Final ASP Program Analysis – Reject

Accident Sequence Precursor Program – Office of Nuclear Regulatory Research			
Brunswick Steam Electric Plant, Unit 1	Degraded Principal Safety Barrier, Technical Specification Shutdown, and Automatic System Actuation		
Event Date: 3/28/2019	**LER:** 325-2019-002 **IR:** 05000324/2019002	**CCDP =** $1{\times}10^{-5}$	
Plant Type:	General Electric Type 4 Boiling-Water Reactor (BWR) with a Mark I Containment		
Plant Operating Mode (Reactor Power Level):	Mode 1 (100% Reactor Power)		
Analyst: Christopher Hunter	**Reviewer:** Felix Gonzalez	**Contributors:** N/A	

EVENT DETAILS

Event Description. At 2:19 p.m. on March 28, 2019, narrow range reactor water level instrument N004B failed high. In addition, containment drywell pressure and floor drain leakage began to increase. Operators vented the drywell and proceeded with a controlled reactor shutdown in accordance with procedures. The reactor was scrammed by operators at 4:03 p.m.

At 4:54 p.m., the outboard main steam isolation valves (MSIVs) closed automatically due to low condenser vacuum. The MSIVs were previously closed by operators during the plant shutdown to control reactor pressure. The automatic closure actuation occurred while operators were in the process of reopening the MSIVs. In addition, a subsequent reactor protection system actuation occurred due to low reactor water level caused by the increased steam flow experienced when the operators attempted to reopen the MSIVs.

During the shutdown, unidentified reactor coolant system (RCS) leakage was greater than 10 gallons per minute (gpm) for greater than or equal to 15 minutes and an *Unusual Event* was declared. The *Unusual Event* was terminated at 2:59 a.m. on March 29th, when RCS leakage was reduced to less than 10 gpm. Additional information is provided in licensee event report (LER) 325-2019-002 (Ref. 1) and inspection report (IR) 05000324/2019002 (Ref. 2).

Cause. Investigation revealed that a 1-inch cryogenic coupling on the steam side of the reference leg for the N004B reactor level instruments experienced a 360° circumferential break near the center of the coupling. This break opened a path for steam from the reactor to leak into the drywell, which resulted in the initial high reactor water level indications.

Background. A simplified drawing of the reactor water level instrumentation is provided in Figure 1. Note that this figure does not contain all reactor water level instrumentation but focuses on the level instrumentation that provide the majority of high-level system trips and low-level system initiation signals.

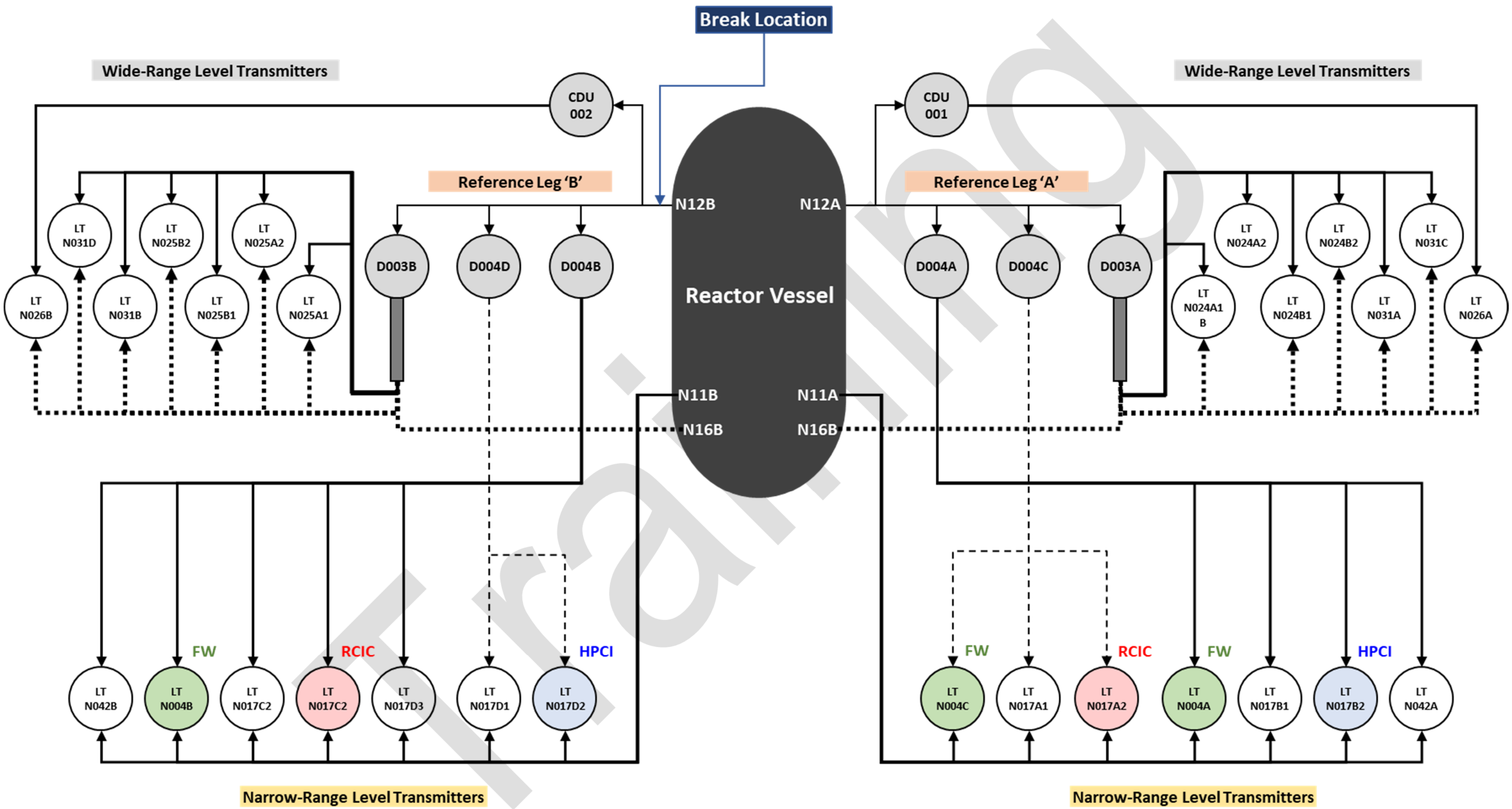

Figure 1. Simplified Drawing of Brunswick (Unit 1) Reactor Water Level Instrumentation

There are two reference legs for the narrow- and wide-range reactor water level instrumentation. These reference legs are supplied from two separate vessel taps (N12A and N12B) above the top of the steam separators within the reactor vessel. The variable legs for the narrow-range instruments are supplied from vessel taps N11A and N11B. The wide-range instruments have variable legs supplied from vessel taps N16A and N16B.

The narrow- and wide-range level transmitters (LTs) detect differential pressure between the reference and variable legs to calculate the reactor water level. The reference leg is designed to maintain a constant hydrostatic pressure equal to the maximum reactor water level.[1] The effect of reactor steam pressure is sensed equally on both the reference and variable legs. Reactor water level is inversely proportional to the differential pressure sensed by the LT (i.e., as differential pressure decreases the reactor water level signal increases and vice versa). A break in the reference leg causes the differential pressure to decrease resulting in an indicated reactor water level higher than actual level.[2]

The narrow-range LTs provide signals to the feedwater (FW) control system, high- and low-water level alarms, low-water level reactor scram, high-water level trip signals to high-pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), the main turbine, and the reactor FW pump turbines. The following narrow-range LTs provide signals for high-level system trips:

- LTs N017B2 and N017D2 2 (out of 2) provide a trip signal to the HPCI system.
- LTs N017A2 and N017C2 (2 out of 2) provide a trip signal to the RCIC system.
- LTs N004A, N004B, and N004C (2 out of 3) provide trip signals to the main turbine and reactor FW pump turbines.

The wide-range LTs provide signals for the automatic initiation of certain systems on a low reactor water level.[3] If the applicable LTs provide erroneous high reactor water level signals, the automatic start of applicable injection systems will not occur on a valid low reactor water level condition. LTs N031A, N031B, N031C, and N031D provide system start signals on low reactor water level for the following systems:[4]

- RCIC
- HPCI
- Low-pressure coolant injection (LPCI)
- Low-pressure coolant spray (LPCS)

During the event on March 28th, the narrow-range LTs connected to the failed reference leg 'B' (N017C2, N017D2, and N004B) provided erroneously high reactor water level signals. The risk

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1 The reference leg is maintained full by the condensing pots. The condensing pots (e.g., D003B, D004B, CDU002) are show as gray shaded circles in Figure 1.

2 Reference leg pressure will significantly decrease due to the loss of reactor steam pressure and potentially lower reference leg level.

3 In addition, the wide-range LTs also provide signals for an isolation of the MSIVs and the starting of the emergency diesel generators on low reactor water level.

4 The high-pressure injection systems (HPCI, RCIC) automatically start on a low level #2 signal, whereas the low-pressure injection systems automatically start on low level #3 signal.

impact of having a single train of LTs failed high is minimal because these failed signals are not enough to cause any system trips.[5,6] However, a system isolation of RCIC and/or HPCI would occur if their opposite train signal also failed high (e.g., reference leg 'A' failure). In addition, the low reactor water level system initiation signals (supplied by the wide-range LTs on reference leg 'B') would be rendered unavailable; however, operators could still manually start the affected systems.

MODELING

SDP Results/Basis for ASP Analysis. The LER associated with this event did not screen into the ASP Program during the initial screening because it did not result in an initiating event (a controlled reactor shutdown occurred) and did not involve a loss of safety function of any equipment. However, due to attention this event was getting in the operating experience community and the rarity of the complete reference line failures, a detailed ASP analysis of this event was performed for training purposes.

Additional LERs were reviewed to determine if concurrent unavailabilities existed during the March 28th event. No windowed events or concurrent degraded operating conditions were identified.

NRC inspectors identified a performance deficiency associated with the licensee failure to ensure material compatibility of the cryogenic coupling used within a hydrogen environment. A detailed risk assessment performed by the Region 2 staff determined that performance deficiency was characterized as *Green* (i.e., very low safety significance) because the failed reference line did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident (LOCA) and did not affect other systems used to mitigate a LOCA resulting in a total loss of their function. The LER remains open. See IR 05000324/2019002 for additional information.

Analysis Type. An initiating event analysis was performed using the test/limited use standardized plant analysis risk (SPAR) model for Brunswick Steam Electric Plant (Unit 1) created on July 18, 2019. This revised model contains added fault tree logic for the high-level trips for RCIC, HPCI, and reactor FW pumps. In addition, this model added basic events to represent (passive) failure of the reference legs, as represented by basic events RCS-ICC-FO-TRAINA (*failure of 'A' division of RPV level instrumentation*) and RCS-ICC-FO-TRAINB (*failure of 'B' division of RPV level instrumentation*).

SPAR Model Modifications. The following modifications were required to properly account for the risk impact of the failed reference leg and postulated failures of opposite train of reactor water level instrumentation:

- The basic events representing the potential failure of the reference legs were added to the RPV-HI-LEVEL (*failure of RPV level channels high*) fault tree. This fault tree models the failure of various injection systems (e.g., RCIC, HPCI, LPCI, LPCS) to automatically start on low reactor water level. Basic event RCS-ICC-FO-TRAINA was inserted under

5 A failure of reference leg 'A' would have resulted in a trip of the main turbine and reactor FW pumps LTs N004A and N004C. Since reference leg 'B' failed, these trips did not occur because only LT N004B was affected.

6 The FW control system was selected to a train 'A' LT and, therefore, was unaffected by the failure of reference leg 'B'. If the FW control system had been selected to a train 'B' LT during this event, FW flow would have been significantly decreased and a subsequent reactor scram would have occurred.

existing ‘OR’ gates LCS-LVL-4 and LCS-LVL-6. Basic event RCS-ICC-FO-TRAINB was inserted under existing ‘OR’ gates LCS-LVL-5 and LCS-LVL-7. The probabilities of these two basic events were set to a screening failure probability of 10^{-3}.[7] This failure probability is a key modeling uncertainty. The modified RPV-LEVEL-HI fault tree is shown in Figure B-1 of Appendix B.

- Given a failure of the automatic start signals for applicable injection systems (e.g., RCIC, HPCI, LPCI, LPCS) on low reactor water level, operators would still have the capability to manually start these systems. To provide this credit, existing fault tree RPV-LEVEL-HI and a new basic event, ECCS-XHE-XM-MANUAL (*operators fail to manually start high- or low-pressure injection systems*) was inserted under new ‘AND’ gates for the applicable injection system fault trees. The specific faults trees modified include: HC1, HCI-B, HCI-HW, LCI-A, LCI-B, LCS-A, LCS-B, RCI-B, and RCI-HW. The probability of ECCS-XHE-XM-MANUAL was set to 10^{-2}. These modified fault trees are shown in Appendix B (Figures B-2 through B-10).

Key Modeling Assumptions. The following modeling assumptions were determined to be significant to the modeling of this initiating event assessment:

- The probability of IE-LOCHS (*loss of condenser heat sink*) was set to 1.0. This is potentially conservative because operators performed a controlled reactor shutdown and subsequently closed the MSIVs to control reactor pressure in response to the failed reference leg.[8] All other initiating event probabilities were set to zero.
  - The operators manually closed the MSIVs to control the RCS cooldown rate and maintain reactor pressure. In addition, there was an automatic closure actuation while the MSIVs were closed due to lowering condenser vacuum. However, the condenser heat sink, including required support systems (e.g., circulating water, condensate), remained available throughout the event. Therefore, credit for recovery of the condenser heat sink is warranted. PCS-XHE-XL-LOCHS (*power conversion system recovery fails during LOCHS*) was set to a screening value of 0.1.[9, 10]
- Basic event RCS-ICC-FO-TRAINB was set to TRUE to represent the failure of reference leg ‘B’.

7 EGG-SSRE-8875, “Generic Component Failure Data Base for Light Water and Liquid Sodium Reactor PRAs” provides a failure rate of 10^{-8} (foot-hour) for 1- to 3-inch piping. The exact length of each reference legs is not known. However, using this failure rate for a 24-hour mission time, the failure probability of the reference leg would not exceed 10^{-3} unless the reference legs were at least 4,200 feet long. Therefore, the use of the 10^{-3} screening value is very likely conservative. However, note that this probability does not account for any potential increase in failure likelihood of the cryogenic pipe couplings on reference leg ‘A’.

8 The Brunswick SPAR model does not have an event tree for manual reactor shutdown. For the SPAR models that do, the difference between it and general transient tree is that the potential for an anticipated transient without scram (ATWS) is eliminated for the manual reactor shutdown scenario. Note that ATWS has a negligible contribution to the overall results for this analysis.

9 NUREG-1792, “Good Practices for Implementing Human Reliability Analysis,” provides that 0.1 is an appropriate screening (i.e., typically conservative) value for most post-initiator human failure events (HFEs).

10 This credit for recovery of the condenser heat sink is only provided for long-term containment temperature and pressure control. It does not provide credit for restoration of the condenser (and feedwater) given the failure of high-pressure injection systems (e.g., HPCI and RCIC) and the failure of reactor depressurization, which is a potentially conservative assumption for the analysis of this event.

ANALYSIS RESULTS

CCDP. The conditional core damage probability (CCDP) for this analysis is calculated to be $1.37{\times}10^{-5}$. The ASP Program precursor threshold is a CCDP of $1{\times}10^{-6}$ or the CCDP equivalent of an uncomplicated reactor trip with a non-recoverable loss of feed water or the condenser heat sink), whichever is greater. This CCDP equivalent for Brunswick Steam Electric Plant (Unit 1) is $1.42{\times}10^{-5}$.[11] Therefore, this event is a not a precursor.

The relatively high CCDP for this event is driven by the risk of a loss of condenser heat sink. The CCDPs for losses of condenser heat sink typically range in 10^{-6} to low 10^{-5} range for most BWRs. The CCDP contribution associated with the failure of reference leg 'B' is approximately $8{\times}10^{-7}$ (approximately 6 percent). The following are considered key modeling uncertainties for this analysis:

- *The lack of a refined failure probability estimate for the postulated failure of reference leg 'A'.* Typically, failure of passive components such as instrument lines are not included in most PRAs. The failure probability used in this analysis is a screening estimate that is believed to be conservative given the likely (but unknown) length of the instrument line. However, an additional unknown is whether the mechanism that failed cryogenic pipe coupling on reference leg 'B' would significantly increase the failure likelihood of a cryogenic pipe coupling on reference leg 'A'. Overall, the screening value used for the failure of reference leg 'A' is believed to be conservative.
- *The treatment of recovery of the condenser heat sink.* Although credit for condenser heat sink is provided in this analysis, its only effect is on containment temperature/pressure control. If operators could recover the condenser heat sink and reactor FW in the short term (i.e., less than 1 hour), the CCDP for this event would be significantly reduced. There is some potential this was possible and, therefore, the limited recovery credit is potentially conservative.
- *The potential that erroneous reactor water level would adversely affect operator performance.* For the postulated event involving the failure of both reference legs, all narrow- and wide-range reactor water level indications would indicate high. In addition, both HPCI and RCIC would be rendered unavailable due to actuation of their high-level trips. If this occurred, operators would need to depressurize the reactor and manually start the low-pressure injection systems (LPCI or LPCS) to provide inventory makeup to the reactor. There is some concern that the nominal human error probability (HEP) used in the SPAR model may be overly optimistic in this scenario. However, a review of procedures indicates that the operators are directed to manually depressurize and flood the reactor when reactor water level cannot be accurately determined. Given the unambiguous language of these procedures, it is not believed that the failure of both reference legs would result in a significant increase in the nominal HEP for operators failing to depressurize the reactor with HPCI and RCIC are unavailable. Therefore, the HEP was kept at its nominal value. In addition, the use of the 10^{-2} probability for operators failing to manually start low-pressure injection systems is believed acceptable for this postulated event.

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[11] For BWRs, a loss of condenser heat sink initiating event typically assumes that the condensate system is available to provide a source of low-pressure injection to the reactor.

Dominant Sequence. The dominant accident sequence is loss of condenser heat sink sequence 46 (CCDP = 1.29×10^{-5}), which contributes approximately 94 percent of the total internal events CCDP. The dominant sequences that contribute at least 1.0 percent to the total internal events CCDP are provided in the following table. The event tree with the dominant sequences is shown graphically in Figure A-1 of Appendix A.

Sequence	CCDP	Percentage	Description
LOCHS 46	1.29×10^{-5}	94.3%	Loss of condenser heat sink initiating event occurs; successful reactor trip; RCIC and HPCI fail; and reactor depressurization fails resulting in core damage
LOCHS 25	2.97×10^{-7}	2.2%	Loss of condenser heat sink initiating event occurs; successful reactor trip; RCIC and/or HPCI are successful; suppression pool cooling fails; and reactor depressurization fails resulting in core damage
LOCHS 9	1.43×10^{-7}	1.1%	Loss of condenser heat sink initiating event occurs; successful reactor trip; RCIC and/or HPCI are successful; suppression pool cooling fails; reactor depressurization succeeds; condensate injection is successful; shutdown cooling fails; operators fail to recover the condenser heat sink; containment spray and venting fail resulting in core damage

REFERENCES

1. Brunswick Steam Electric Plant (Unit 1), "LER 325-2019-002 – Degraded Principal Safety Barrier, Technical Specification Shutdown, and Automatic System Actuation," dated May 23, 2019 (ADAMS Accession No. ML19143A375).

2. U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant – Integrated Inspection Report 05000324/2019002 and 05000325/2019002; 07200006/2019001," dated August 9, 2019 (ADAMS Accession No. ML19221B744).

Appendix A: Key Event Tree

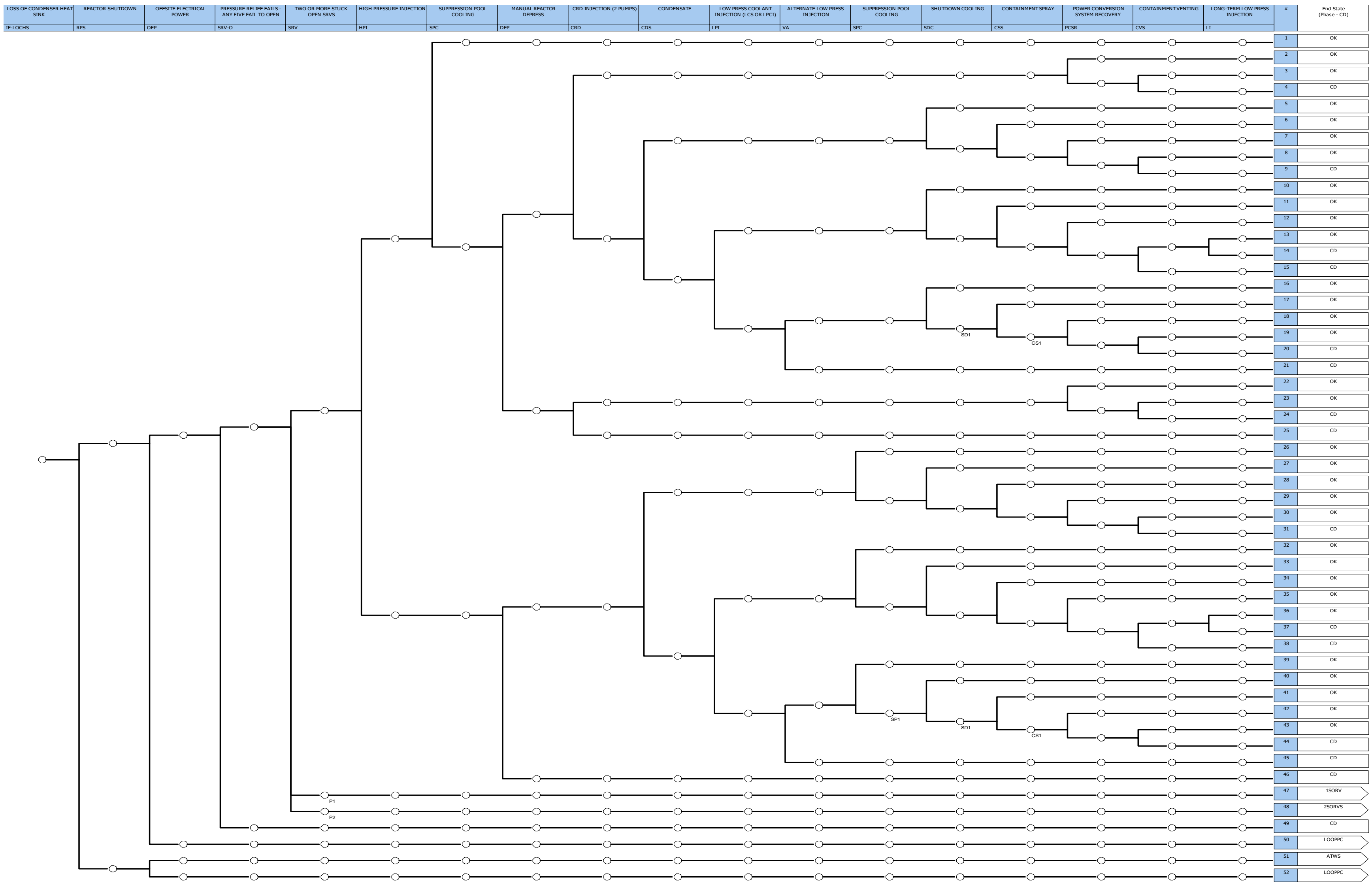

Figure A-1. Brunswick Loss of Condenser Heat Sink Event Tree

Appendix B: Modified Fault Trees

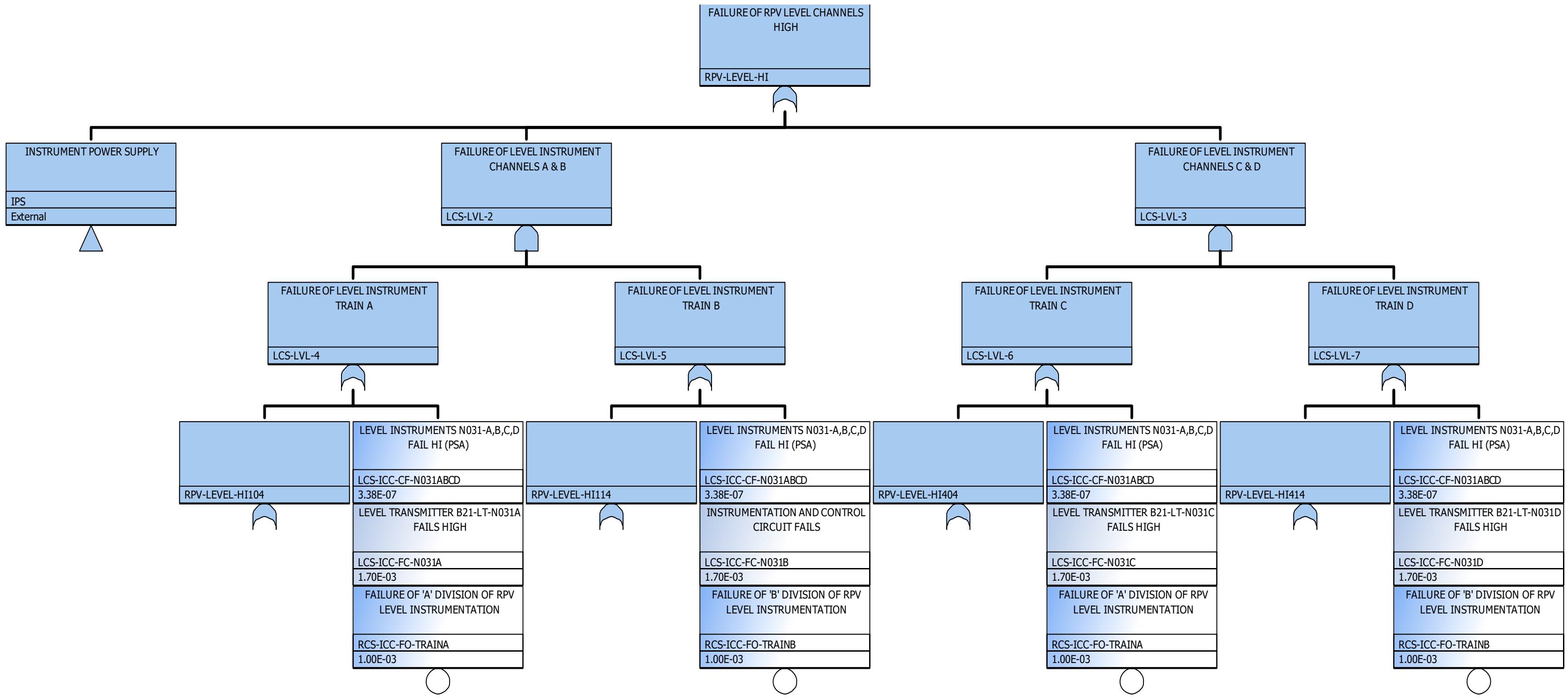

Figure B-1. Modified RPV-HI-LEVEL Fault Tree

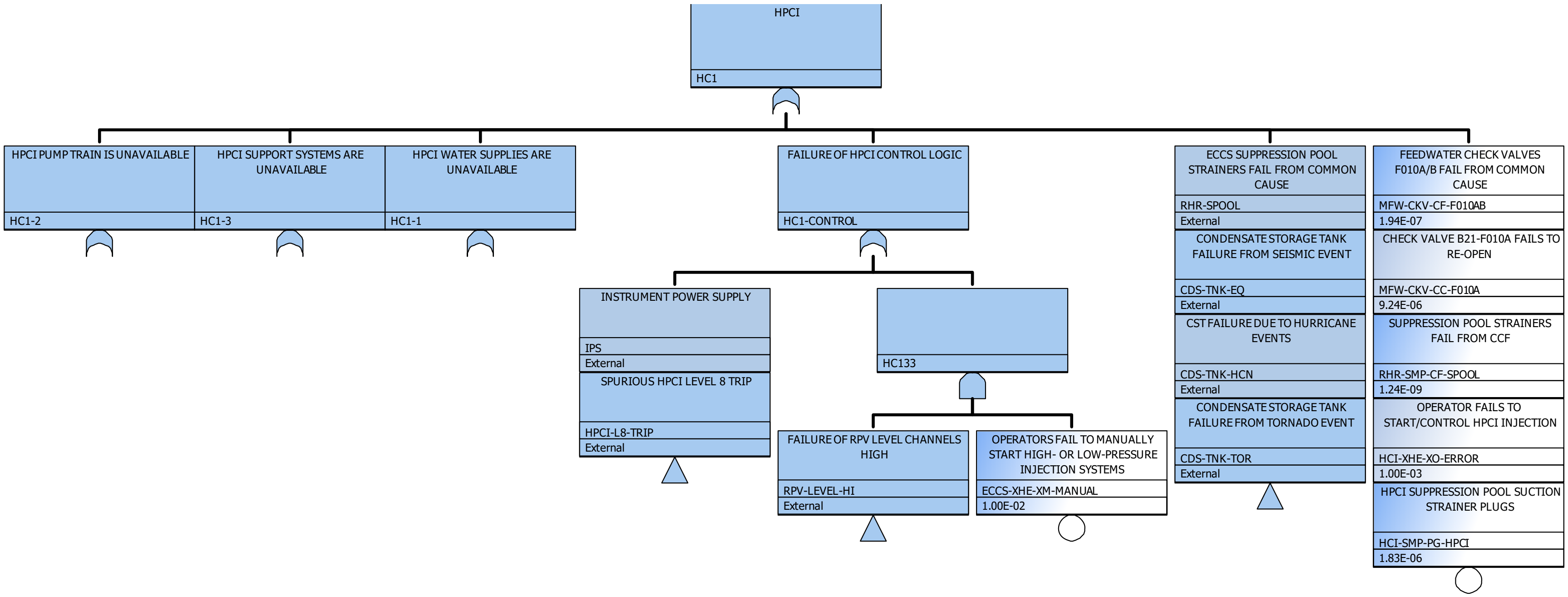

Figure B-2. Modified HC1 Fault Tree

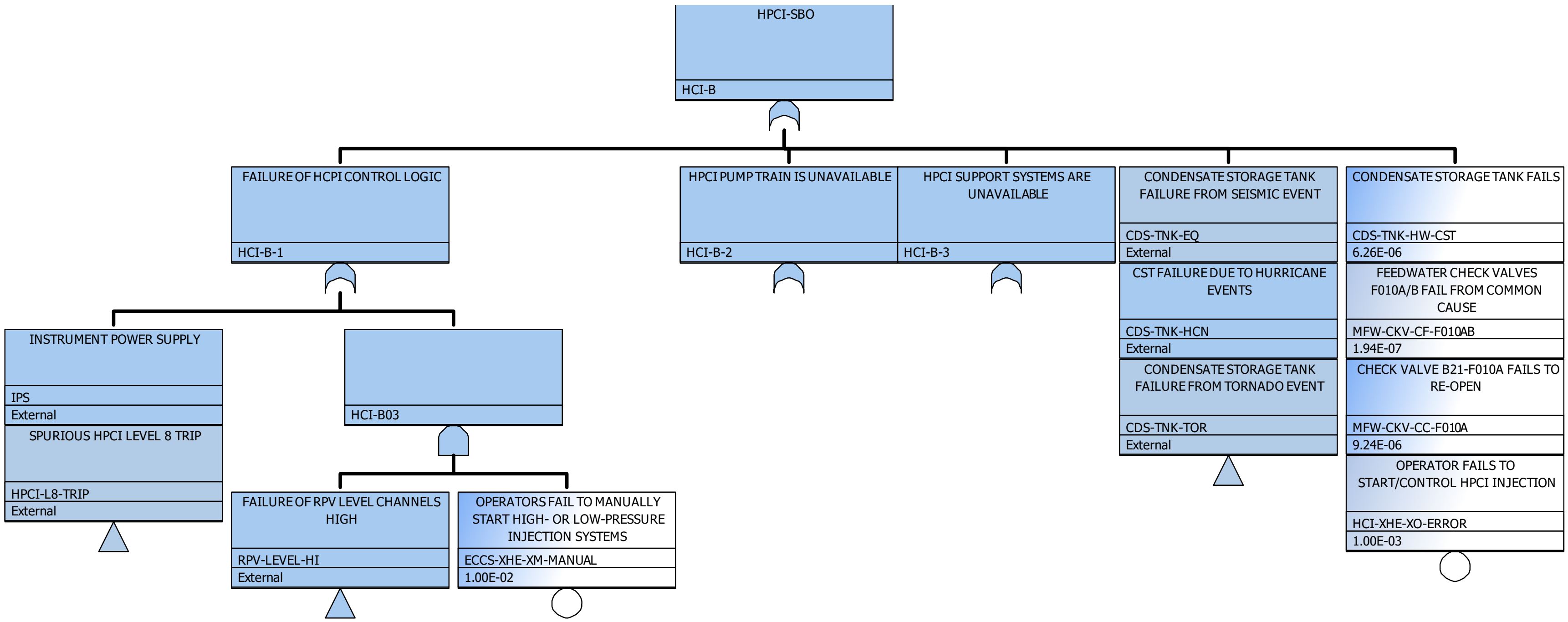

Figure B-3. Modified HCI-B Fault Tree

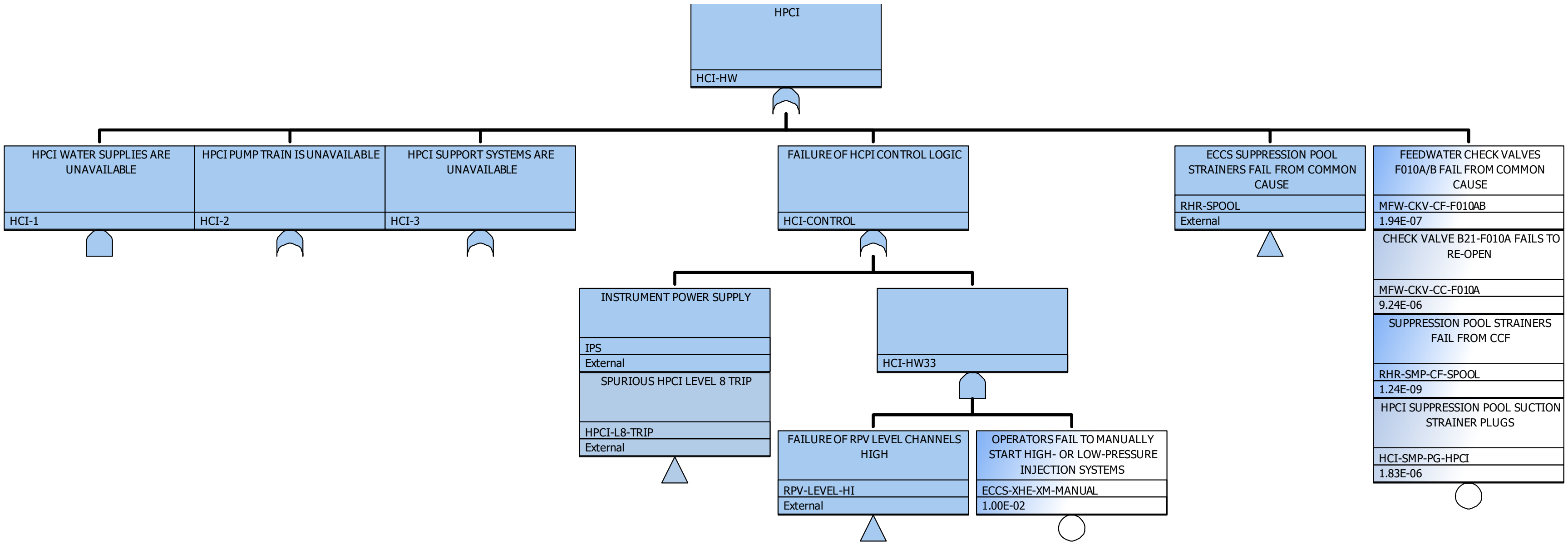

Figure B-4. Modified HCI-HW Fault Tree

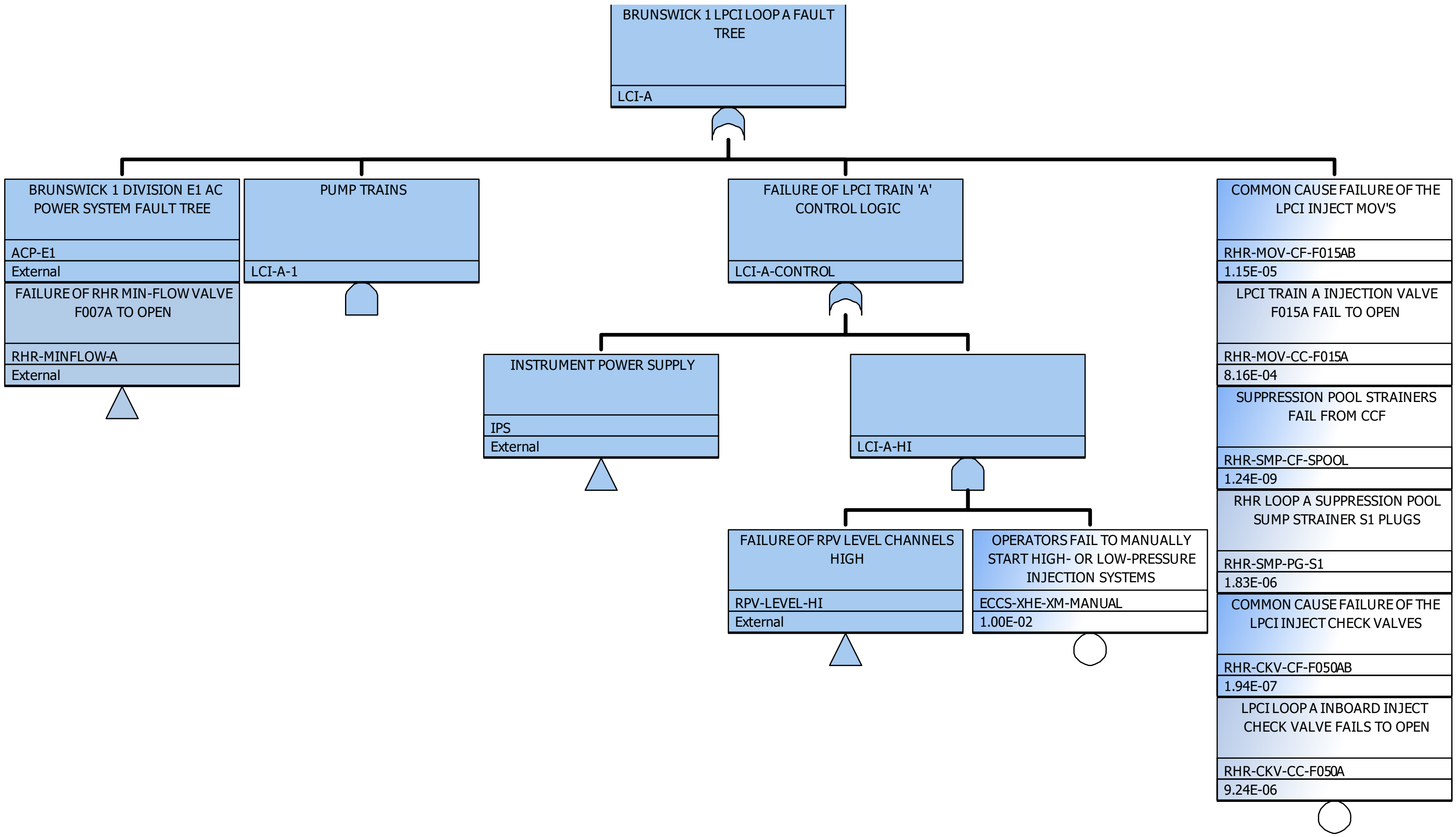

Figure B-5. Modified LCI-A Fault Tree

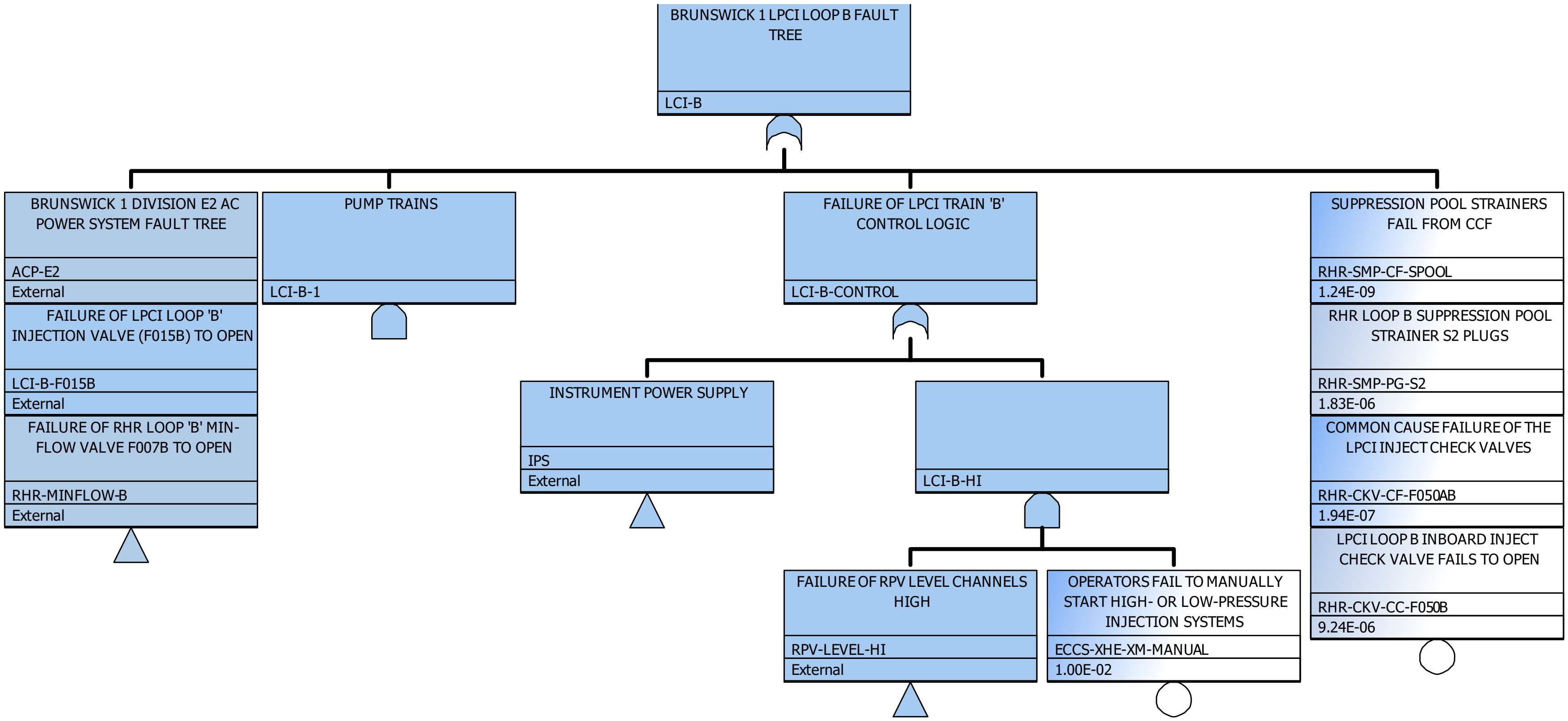

Figure B-6. Modified LCI-B Fault Tree

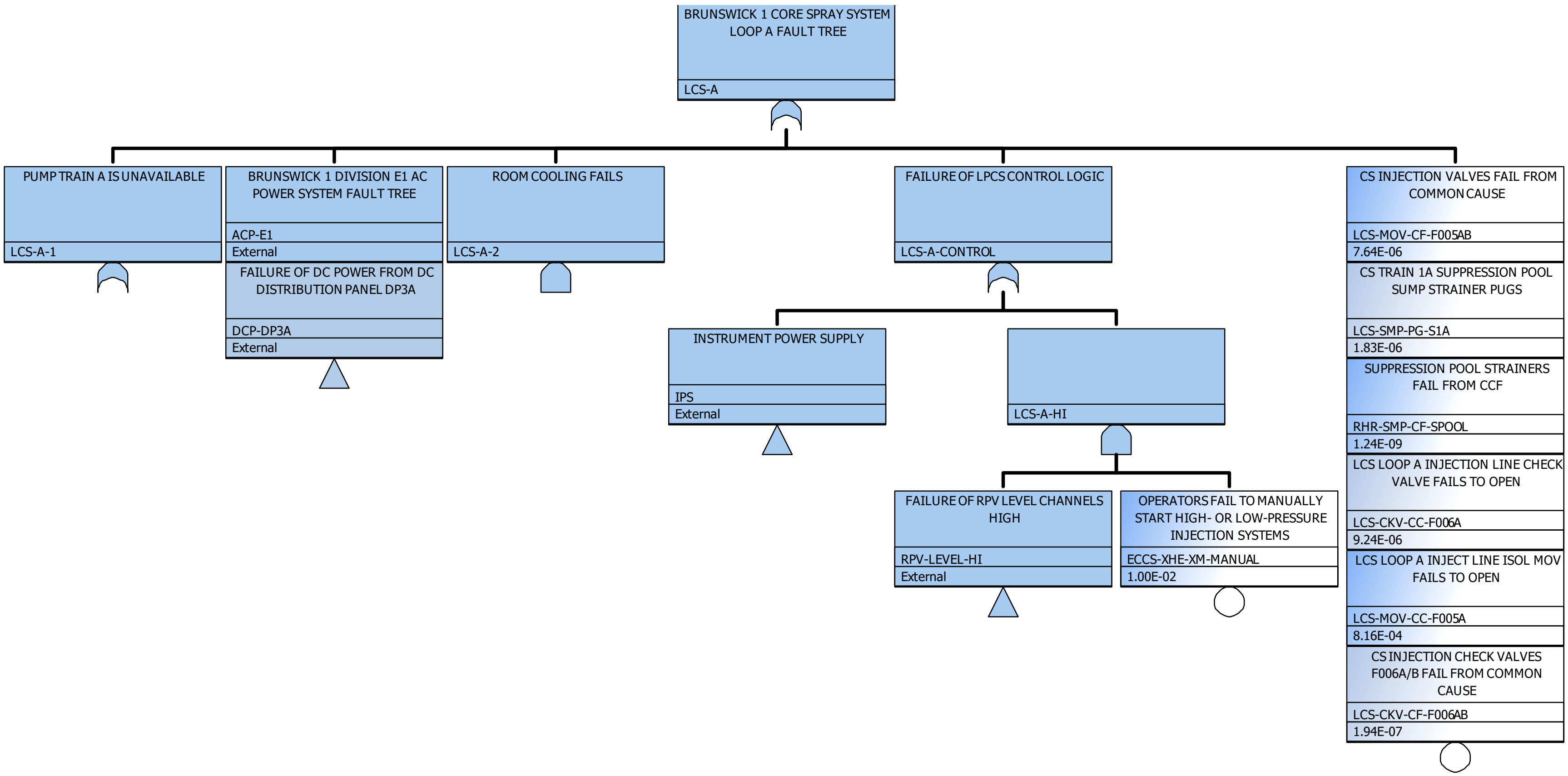

Figure B-7. Modified LCS-A Fault Tree

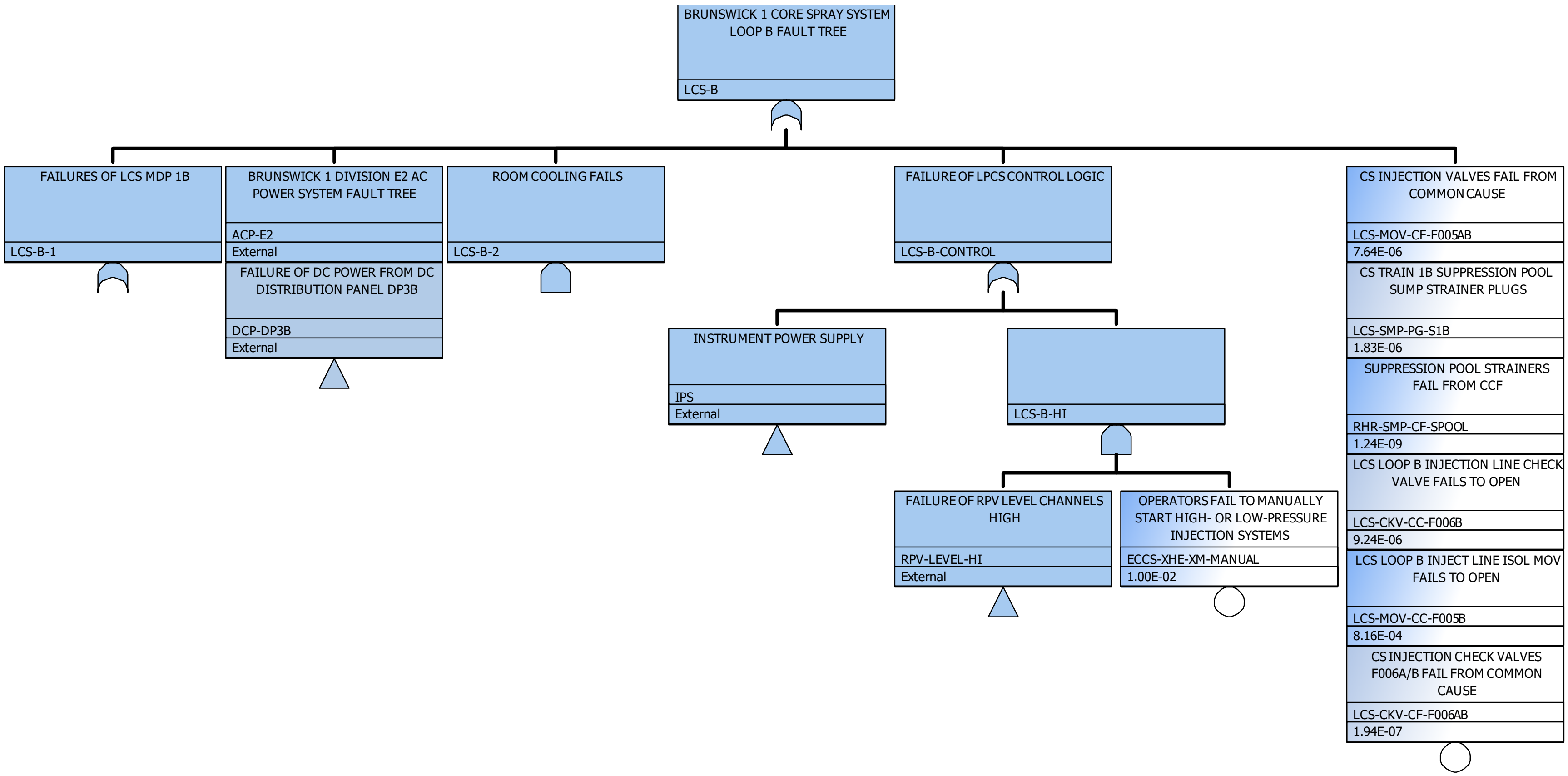

Figure B-8. Modified LCS-B Fault Tree

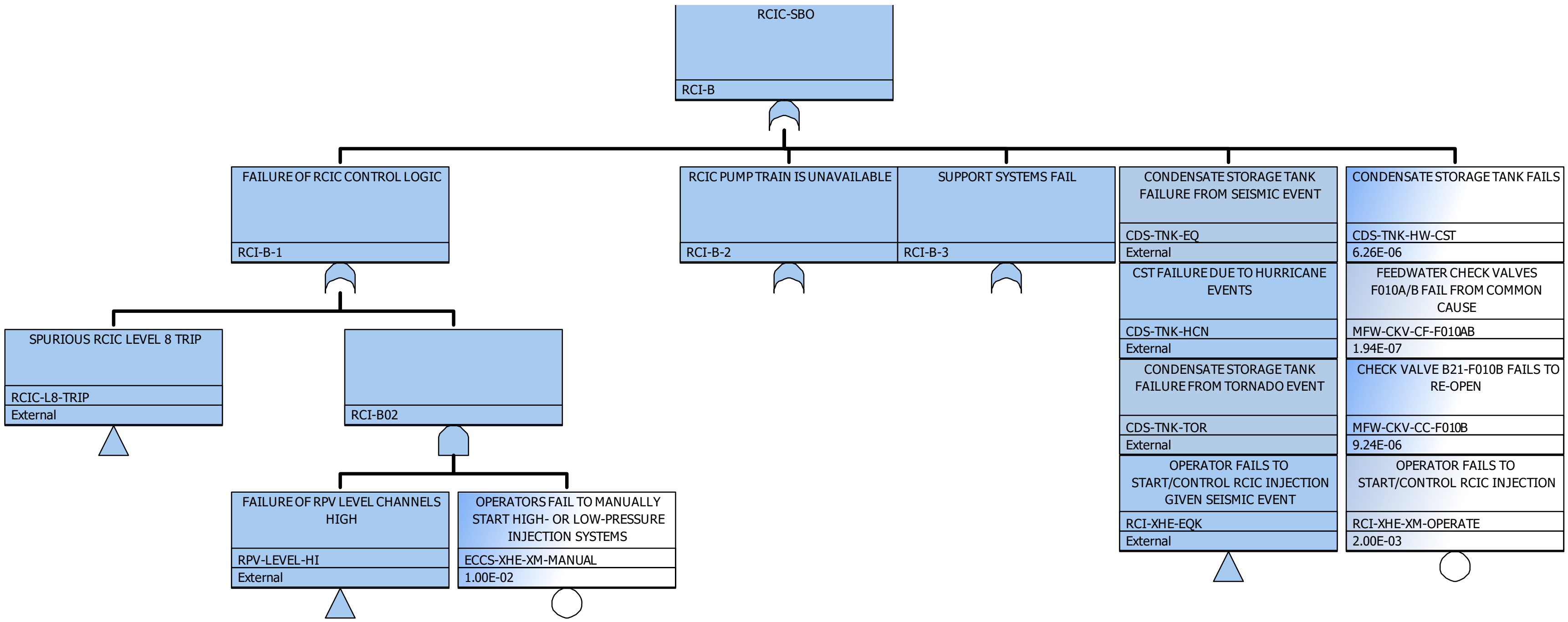

Figure B-9. Modified RCI-B Fault Tree

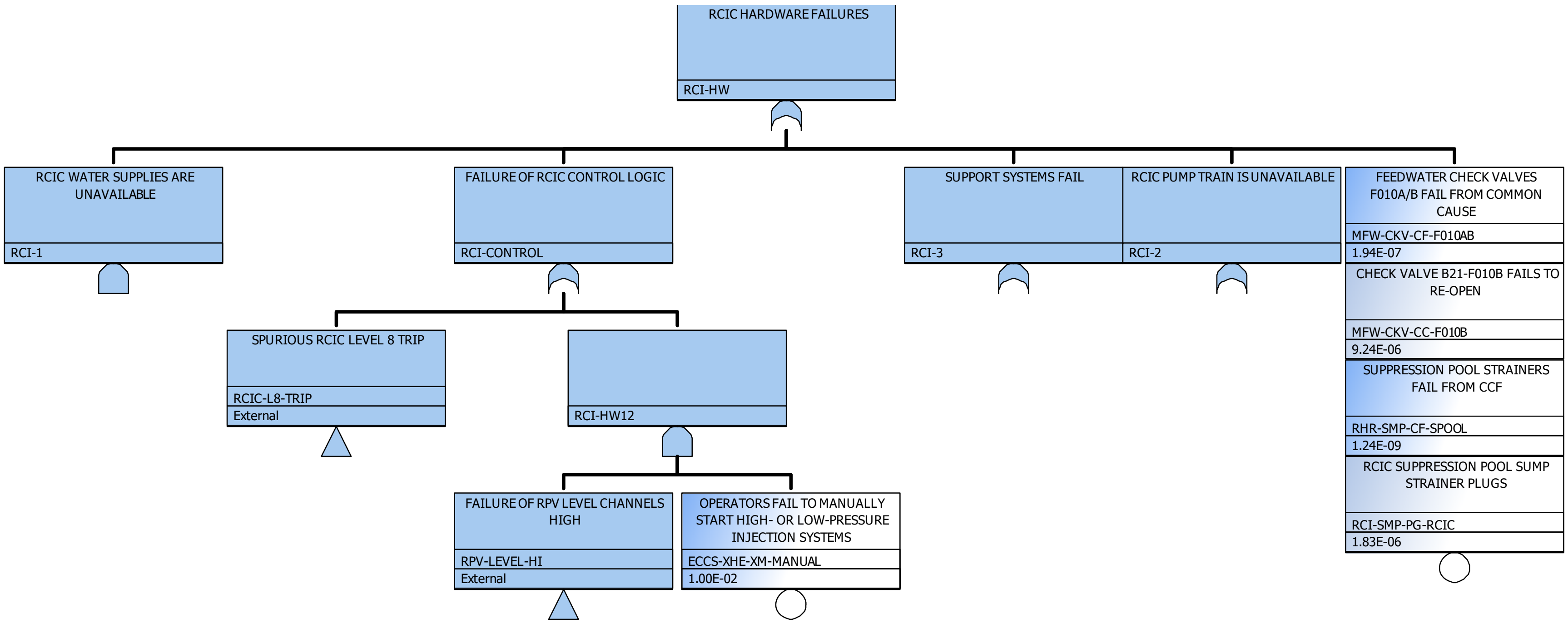

Figure B-10. Modified RCI-HW Fault Tree