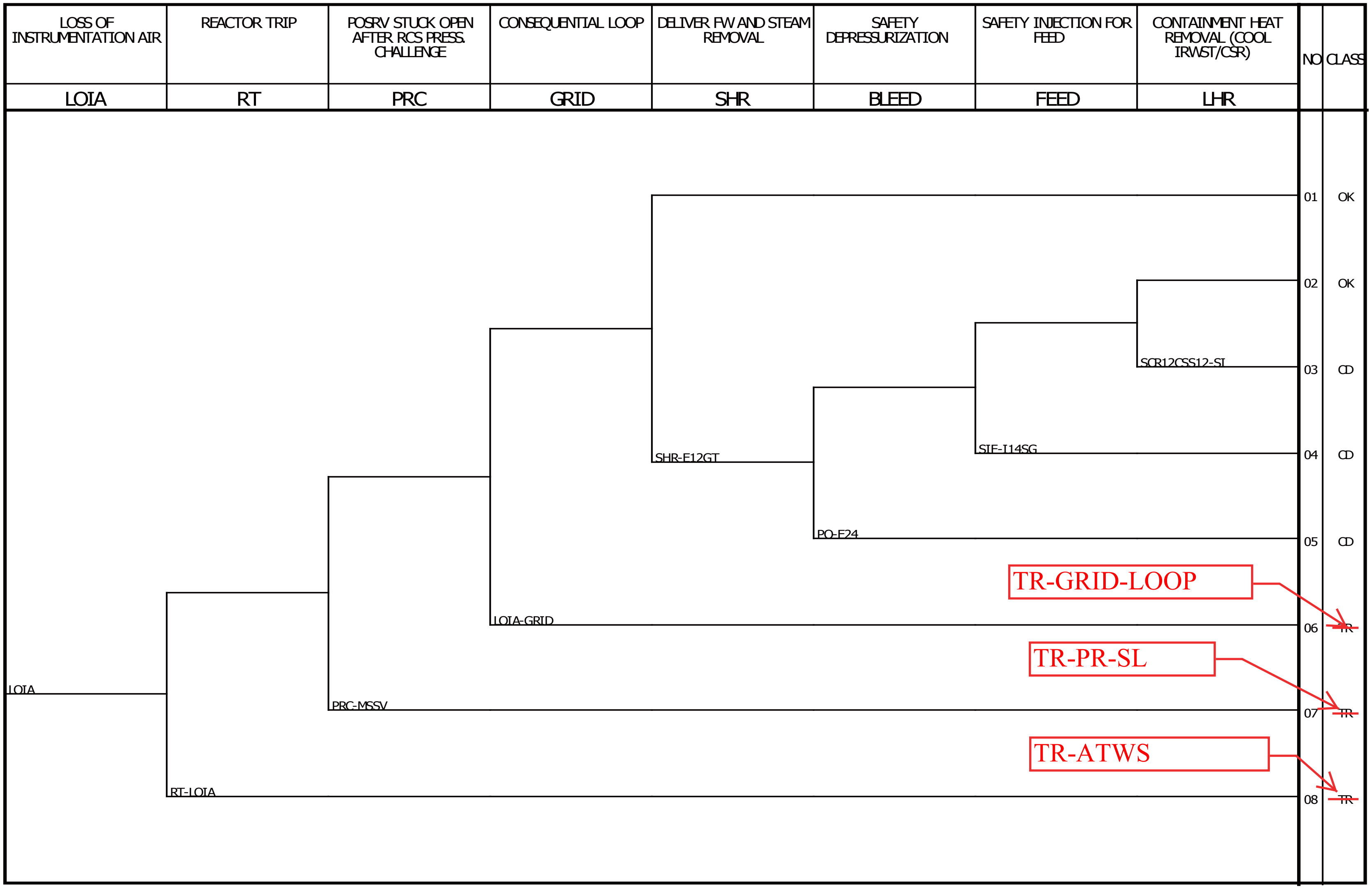

Figure 19.1-27 Level 1 Event Tree - Loss of Instrument Air (LOIA)

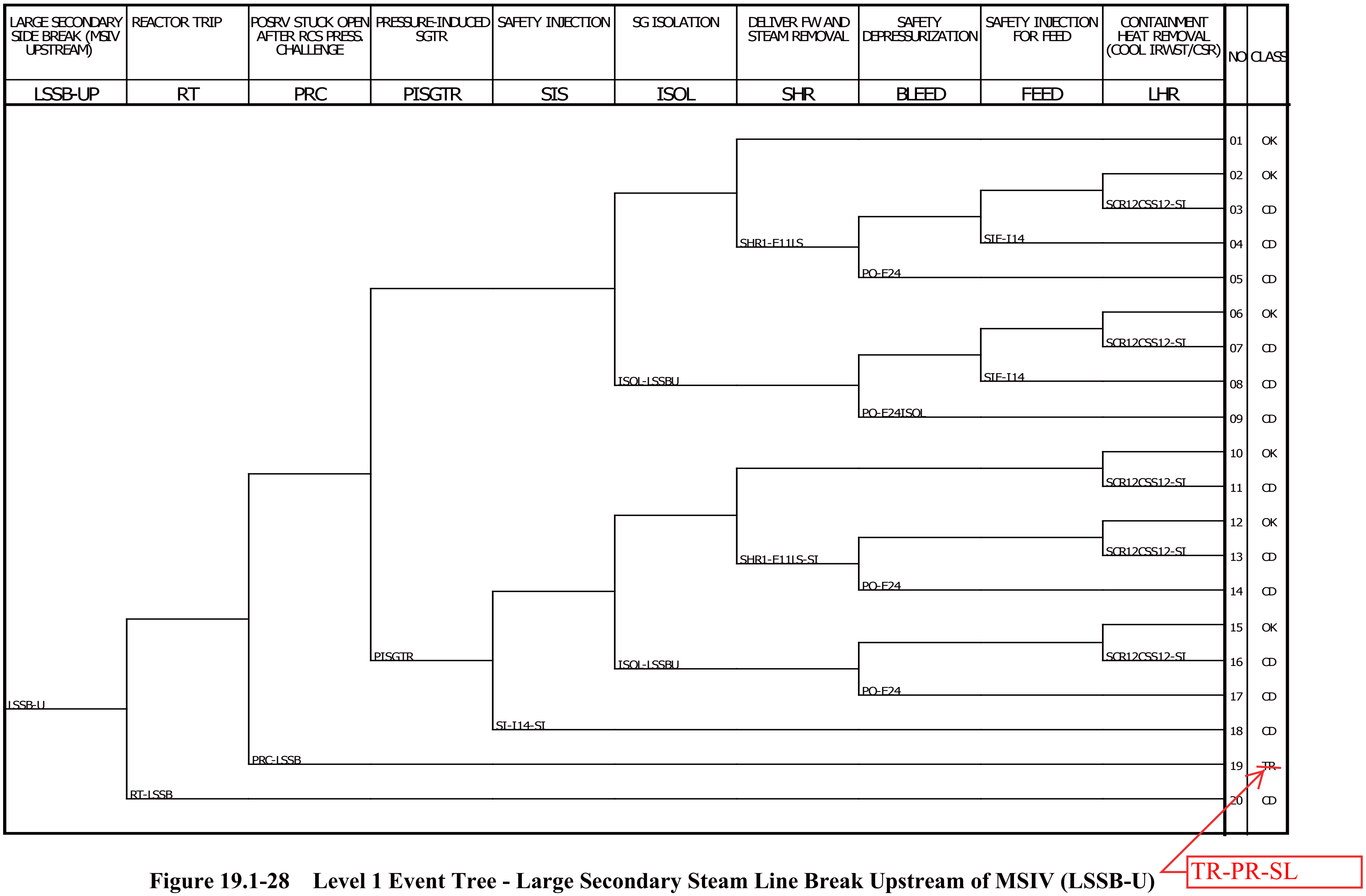

Figure 19.1-28 Level 1 Event Tree - Large Secondary Steam Line Break Upstream of MSIV (LSSB-U)

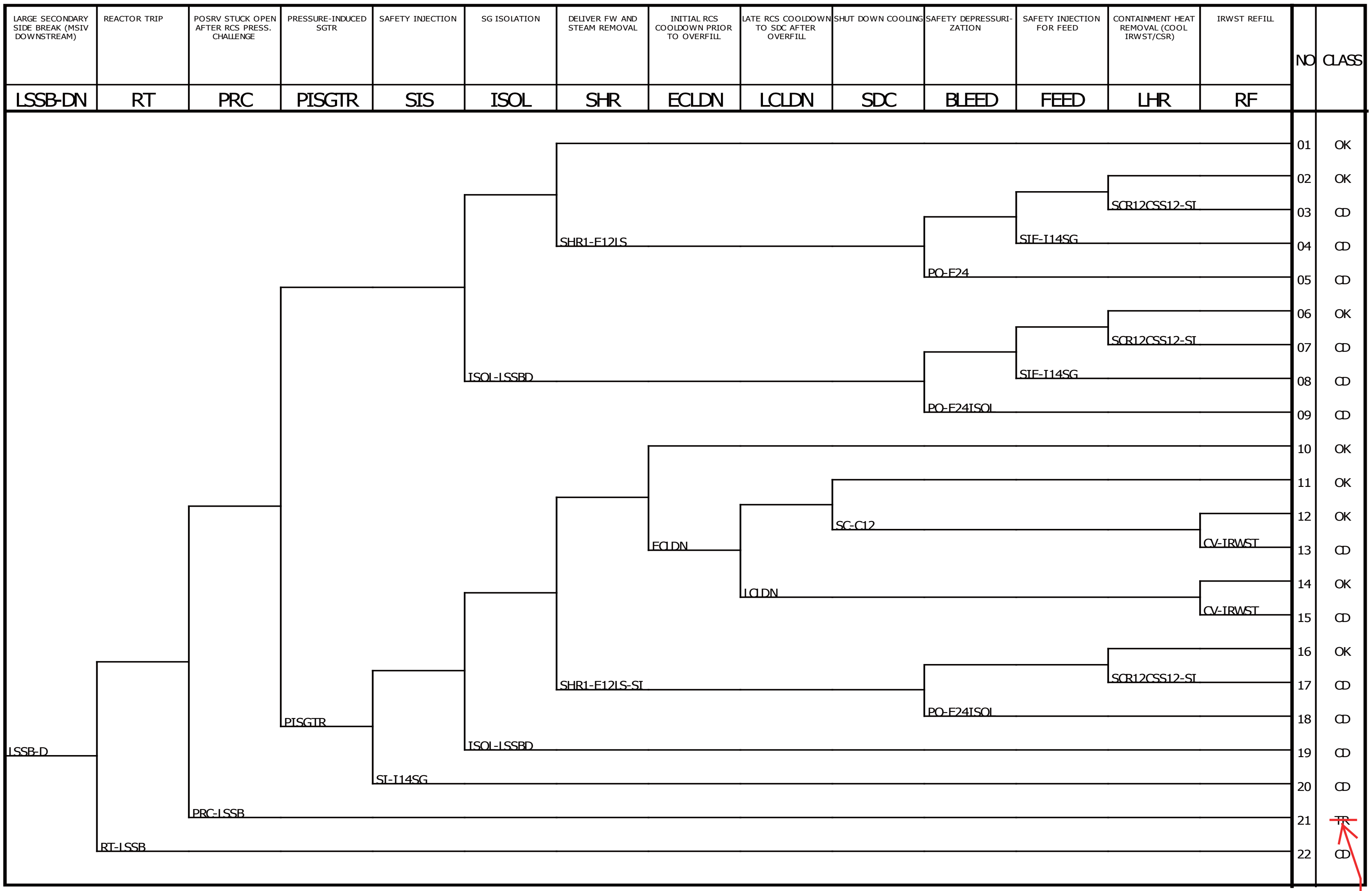

Figure 19.1-29 Level 1 Event Tree - Large Secondary Steam Line Break Downstream of MSIV (LSSB-D)

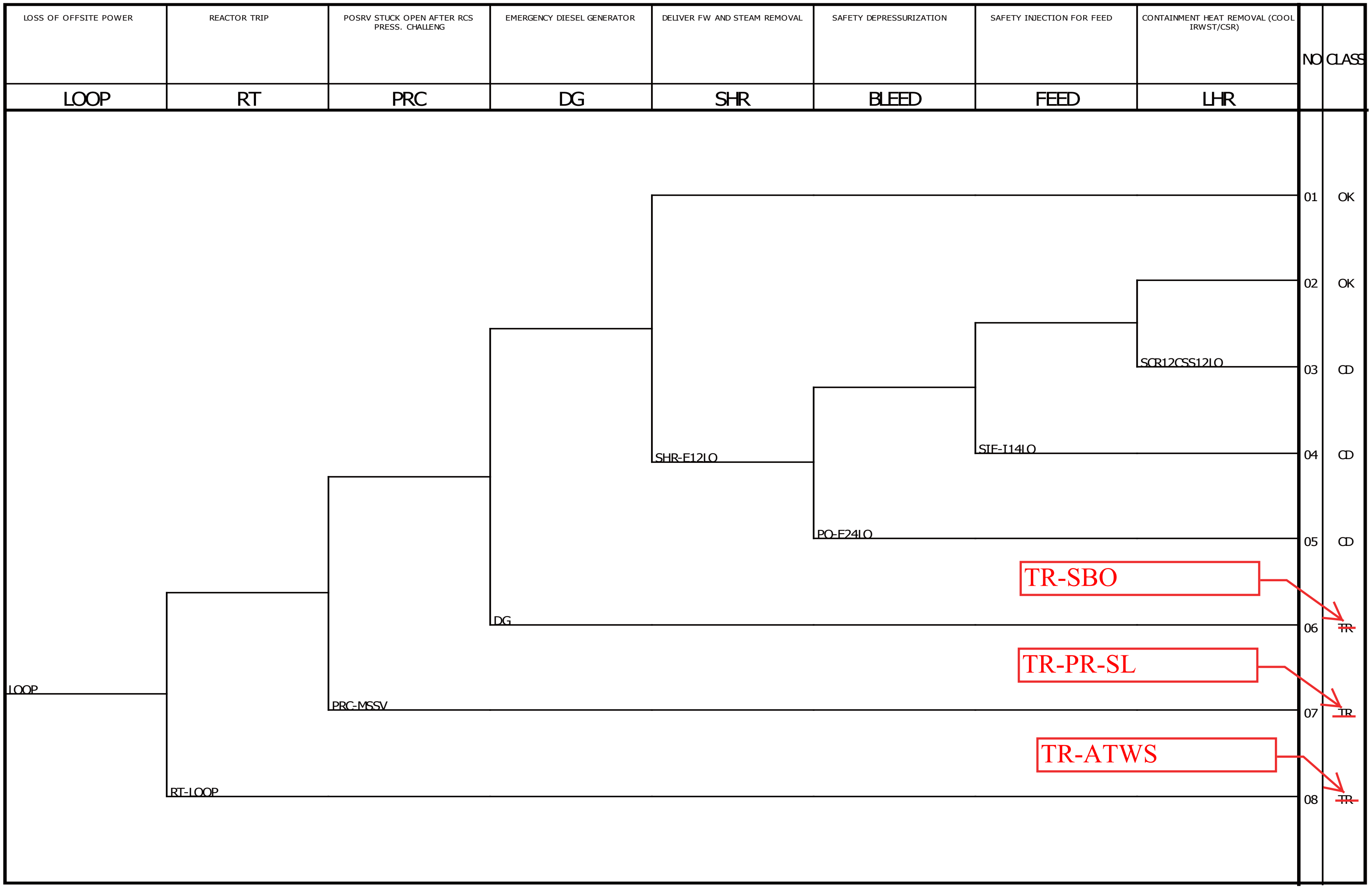

Figure 19.1-31 Level 1 Event Tree - Loss of Offsite Power (LOOP)

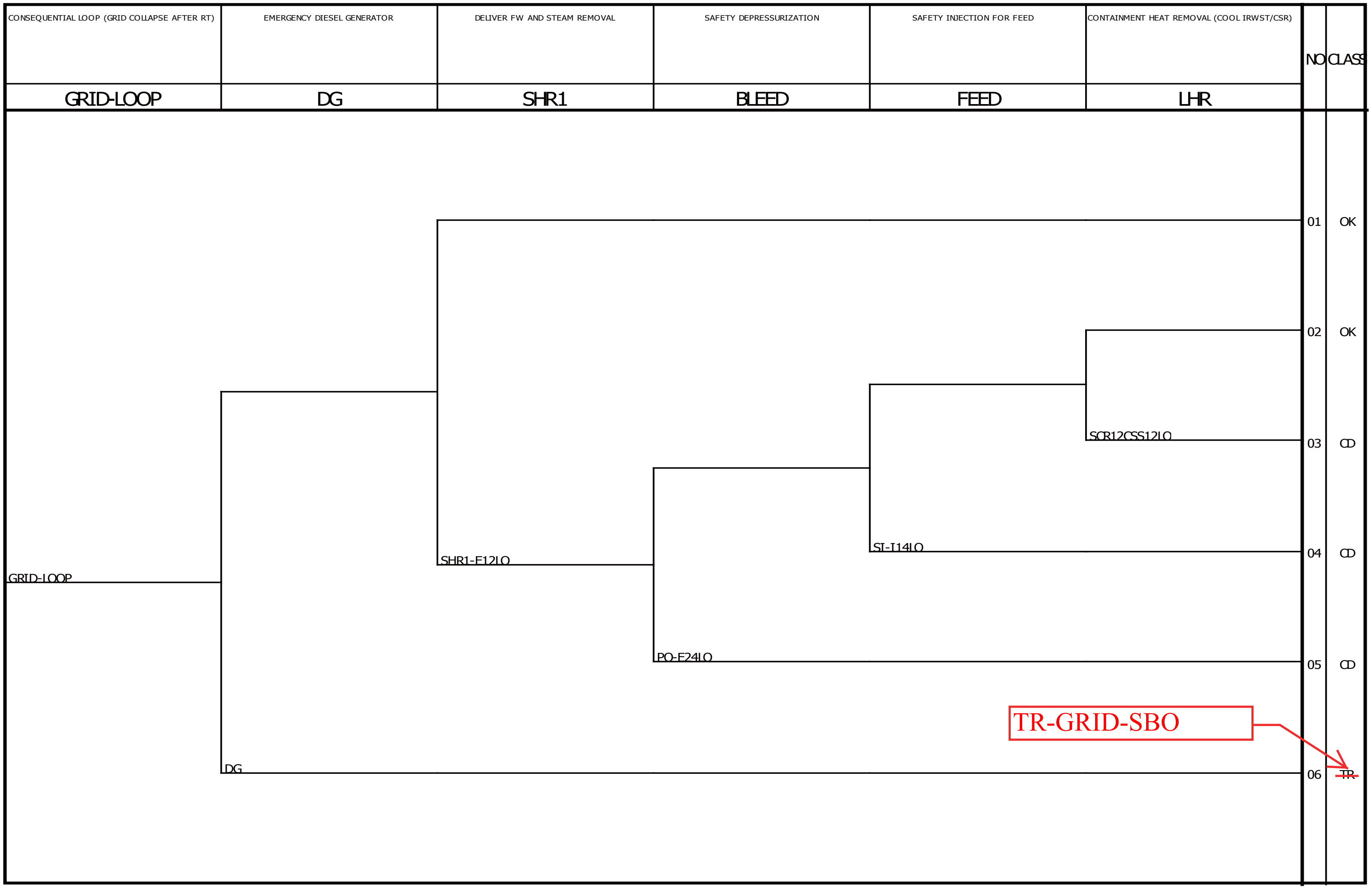

Figure 19.1-32 Level 1 Event Tree - Consequential LOOP (GRID-LOOP)

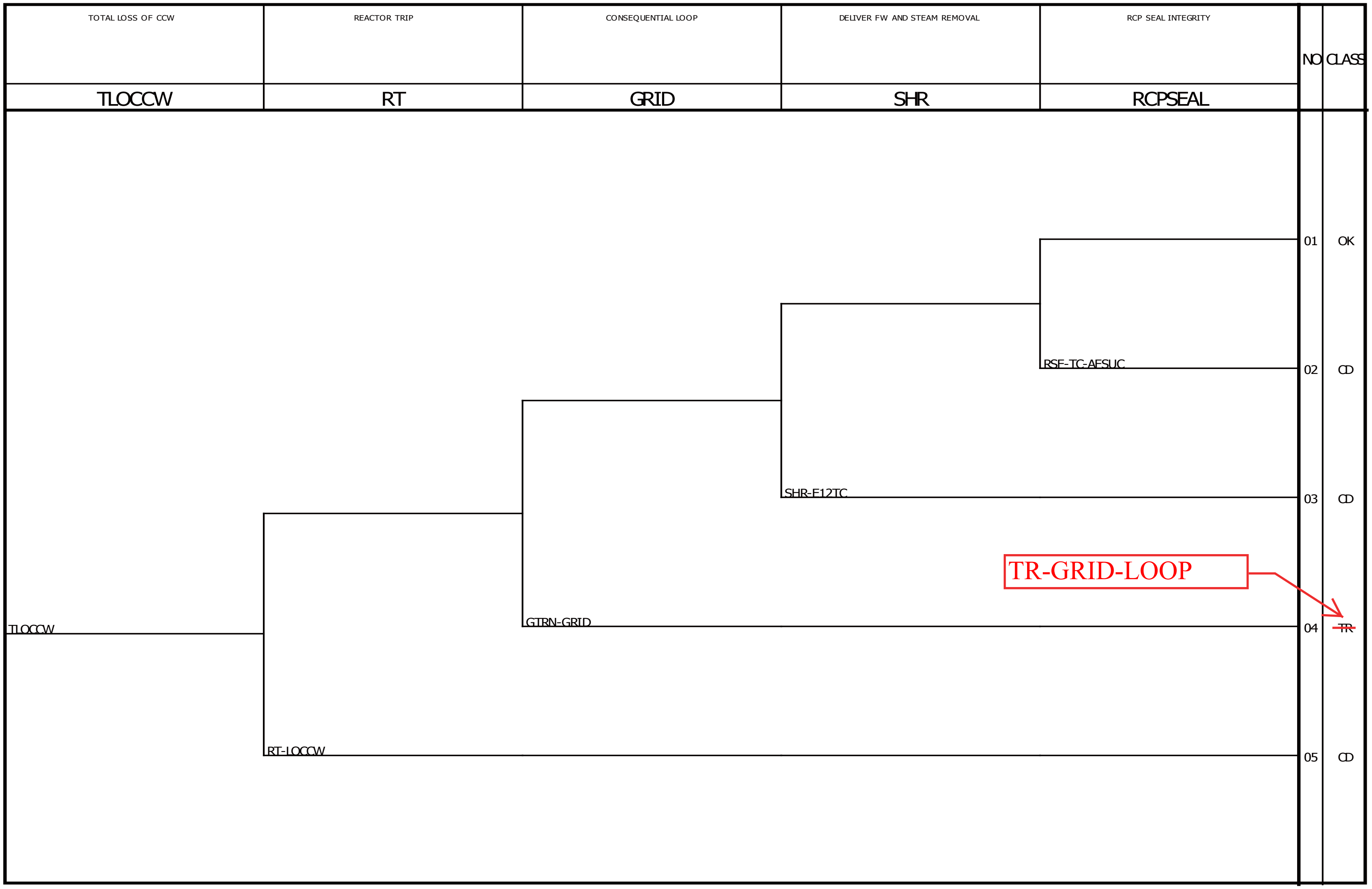

Figure 19.1-36 Level 1 Event Tree - Total Loss of CCW (TLOCCW)

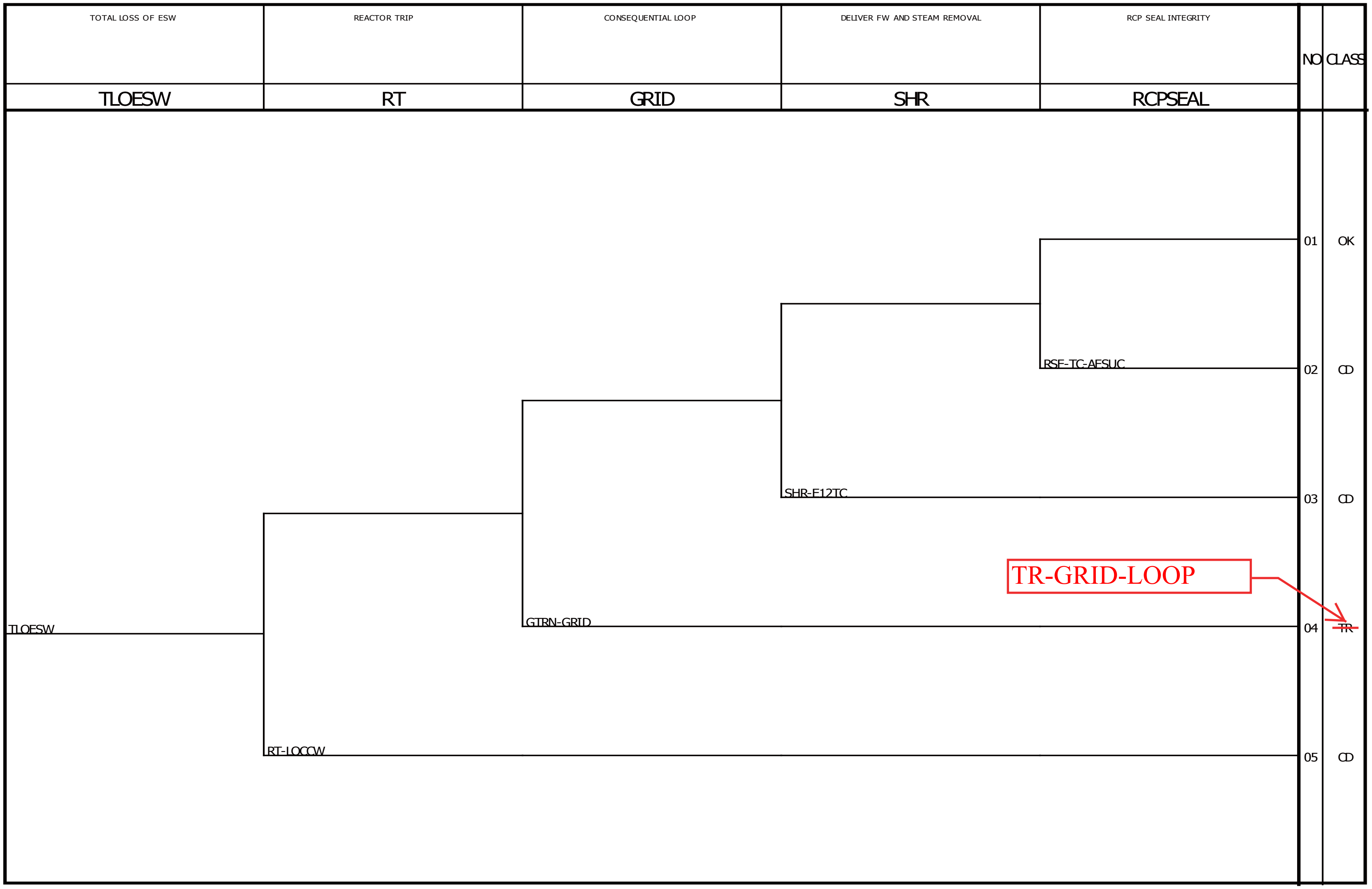

Figure 19.1-38 Level 1 Event Tree - Total Loss of ESW (TLOESW)

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The staff reviewed the APR1400 design control document (DCD) Section 19.1.4.1.1, “Description of Level 1 Internal Events PRA for Operations at Power,” and found insufficient information describing the accident sequence analysis performed. Specifically, the applicant did not identify and describe dependencies that can impact the ability of mitigating systems to operate and function (ASME/ANS PRA Standard – HLR-AS-B). Therefore, in order for the staff to reach an assurance finding on the conformance to SRP Chapter 19.0 regarding PRA technical adequacy, please revise the DCD accordingly with the information needed.

Response

Note that section 19.1.4.1.1 of the DCD already describes in numerous places the incorporation of dependencies into the mitigating system models. For example:

- Section 19.1.4.1.1.c item 3) states: “Fault trees are constructed for the systems represented in the top functional events in the event trees (the front-line systems) and various systems needed to support these systems (support systems). The system dependencies are explicitly considered.”

- Section 19,1,4,1,1,4 general assumption a. 4) states: “Models capture the impact of dependencies, including support systems …”

- Section 19,1,4,1,1,4 general assumption b. 2) states: “Models include contributions due to random component failures, outages for maintenance and test, support systems …”

In addition, in response to action item, AI 19-39 (PRA-039), the updated DCD Section 19.1.4.1.1.4 under the dependency analysis sub-heading (top of DCD page 19.1-49) to include additional dependency Tables 19.1-11a and 19.1.11b was provided. These two new tables identify the front line system dependencies on support systems, and support system dependencies on other support systems. The Section 19.1.4.1.1.4 of the DCD will be revised as shown in Attachment.

Impact on DCD

The DCD will be revised as stated in the response as shown in 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 Technical/Topical/Environmental Report.

Dependency Analysis

The systems that are included in the systems analysis for internal events are provided in Table 19.1-9. Simplified diagrams of major systems are shown in Figures 19.1-1 through 19.1-14. Tables are provided to summarize the initiator-to-system dependencies.

a. Dependency between Initiating Events and Front Line Systems (Table 19.1-10)

b. Dependency between Initiating Events and Support Systems (Table 19.1-11)

c. Dependency between Front Line System and Supporting Systems (Table 19.1-11a)
d. Dependency between Supporting System and Other Supporting Systems (Table 19.1-11b)

19.1.4.1.1.5 Data Analysis

The purpose of the data analysis task is to tabulate estimates of the failure rates, demand failure probabilities, and unavailability data for basic events in the PRA model. The data developed during this task include:

a. Component unreliability data

b. Component unavailability data due to test and maintenance

c. CCF data

d. Special event data including recovery action failures

For each component type and failure mode identified in the system analysis, the failure rates are extracted from available generic data sources. Potential sources of generic failure data are:

a. NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, "Industry Average Parameter Estimates, 2010 Update."

b. NUREG/CR-5500, Vol. 10, "Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998," U.S. Nuclear Regulatory Commission, November 2001 (Reference 13).