UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 22, 2017

Mr. Tom Simril
Site Vice President
Catawba Nuclear Station, Unit 2
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: TRANSMITTAL OF FINAL CATAWBA NUCLEAR STATION, UNIT 1 ACCIDENT SEQUENCE PRECURSOR REPORT (LICENSEE EVENT REPORT 413-2016-001) (EPID L-LRO-2017-0057)

Dear Mr. Simril:

By letter dated June 23, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16179A207), Catawba Nuclear Station, Unit 1 submitted licensee event report (LER) 413-2016-001 to the U.S. Nuclear Regulatory Commission (NRC) staff pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.73. As part of the Accident Sequence Precursor (ASP) Program, the NRC staff reviewed the event to identify potential precursors and to determine the probability of the event leading to a core damage state. The results of the analysis are provided in the enclosure to this letter.

The NRC does not request a formal analysis review, in accordance with Regulatory Issue Summary 2006-24, "Revised Review and Transmittal Process for Accident Sequence Precursor Analyses" (ADAMS Accession No. ML060900007), because the analysis resulted in an increase in core damage probability (ΔCDP) of less than 1×10^{-4}.

Final ASP Analysis Summary. A brief summary of the final ASP analysis, including the results, is provided below.

Mis-Positioned Breaker with Concurrent Emergency Diesel Generator Unavailability Results in Potential Loss of Recirculation Capability. This event is documented in LER 413-2016-001 and IR 05000413/2016002.

Executive Summary. On March 28, 2016, while performing an emergency core cooling system (ECCS) interlock test, operators determined that breaker 1EMXD-F02A for the residual heat removal (RHR) pump B hot-leg suction valve 1ND-36B was open. Breaker 1EMXD-F02A is required to be closed in Modes 1–3 of plant operation to provide an interlock with the RHR supply valve to safety injection pump B (1NI-136B), which operators must manually open for high-pressure, cold-leg recirculation. An investigation determined that the last operation of 1EMXD-F02A was during the previous refueling outage (December 2015). During this exposure period, emergency diesel generator (EDG) A was declared inoperable for approximately 51 hours. An alternate interlock for 1NI-136B is provided by the other RHR pump B hot-leg suction valve 1ND-37A; however, this redundant signal could be lost during a postulated loss-of-offsite power (LOOP) and unavailability of EDG A. Therefore, if high-pressure, cold-leg recirculation were needed to mitigate a postulated a loss-of-coolant accident or provide sustained feed and cooling during a concurrent LOOP with EDG A unavailability, ECCS train B would be unavailable.

According to the risk analysis modeling assumptions used in this ASP analysis, the most likely core damage scenario is a loss of 4.16kV bus A with subsequent failure of auxiliary feedwater (AFW), successful initiation of feed and bleed cooling, and failure of high-pressure recirculation (for sustained operation). In addition to the unavailability of the ECCS recirculation train B, this scenario's risk significance is largely attributed to Catawba's AFW system design of a single motor-driven (train B) and single turbine driven pump (train A).

A *Green* finding (i.e., very low safety significance) was identified due to the licensee's failure to adequately implement procedures for operation of the RHR system. This licensee's performance deficiency led to breaker 1EMXD-F02A being left open during plant startup, resulting in ECCS train B inoperability for greater than the allowed outage time per plant technical specifications. Risk assessments performed as part of the Significance Determination Process are limited to the analysis of individual performance deficiencies. An independent ASP analysis is required because EDG A was unavailable concurrently with breaker 1EMXD-F02A being open.

Summary of Analysis Results. This operational event resulted in a best estimate ΔCDP of 1×10^{-6}. The detailed ASP analysis can be found in the enclosure.

If you have any questions, please contact me at 301-415-3867 or via e-mail at Michael.Mahoney@nrc.gov.

Sincerely,

Michael Mahoney, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413

Enclosure:
Mis-Positioned Breaker with Concurrent
Emergency Diesel Generator Unavailability
Results in Potential Loss of Recirculation Capability

cc: ListServ

ENCLOSURE

Mis-Positioned Breaker with Concurrent Emergency Diesel Generator Unavailability Results in Potential Loss of Recirculation Capability

Final Precursor Analysis

Accident Sequence Precursor Program – Office of Nuclear Regulatory Research			
Catawba Nuclear Station, Unit 1	Mis-Positioned Breaker with Concurrent Emergency Diesel Generator Unavailability Results in Potential Loss of Recirculation Capability		
Event Date: 3/28/2016	**LERs:** 413-2016-001 **IRs:** 05000413/2016002		**ΔCDP** = 1×10^{-6}
Plant Type: Pressurized-Water Reactor (PWR); Westinghouse Four-Loop with a Wet, Ice Condenser Containment			
Plant Operating Mode (Reactor Power Level): Mode 1 (100% Reactor Power)			
Analyst: Christopher Hunter	**Reviewer:** Ian Gifford	**Contributors:** N/A	**BC Review Date:** 1/18/2017

EXECUTIVE SUMMARY

On March 28, 2016, while performing an emergency core cooling system (ECCS) interlock test, operators determined that breaker 1EMXD-F02A for the residual heat removal (RHR) pump B hot-leg suction valve 1ND-36B was open. Breaker 1EMXD-F02A is required to be closed in Modes 1–3 of plant operation to provide an interlock with the RHR supply valve to safety injection (SI) pump B (1NI-136B), which operators must manually open for high-pressure, cold leg-recirculation. An investigation determined that the last operation of 1EMXD-F02A was during the previous refueling outage (December 2015). During this exposure period, emergency diesel generator (EDG) A was declared inoperable for approximately 51 hours. An alternate interlock for 1NI-136B is provided by the other RHR pump B hot-leg suction valve 1ND-37A; however, this redundant signal could be lost during a postulated loss of offsite power (LOOP) and unavailability of EDG A. Therefore, if high-pressure, cold-leg recirculation were needed to mitigate a postulated a loss-of-coolant accident (LOCA) or provide sustained feed and cooling during a concurrent LOOP with EDG A unavailability, ECCS train B would be unavailable.

According to the risk analysis modeling assumptions used in this Accident Sequence Precursor (ASP) analysis, the most likely core damage scenario is a loss of 4.16kV bus A with subsequent failure of auxiliary feedwater (AFW), successful initiation of feed and bleed cooling, and failure of high-pressure recirculation (for sustained operation). In addition to the unavailability of the ECCS recirculation train B, this scenario's risk significance is largely attributed to Catawba's AFW system design of a single motor-driven (train B) and single turbine-driven pump (train A).

A *Green* finding (i.e., very low safety significance) was identified due to the licensee failure to adequately implement procedures for operation of the RHR system. This licensee performance deficiency led to breaker 1EMXD-F02A being left open during plant startup, resulting in ECCS train B inoperability for greater than the allowed outage time per plant technical specifications. Risk assessments performed as part of the Significance Determination Process (SDP) are limited to the analysis of individual performance deficiencies. An independent ASP analysis is required because EDG A was unavailable concurrently with breaker 1EMXD-F02A being open.

EVENT DETAILS

Event Description. On March 28, 2016, while performing an interlock test on ECCS cold-leg recirculation train 1B, operators determined that that breaker 1EMXD-F02A for the RHR pump B hot-leg suction valve 1ND-36B was open. Breaker 1EMXD-F02A is required to be closed in Modes 1–3 of plant operation to provide an interlock with the RHR supply valve to SI pump B (1NI-136B), which operators must manually open for high-pressure, cold-leg recirculation (i.e., "piggyback" mode of recirculation). Therefore, ECCS train 1B was declared inoperable. An investigation determined that the last operation of 1EMXD-F02A was during the previous refueling outage on December 13, 2015.

During this exposure period, EDG 1A was declared inoperable on five separate occasions for a total of approximately 51 hours. With breaker 1EMXD-F02A opened and EDG 1A unavailable to perform its safety function, cold-leg recirculation could be rendered unavailable during a postulated LOOP with concurrent LOCA.

Additional information is provided in Licensee Event Report (LER) 413/2016-001 (Ref.1) and Inspection Report (IR) 05000413/2016002 (Ref. 2).

Cause. The test procedure for pressure boundary valve testing did not contain specific guidance for establishing a suction source for RHR pump 1B; therefore, coordination was required with the standby readiness alignment procedure. The lack of specific procedural guidance allowed breaker 1EMXD-F02A to remain open. In addition, there was ineffective coordination by licensee personnel while performing both procedures. Note that there is no control room indication for the position of breaker 1EMXD-F02A.

Simplified RHR System Drawing. A simplified drawing of the Catawba, Unit 1 RHR systems drawing is shown in Figure B-1 in Appendix B.

MODELING ASSUMPTIONS

Analysis Type. The Catawba Standardized Plant Analysis Risk (SPAR) Model Version 8.20 dated May 20, 2014, was used for this event analysis. A condition assessment was performed for the mis-positioned breaker (1EMXD-F02A) and the concurrent unavailability of EDG A due to maintenance.

SDP Results/Basis for ASP Analysis. The ASP Program uses SDP results for degraded conditions when available (and applicable). A *Green* finding (i.e., very low safety significance) was identified (see IR 05000413/2016002 for additional information). This finding was due to the licensee failure to adequately implement procedures for operation of the RHR system as required by Section 3 of Regulatory Guide 1.33 (Ref. 3). Specifically the licensee failed to align the RHR system for standby readiness prior to plant startup. This resulted in breaker 1EMXD-F02A being left open during plant startup and ECCS train 1B being inoperable for greater than the allowed outage time per plant technical specifications. However, a redundant signal from the RHR pump B hot-leg suction valve 1ND-37A was available to provide the required permissive signal to open 1ND-136B; therefore, the SDP evaluation determined that there was no loss of safety function of ECCS train B. The LER was closed in IR 05000413/2016002.

A search for additional Catawba (Unit 1) LERs was performed to determine if additional unavailabilities existed during the exposure period of the mis-positioned breaker. There have been no additional LERs submitted by Catawba Nuclear Station.

An independent ASP analysis is required because EDG A was unavailable concurrently with breaker 1EMXD-F02A being open. Risk assessments performed as part of the SDP are limited to the analysis of individual performance deficiencies and, therefore, the analysis of the inspection finding associated with the mis-positioned breaker did not factor the concurrent unavailability of EDG A.

SPAR Model Modifications. The following modifications were required for this condition assessment:

- The applicable EDG recovery basic events EPS-XHE-XL-NR01H (*operator fails to recover emergency diesel in 1 hour*), EPS-XHE-XL-NR02H (*operator fails to recover emergency diesel in 2 hours*), and EPS-XHE-XL-NR04H (*operator fails to recover emergency diesel in 4 hours*) were set to TRUE in the base SPAR model. These basic events are set to TRUE for applicable ASP condition assessments and their use is limited to cases where event information supports credit for EDG recovery.

- The current SPAR model for Catawba Nuclear Station does not include fault tree logic for the interlocks between the RHR hot-leg suction valves (1ND-1B, 1ND-2A, 1ND-36B, and 1ND-37A) and the RHR supply valves to the SI pumps (1ND-28A and 1NI-136B).[1] To model the breaker dependency of these interlocks, the HPR-RHRA (*RHR train A flow path*) and HPR-RHRB (*RHR train B flow path*) fault trees were modified. In the HPR-RHRA fault tree, a new AND gate HPR-RHRA-2 (*ND-28A fails to open due to hot leg suction valve interlocks*) was inserted under existing OR gate HPR-RHRA-1 (ND28A *fails to open*). Two new OR gates HPR-INTERLOCK-ND1B (*interlock from ND-1B fails*) and HPR-INTERLOCK-ND2A (*interlock from ND-2A fails*) were inserted under gate HPR-RHRA-2. To represent the potential unavailability of the interlock signals from valves 1ND-1B and 1ND-2A to 1ND-28A due to mis-positioned valve breakers, new basic events HPR-CRB-ND1B (*breaker for ND-1B is open*) and HPR-CRB-ND2A (*breaker for ND-2A is open*) were inserted under gates HPR-INTERLOCK-ND1B and HPR-INTERLOCK-ND2A, respectively. These basic events were set to a screening value of 1×10^{-2}.[2] In addition to the mis-positioned breakers causing the unavailability of the interlock signals, a loss of the associated safety-related AC electrical power would render the interlock signal unavailable. To model this electrical dependency, existing fault tree transfers ACP-1ETA (*division 1A AC power bus 1ETA fails*) and ACP-1ETB (*division 1B AC power bus 1ETB fails*) were inserted under HPR-INTERLOCK-ND2A and HPR-INTERLOCK-ND1B, respectively. The modified HPR-RHRA fault tree is shown in Figure C-1. Similar changes were made to the HPR-RHRB fault tree (see Figure C-2).

Exposure Periods. The breaker was opened during the nightshift on December 13, 2015. The plant transitioned to Mode 3 on December 15th at 1:52 p.m. The breaker was discovered to be open on March 28, 2016, and was subsequently closed at 12:01 p.m. During this period,

[1] Valves 1ND-1B and 1ND-2A provide the interlock signal to valve 1ND-28A, while valves 1ND-36B and 1ND-37A provide the interlock signal to valve 1NI-136B. Only one operable interlock signal (per valve) is needed to allow the manual operation of 1ND-28A and 1NI-136B.

[2] NUREG-1792, "Good Practices for Implementing Human Reliability Analysis," provides that 1×10^{-2} is an appropriate screening (i.e., typically conservative) value for most pre-initiator HFEs.

EDG 1A was concurrently unavailable for approximately 51 hours. Therefore, the following two exposure periods were identified for this condition assessment:

- *Exposure Period #1*: Breaker 1EMXD-F02A is open concurrently with EDG 1A unavailable due to maintenance for approximately 51 hours.

- *Exposure Period #2*: While in Modes 1–3, breaker 1EMXD-F02A was open from December 15, 2015, at 1:52 p.m., until it is closed at 12:01 p.m. on March 28, 2016, which is approximately 2443 hours (after subtracting the 51 hours from Exposure Period #1).

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

Exposure Period #1 (51 Hours)

- Basic event HPR-CRB-ND36B was set to TRUE due to breaker 1EMXD-F02A being open.
  - During a postulated loss of the redundant interlock signal from the other RHR hot-leg suction valve 1ND-37A due to the unavailability of train A safety-related AC power, operators would be unable to manually open 1NI-136B from the main control room. During scenarios where ECCS injection is required, operators will enter ES-1.3, "*Transfer to Cold Leg Recirculation,*" when the refueling water storage tank (FWST) level reaches its low level set point (20 percent). Step 6.i of ES-1.3 directs operators to manually open 1NI-136B (and 1ND-28A) when FWST level decreases to 5 percent. This procedure does not contain guidance on the applicable interlock or direct verification of breaker positions of the RHR hot-leg suction valves (1ND-1B, 1ND-2A, 1ND-36B, and 1ND-37A). In addition, there are no readily available cues/indications of the breaker positions for the RHR hot leg suction valves.[3, 4]

- Basic event EPS-DGN-TM-1A (*diesel generator 1A unavailable due to test and maintenance*) was set to TRUE.[5,6]
  - Maintenance activities on critical equipment would have been restricted while EDG A was inoperable. Therefore, basic events EPS-DGN-TM-1B (*diesel generator 1B unavailable due to test and maintenance*), AFW-TDP-TM-TDP (*AFW TDP unavailable due to test and maintenance*), and AFW-MDP-TM-1B (*AFW motor-driven 1B unavailable due to test and maintenance*) were set to FALSE.

Exposure Period #2 (2443 Hours)

- Basic event HPR-CRB-ND36B was set to TRUE due to breaker 1EMXD-F02A being open.

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3 The ASME/ANS PRA Standard requires applicable procedures to credit operator recovery actions.

4 Time available for operators to troubleshoot the failure of 1NI-136B to open would be limited given that only 5 percent of FWST inventory would be available when operators attempt to manually open 1NI-136B.

5 No credit for EDG recovery is provided in this analysis, which is potentially conservative. However, information on the recoverability of EDG A during the periods of inoperability reported in LER 413/2016-001 is unknown.

6 When setting EPS-DGN-TM-1A to TRUE, SAPHIRE will automatically set the fail- to-run and fail-to-start basic events for the effected EDG to 1.0. Given these changes, SAPHIRE will recalculate the common-cause failure (CCF) probabilities based on the math contained in Appendix E of NUREG/CR-5485. Since the common cause component group size for the Unit 1 EDGs is two, the CCF failure probabilities were left at their nominal values.

ANALYSIS RESULTS

ΔCDP. The point estimate increase in core damage probability (ΔCDP) for this event is 1.1×10^{-6}, which is the sum of both exposure periods. The ASP Program acceptance threshold is a ΔCDP of 1×10^{-6} for degraded conditions. The ΔCDP for this event exceeds this threshold; therefore, this event is a precursor. The dominant initiating events for this analysis are provided in the following table.

Event Tree	ΔCDP	Percentage	Description
LOOP	5.38E-07	47.6%	Loss of Offsite Power
LOACA	4.33E-07	38.4%	Loss of Essential AC Bus A
SORV	1.27E-07	11.2%	Stuck-Open Pressurizer Safety Relief Valve
TRANS	1.17E-08	1.0%	Transient
MLOCA	8.69E-09	0.8%	Medium Loss-of-Coolant Accident

Dominant Sequence. The dominant accident sequence is loss of 4.16kV, safety-related AC bus A (LOACA) Sequence 19 (ΔCDP = 3.3×10^{-7}), which contributes approximately 29% of the total internal events ΔCDP. The dominant sequences that contribute at least 1.0 percent to the total internal events ΔCDP are provided in the following table. The dominant sequence is shown graphically in Figure A-1 in Appendix A.

Sequence	ΔCDP	Percentage	Description
LOACA 19	3.32E-07	29.4%	Loss of essential AC bus A initiating event; successful reactor trip; auxiliary feedwater (AFW) and main feedwater (MFW) fail; feed and bleed cooling succeeds; recovery of secondary side cooling fails; and high-pressure recirculation fails
LOOP 19-02	1.34E-07	11.9%	LOOP initiating event; successful reactor trip; emergency power system failure results in SBO; AFW succeeds; standby shutdown facility successfully provide reactor coolant pump (RCP) seal cooling and steam generator inventory makeup; operators fail to restore offsite power within 24 hours
SORV 3	1.24E-07	11.0%	Stuck-open pressurizer relief valve (LOCA) initiating event; successful reactor trip; high-pressure injection succeeds; AFW succeeds; successful secondary side cooldown/depressurization; and high-pressure/ low-pressure recirculation fail
LOOP 19-77	1.07E-07	9.5%	LOOP initiating event; successful reactor trip; emergency power system failure results in SBO; AFW fails; operators fail to restore offsite power within 2 hours
LOACA 5	9.79E-08	8.7%	Loss of essential AC bus A initiating event; successful reactor trip; AFW succeeds; pressurizer relief valve fail to close (LOCA); high-pressure injection succeeds; successful secondary side cooldown/depressurization; shutdown cooling fails; and high-pressure recirculation fails

Sequence	ΔCDP	Percentage	Description
LOOP 19-11-10	9.39E-08	8.3%	LOOP initiating event; successful reactor trip; emergency power system failure results in SBO; AFW succeeds; standby shutdown facility fails to provide RCP seal cooling; RCP seal integrity is maintained; operators successfully shed DC loads to extend battery depletion time to 4 hours; operators fail to restore offsite power within 4 hours
LOOP 18	6.61E-08	5.9%	LOOP initiating event; successful reactor trip; emergency power system succeeds; AFW fails; feed and bleed cooling fails; operators fail to restore offsite power within 2 hours
LOOP 02-05	3.21E-08	2.8%	LOOP initiating event; successful reactor trip; emergency power system succeeds; AFW fails; loss of RCP seal injection/cooling occurs; RCP stage 2 seals fail (LOCA); operators fail to restore offsite power within 2 hours; high-pressure injection fails
LOOP 19-17	2.35E-08	2.1%	LOOP initiating event; successful reactor trip; emergency power system failure results in SBO; AFW succeeds; standby shutdown facility fails to provide RCP seal cooling; RCP stage 2 seals fail (LOCA); operators successfully shed DC loads to extend battery depletion time to 4 hours; operators fail to restore offsite power within 4 hours
LOOP 9	2.00E-08	1.8%	LOOP initiating event; successful reactor trip; emergency power system succeeds; AFW fails; pressurizer relief valve fail to close (LOCA); high-pressure injection succeeds; operators fail to restore offsite power within 2 hours; high-pressure recirculation fails
LOOP 19-05-10	1.50E-08	1.3%	LOOP initiating event; successful reactor trip; emergency power system failure results in SBO; AFW succeeds; standby shutdown facility successfully RCP seal cooling; but fails to provide steam generator inventory makeup; operators successfully shed DC loads to extend battery depletion time to 4 hours; operators fail to restore offsite power within 4 hours
LOOP 15	1.10E-08	1.0%	LOOP initiating event; successful reactor trip; emergency power system succeeds; AFW fails; feed and bleed cooling succeeds; operators fail to recover offsite power within 6 hours; and high-pressure recirculation fails

REFERENCES

1. Catawba Nuclear Station, "LER 413/16-001 – Mispositioned Breaker for Residual Heat Removal Loop Suction Results in Inoperable Train of Emergency Core Cooling System," dated June 23, 2016 (ML16179A207).

2. U.S. Nuclear Regulatory Commission, "Catawba Nuclear Station - NRC Integrated Inspection Report 05000413/2016002, 05000414/2016002," dated July 19, 2016 (ML16202A116).

3. U.S. Nuclear Regulatory Commission, "Quality Assurance Program Requirements (Operation)," Regulatory Guide 1.33, Revision 3, dated June 2013 (ML13109A458).

Appendix A: Key Event Tree

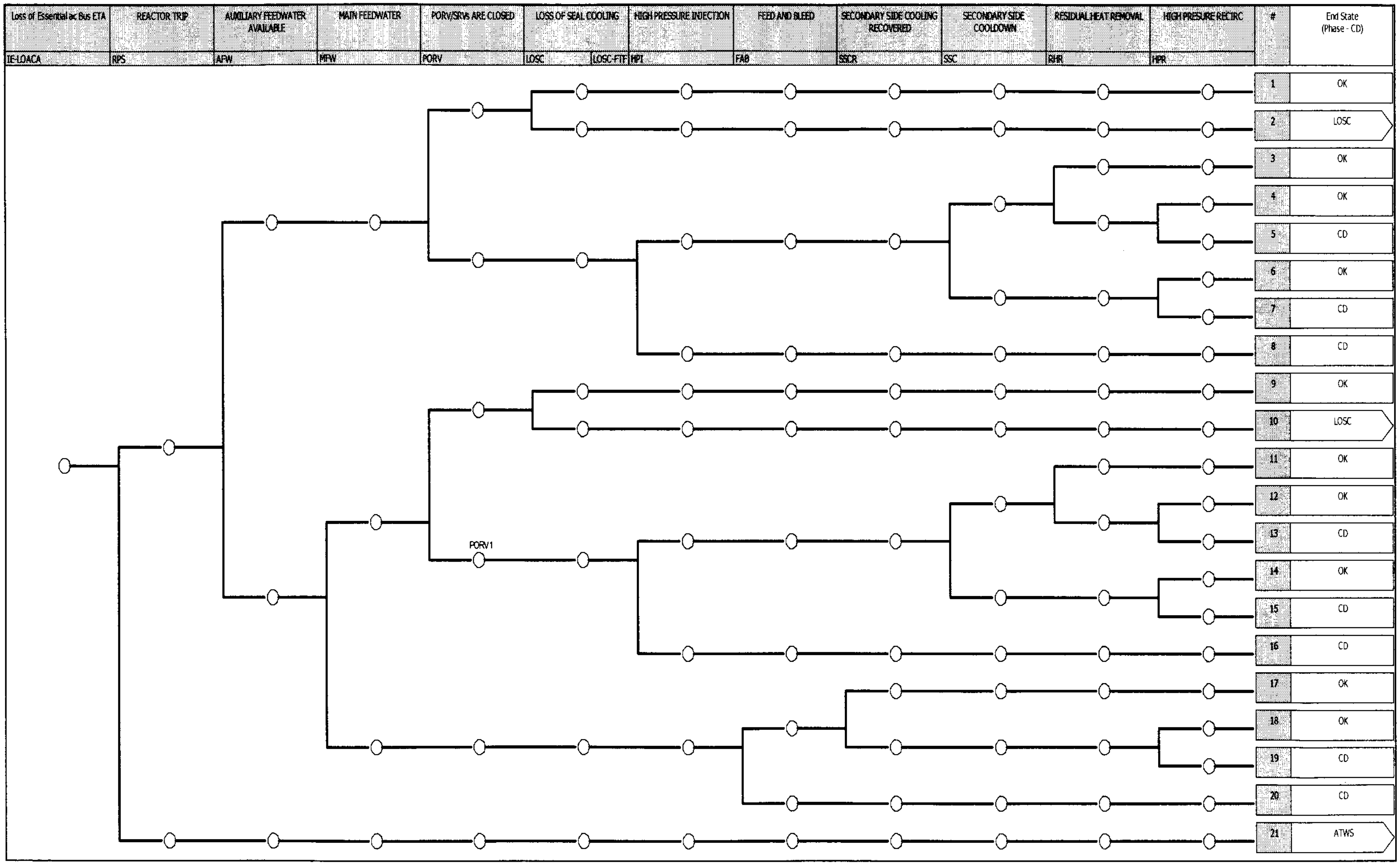

Figure A-1. Catawba Loss of 4.16kV, Safety-Related AC Bus A

Appendix B: System Drawing

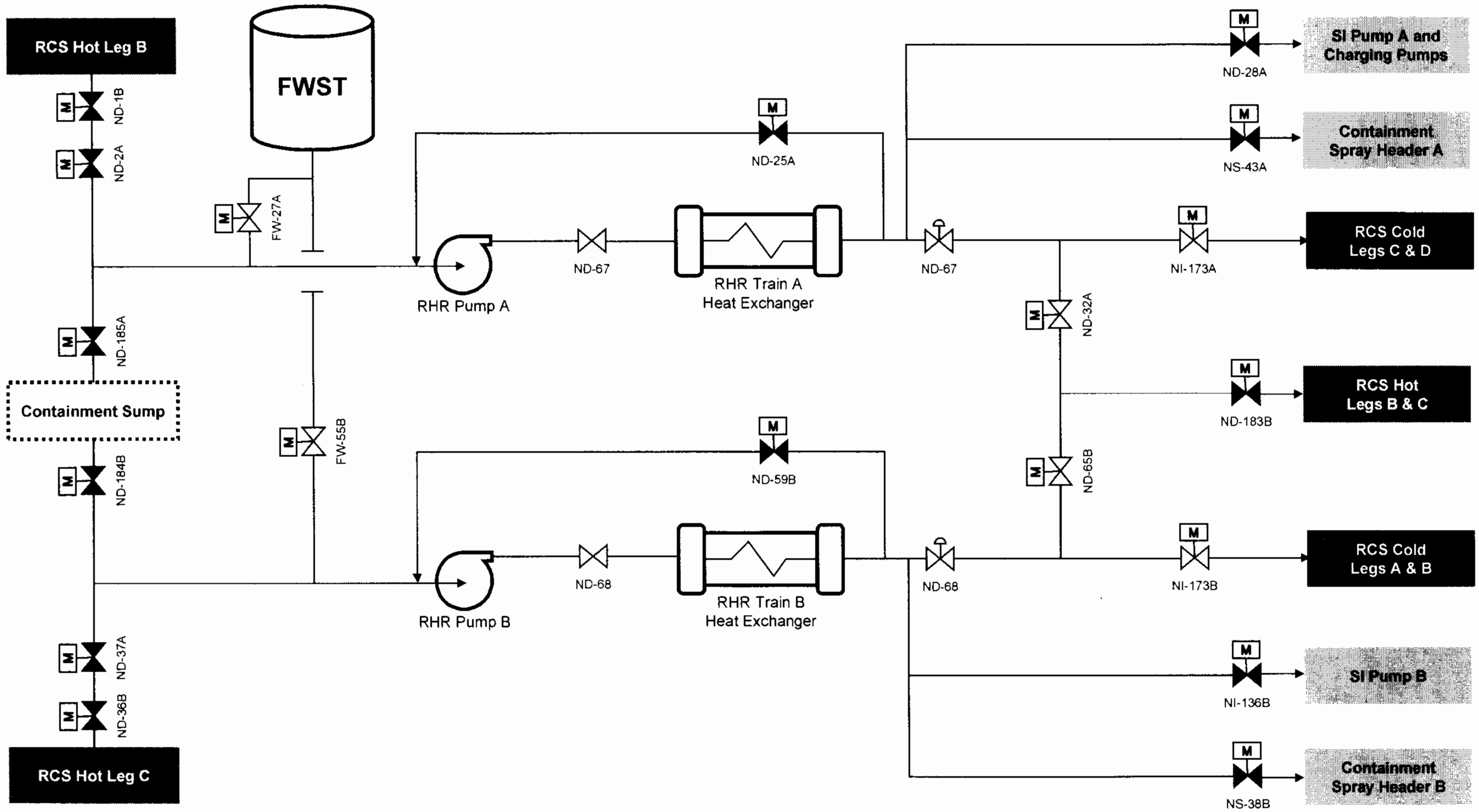

Figure B-1. Simplified RHR System Drawing for Catawba Unit 1

Appendix C: Modified Fault Trees

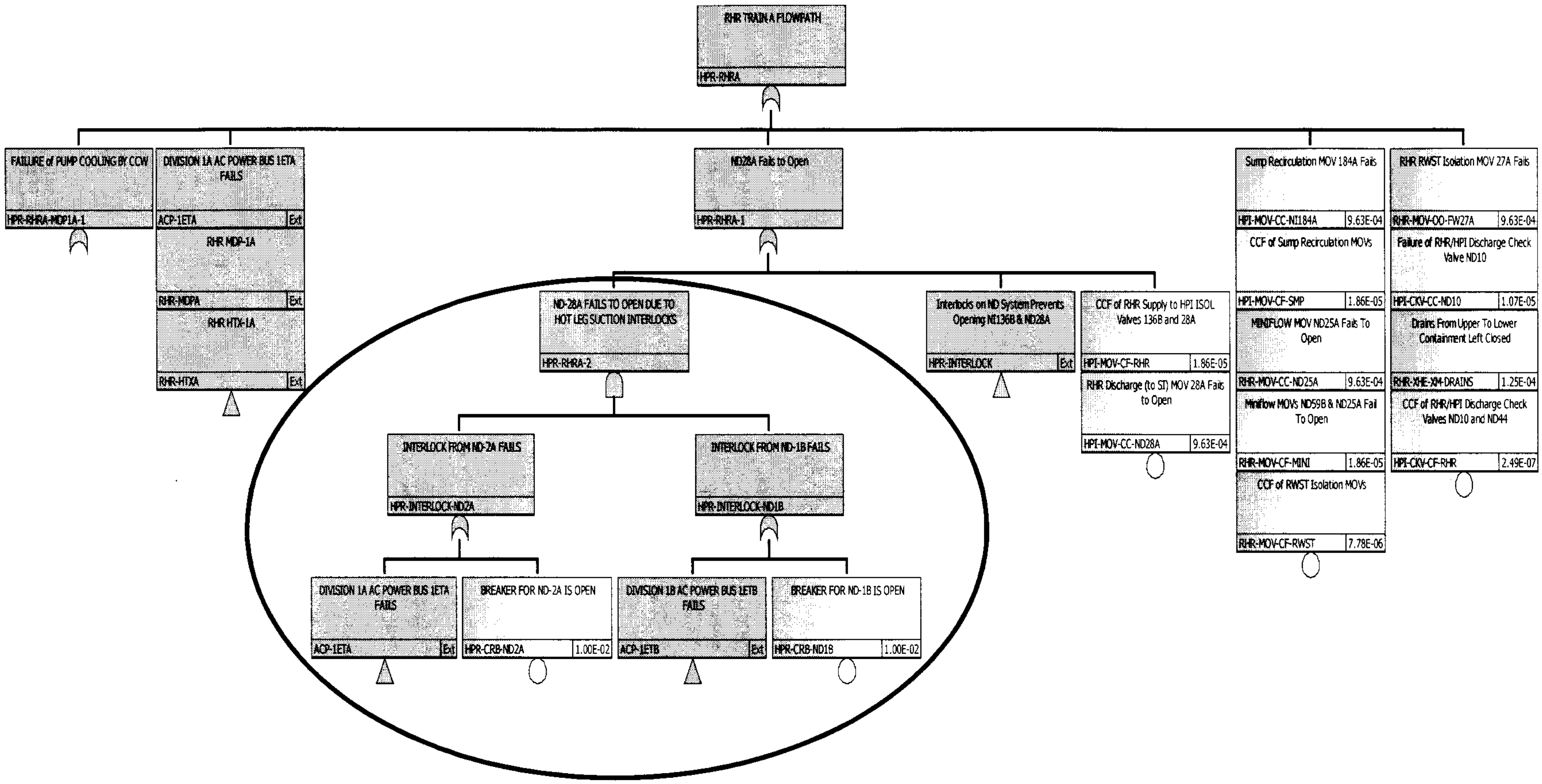

Figure C-1. Modified HPR-RHRA Fault Tree

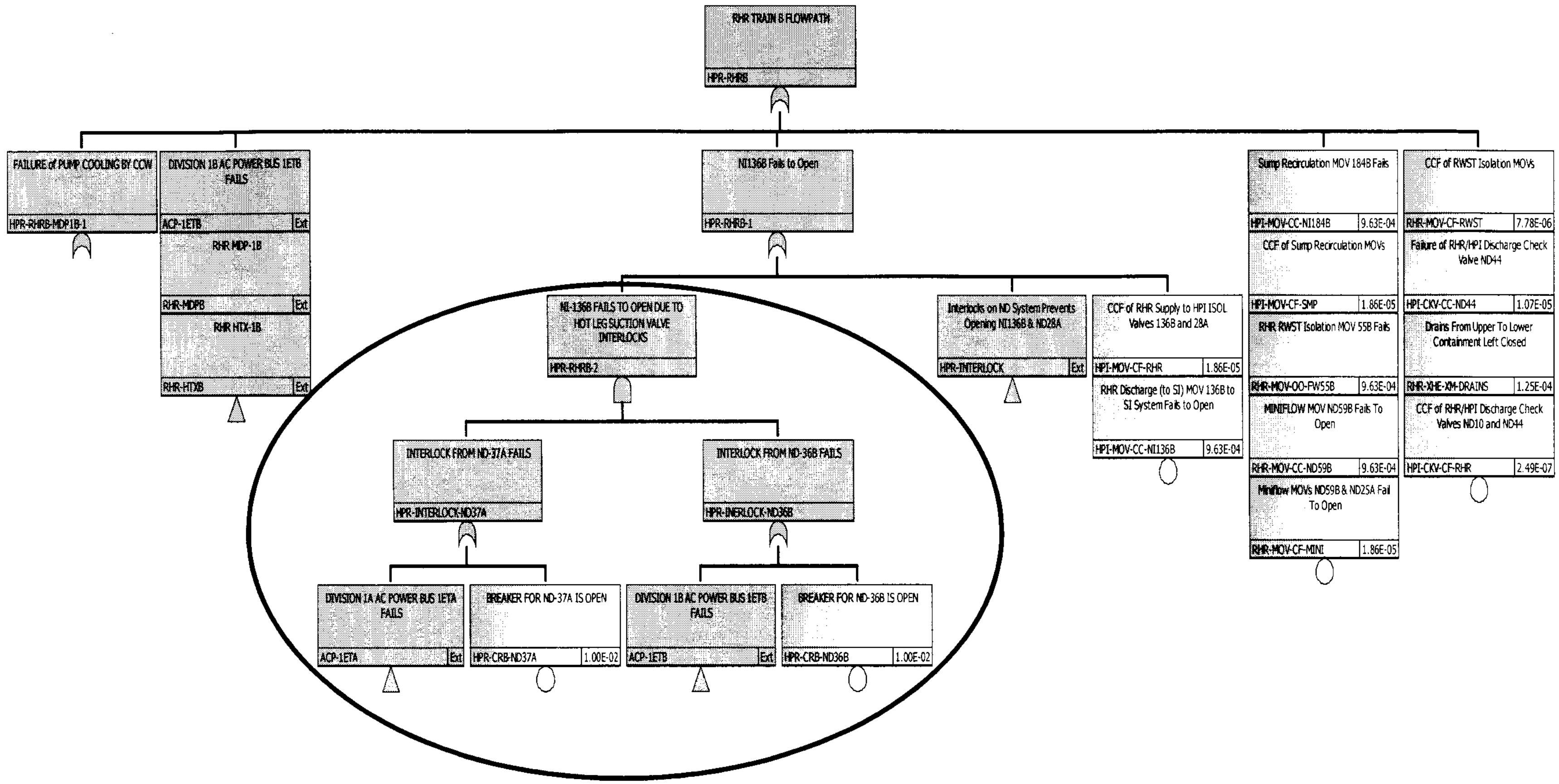

Figure C-2. Modified HPR-RHRB Fault Tree

SUBJECT: TRANSMITTAL OF FINAL CATAWBA NUCLEAR STATION, UNIT 1 ACCIDENT SEQUENCE PRECURSOR REPORT (LICENSEE EVENT REPORT 413-2016-001) (EPID L-LRO-2017-0057) DATED NOVEMBER 22, 2017

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