Final ASP Program Analysis–Reject

Accident Sequence Precursor Program – Office of Nuclear Regulatory Research

James E. FitzPatrick Nuclear Power Plant	Frazil Ice Leads to Manual Scram with Subsequent Loss of Non-Vital Buses due to Fast Transfer Failure	
Event Date: 1/23/2016	**LERs:** 333-2016-001, 333-2016-002 **IRs:** 05000333/2016001	**CCDP =** 8×10^{-6}
Plant Type: Boiling-Water Reactor (BWR); General Electric-4 with a Mark I Containment		
Plant Operating Mode (Reactor Power Level): Mode 1 (89% Reactor Power)		

Analyst:	**Reviewer:**	**Contributors:**	**BC Review Date:**
Christopher Hunter	Ian Gifford	N/A	**12/12/2016**

EVENT DETAILS

Event Description. On January 23, 2016, at 10:17 p.m., with reactor power at 89 percent, operators received an alarm for low screenwell level (242 feet).[1] Based on lake and outside environmental conditions, this was considered likely to be due to frazil ice. Operators entered Abnormal Operating Procedure (AOP) 56, "Intake Water Level Trouble," and began reducing power. When power was less than 75 percent, operators secured one of the three circulating water pumps (lower water velocity tends to slow the formation of frazil ice). Intake level increased slightly, but then resumed its lowering trend. Operators continued to reduce power, but when screenwell level reached 240 feet at 10:40 p.m., operators inserted a manual scram as directed by AOP-56.

Following the reactor scram and turbine trip, the expected automatic "fast" transfer of station electrical loads from the main generator through the normal station service transformer did not occur. Within 3 seconds, the backup automatic "residual" transfer did occur; however, all previously running non-vital equipment (e.g., reactor recirculating water pumps, circulating water pumps, feedwater/condensate pumps) were lost. Operators shut the main steam isolation valves (MSIVs) due to the loss of all circulating water system pumps. With the main condenser unavailable, the residual heat removal (RHR) system provided the ultimate heat sink. High-pressure coolant injection (HPCI), reactor core isolation cooling (RCIC) and the safety relief valves successfully provided reactor pressure vessel (RPV) level and pressure control. Operators used these systems to perform a slow plant cooldown while working to restore normally operating plant systems to service. Shutdown cooling mode of RHR was initiated at 10:59 p.m. on January 24, 2016.

Additional information is provided in Licensee Event Report (LER) 333-2016-001 (Ref.1) and Inspection Report (IR) 05000333/2016001 (Ref. 2).

Cause. Operators initiated a manual scram, per procedures, due to low screenwell level caused by frazil ice. The failure of the "fast" transfer appears to have been caused by lubrication hardening in the lower control valve assembly of breaker 71PCB-10042, which

[1] Normal screenwell intake level is approximately 244 feet.

resulted in the breaker opening slower than designed; therefore, causing the “residual” transfer logic to actuate.

MODELING ASSUMPTIONS

Analysis Type. The FitzPatrick Standardized Plant Analysis Risk (SPAR) Model Version 8.17 dated May 20, 2014, was used for this event analysis.

SDP Results/Basis for ASP Analysis. The ASP Program uses Significance Determination Process (SDP) results for degraded conditions when available. Two Green findings (i.e., very low safety significance) were identified (see IR 05000333/2016001 for additional information).[2, 3] These findings include:

- The licensee failed to maintain a condition specified in an emergency operating procedure. Specifically, while operating HPCI in the pressure control mode, operators failed to override automatic transfer of the HPCI pump suction from the condensate storage tank to the suppression pool prior to the transfer actually occurring. As a result, operators reverted to using the safety relief valves (SRVs) for pressure control, which introduced unnecessary plant challenges. This performance deficiency was not determined to result in a potential or actual loss of safety function of the HPCI system; therefore, this finding was not considered in this ASP analysis.
- The licensee failed to take actions specified in the procedure for initiation of shutdown cooling. Specifically, prior to placing the RHR loop A into shutdown cooling mode, an operator was not stationed to close the condensate transfer system cross-connect valve, nor was the valve immediately closed after initiation of shutdown cooling, as specified by the operating procedure. This resulted in a significant loss of operational control, in that RPV level increased to the point of putting water down the main steam lines. This performance deficiency was not determined to result in a potential or actual loss of safety function; therefore, this finding was not considered in this ASP analysis.

The ASP Program performs independent analyses for initiating events. ASP analyses of initiating events account for all failures/degraded conditions and unavailabilities (e.g., equipment out for test/maintenance) that occurred during the event, regardless of licensee performance.[4] Additional LERs were reviewed to determine if concurrent unavailabilities existed during the January 23, 2016, event. This review revealed LER 333-2016-002 (Ref. 3), which was issued due to the slow closing of a MSIV during the January 23, 2016, event. In addition, subsequent testing revealed an additional MSIV closed slowly. The slow closing of these MSIVs is attributed to the sticking of their DC solenoid valves. The slow closing of these two outboard MSIVs was not included in the ASP analysis because the time to MSIV closure is typically only important during a postulated main steam line break. In addition, the inboard MSIVs were unaffected by this issue.

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

- This analysis models the January 23, 2016, manual reactor trip at FitzPatrick as a loss of condenser heat sink initiating event due to loss of the circulating water, feedwater, and condensate pumps due to the failure of the fast transfer. Therefore, the probability of a loss

[2] The SDP evaluates each inspection finding (i.e., licensee performance deficiency) individually.

[3] LER 333-2016-001 is not closed (to date).

[4] ASP analyses also account for any degraded condition(s) that were identified after the initiating event occurred if the failure/degradation exposure period(s) overlapped the initiating event date.

of condenser heat sink (*IE-LOCHS*) was set to 1.0; all other initiating event probabilities were set to zero.

- The failure of the automatic “fast” transfer that resulted in the loss of electrical power to the non-vital buses is implicit within the loss of condenser heat sink initiating event. This initiating event (and its associated event tree) assumes that key systems normally powered by the non-vital buses (e.g., circulating water, condensate, feedwater) are rendered unavailable by the initiating event.

- During the frazil ice conditions experienced on January 23, 2016, operators successfully implemented AOP-56 to reduce reactor power and secure a running circulating water pump.[5, 6] In addition, operators manually scrammed the reactor when screenwell level reached 240 feet. During the actual event, the failure of the “fast” transfer of electrical loads resulted in what could be considered a “benevolent” failure (in terms of restoring screenwell level) since it resulted in a loss of the two running circulating water pumps. Within 5 minutes of the loss of all circulating water, screenwell level was fully restored (approximately 245 feet). No subsequent decrease in screenwell level was observed.

  If operators had failed to manually scram the reactor and perform the mitigation actions of AOP-56, screenwell level would have continued to decrease (note that operators successfully implemented all steps in AOP-56 during the event). The minimum suction level of the circulating water pumps is significantly higher (239.5 feet) than the minimum level required to support operation of the essential service water (ESW) and RHR service water pumps (235 feet). Although it is likely that circulating water pumps would continue to run with screenwell level below 239.5 feet, it is likely that the circulating water pumps would fail prior to the loss of suction of ESW and RHR service water pumps, resulting in a similar event progression to that experienced during the actual event (i.e., loss of all circulating water).[7] Therefore, this alternate scenario was not considered as part of this analysis.

ANALYSIS RESULTS

Conditional Core Damage Probability (CCDP). The point estimate CCDP for this event is 8.02×10^{-6}. The ASP Program acceptance threshold is a CCDP of 1×10^{-6} or the CCDP equivalent of an uncomplicated reactor trip with a non-recoverable loss of feedwater and condenser heat sink, whichever is greater. This CCDP equivalent for FitzPatrick is 8.02×10^{-6}. The CCDP for this event does not exceed the initiating event threshold for FitzPatrick; therefore, this event is screened out of the ASP Program.

5 Additional mitigation actions directed by AOP-56 are to reduce reactor power and secure one of the running circulating water pumps. If screenwell level is still decreasing after completion of these steps, operators are directed to perform a normal reactor shutdown.

6 Note that prior to the January 23, 2016, event, circulating water tempering flow was in use to direct warm discharge water to the screenwell. The warm discharge water increases intake bay water temperature to minimize frazil ice buildup on the trash rack bars and concrete surfaces of the intake structure. In addition, tempering reduces the intake velocity; therefore, reducing the potential for drawing frazil ice into the intake structure. However, the frazil ice buildup still resulted in lowering screenwell levels, which led operators to enter AOP-56 and perform the directed mitigation actions.

7 Based on discussions with licensee staff, if screenwell level continued to decrease, circulating water flow would continue to decrease due to air binding of the pumps resulting in decreasing condenser vacuum and a subsequent (automatic) reactor scram. The circulating water pump impellers and pump casing center-line is at a screenwell level of 235.5 feet; therefore, the pumps would not continue to operate below this level (the pumps would likely trip on over-current).

Dominant Sequence. The dominant accident sequence is LOCHS Sequence 51 (CCDP = 7.9×10^{-6}), which contributes approximately 99% of the total internal events CCDP. The cut sets/sequences that contribute to the top 95% and/or at least 1% of the total internal events CCDP are provided in Appendix A.

The dominant sequence is shown graphically in Figure B-1 in Appendix B. The events and important component failures in LOCHS Sequence 51 are:

- A non-recoverable loss of condenser heat sink occurs,
- Reactor scram succeeds,
- Safety relief valves reclose,
- High-pressure injection (RCIC or HPCI) fails, and
- Manual reactor depressurization fails.

REFERENCES

1. James A. FitzPatrick Nuclear Power Plant, "LER 333/16-001 – System Actuations during Manual Scram in Response to Frazil Ice Blockage and Residual Transfer," dated March 23, 2016 (ML16083A504).

2. U.S. Nuclear Regulatory Commission, "James A. Fitzpatrick Nuclear Power Plant – Integrated Inspection Report 05000333/2016001," dated May 13, 2016 (ML16134A301).

3. James A. FitzPatrick Nuclear Power Plant, "LER 333/16-002 – Sticking DC Pilot in Solenoid Valve Cluster Assembly Results in Slow MSIV Closures," dated April 25, 2016 (ML16116A245).

Appendix A: Analysis Results

Summary of Conditional Event Changes

Event	Description	Conditional Value	Nominal Value
IE-LOCHS	LOCHS LOSS OF CONDENSER HEAT SINK	1.0[a]	1.39E-1

a. All other initiating event probabilities were set to zero.

Dominant Sequence Results

Only items contributing at least 1.0% to the total CCDP are displayed.

Event Tree	Sequence	CCDP	% Contribution	Description
LOCHS	51	7.92E-6	98.7	/RPS,/SRV,HPI,DEP
	Total	**8.02E-6**	**100%**	

Referenced Fault Trees

Fault Tree	Description
DEP	MANUAL REACTOR DEPRESSURIZATION
HPI	HIGH PRESSURE INJECTION (RCIC or HPCI)
RPS	REACTOR PROTECTION SYSTEM
SRV	SAFETT RELIEF VALVES RECLOSE

Cut Set Report – LOCHS 51

Only items contributing at least 1% to the total are displayed.

#	CCDP	Total%	Cut Set
	7.92E-6	100	Displaying 258 Cut Sets. (258 Original)
1	2.96E-6	37.42	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-MOV-CC-IVFRO,RCI-TDP-FR-TRAIN
2	8.21E-7	10.37	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-MOV-CC-IVFRO,RCI-TDP-TM-TRAIN
3	7.89E-7	9.97	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-MOV-CC-IVFRO,RCI-RESTART, RCI-TDP-FS-RSTRT,RCI-XHE-XL-RSTRT
4	7.80E-7	9.86	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-TDP-FR-TRAIN,RCI-TDP-FR-TRAIN
5	4.87E-7	6.15	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-MOV-CC-IVFRO,RCI-TDP-FS-TRAIN
6	2.24E-7	2.83	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-TDP-TM-TRAIN,RCI-TDP-FR-TRAIN
7	2.16E-7	2.73	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-TDP-FR-TRAIN,RCI-TDP-TM-TRAIN
8	2.08E-7	2.62	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-TDP-FR-TRAIN,RCI-RESTART, RCI-TDP-FS-RSTRT,RCI-XHE-XL-RSTRT
9	1.28E-7	1.62	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-TDP-FS-TRAIN,RCI-TDP-FR-TRAIN
10	1.28E-7	1.62	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-TDP-FR-TRAIN,RCI-TDP-FS-TRAIN
11	1.02E-7	1.28	IE-LOCHS,ADS-XHE-XE-MDEPR,HCI-MOV-CC-IVFRO,RCI-MOV-FC-XFER, RCI-XHE-XL-XFER
12	9.97E-8	1.26	IE-LOCHS,DCP-BAT-CF-BATT

Referenced Events

Event	Description	Probability
ADS-XHE-XE-MDEPR	OPERATOR FAILS TO DEPRESSURIZE THE REACTOR	5.00E-4
DCP-BAT-CF-BATT	COMMON CAUSE FAILURE OF DIVISION BATTERIES	9.97E-8
HCI-MOV-CC-IVFRO	HPCI INJECTION VALVE FAILS TO REOPEN	1.50E-1
HCI-TDP-FR-TRAIN	HPCI PUMP TRAIN FAILS TO RUN GIVEN IT STARTED	3.95E-2
HCI-TDP-FS-TRAIN	HPCI PUMP FAILS TO START	6.49E-3
HCI-TDP-TM-TRAIN	HPCI TRAIN IS UNAVAILABLE BECAUSE OF MAINTENANCE	1.13E-2

Event	Description	Probability
IE-LOCHS	LOSS OF CONDENSER HEAT SINK	1.00E+0
RCI-MOV-FC-XFER	RCIC FAILS TO TRANSFER DURING RECIRCULATION	7.97E-3
RCI-RESTART	RESTART OF RCIC IS REQUIRED	2.63E-1
RCI-TDP-FR-TRAIN	RCIC PUMP FAILS TO RUN GIVEN THAT IT STARTED	3.95E-2
RCI-TDP-FS-RSTRT	RCIC FAILS TO RESTART GIVEN START AND SHORT-TERM RUN	8.00E-2
RCI-TDP-FS-TRAIN	RCIC PUMP FAILS TO START	6.49E-3
RCI-TDP-TM-TRAIN	RCIC TRAIN IS UNAVAILABLE BECAUSE OF MAINTENANCE	1.09E-2
RCI-XHE-XL-RSTRT	OPERATOR FAILS TO RECOVER RCIC FAILURE TO RESTART	5.00E-1
RCI-XHE-XL-XFER	OPERATOR FAILS TO RECOVER TRANSFER FAILURE	1.70E-1

Appendix B: Key Event Tree

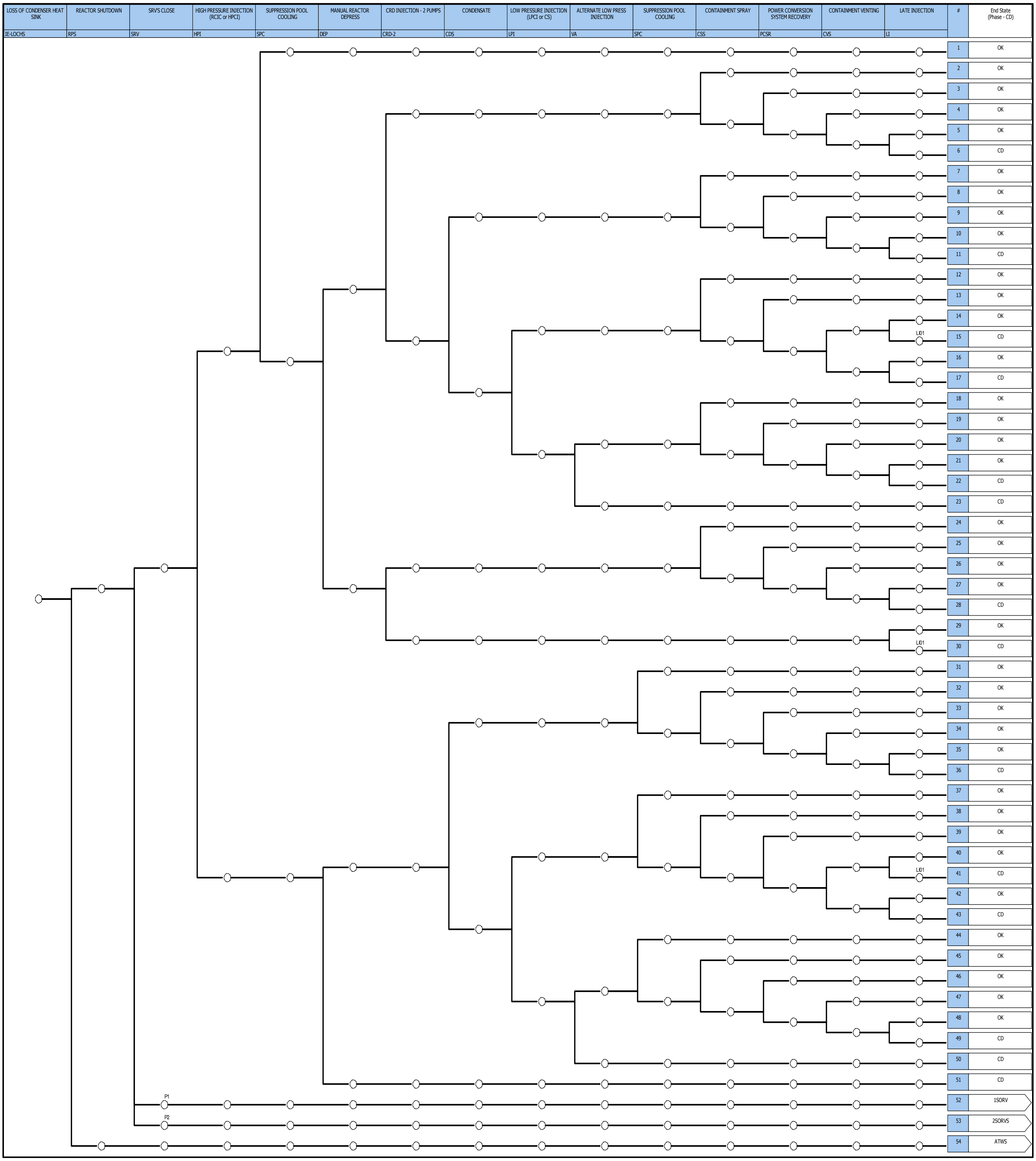

Figure B-1. FitzPatrick Loss of Condenser Heat Sink Event Tree