Final ASP Program Analysis - Reject

Accident Sequence Precursor Program – Office of Nuclear Regulatory Research			
Diablo Canyon Power Plant, Unit 1	Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld		
Event Date: 12/31/2014	**LERs:** 275-2015-001 **IR:** 50-275/2015-004		**CCDP=** 2×10^{-8}
Plant Type: Westinghouse 4-Loop Pressurized Water Reactor with Large Dry Containment			
Plant Operating Mode (Reactor Power Level): Mode 3 (0 Percent Reactor Power, Operating Temperature and Pressure)			
Analyst: Candace Pfefferkorn	**Reviewer:** Keith Tetter	**Contributors:** David Aird	**BC Approved Date:** 6/9/2016

EVENT DETAILS

Event Description. On December 31, 2014, both trains of residual heat removal (RHR) at Diablo Canyon Power Plant, Unit 1, were declared inoperable. The declaration was made after plant personnel performing a walk-down identified boric acid accumulation/through-wall seepage (30 drops per minute) originating from a circumferential crack on the socket weld connection from the RHR pump common discharge header to relief valve (RV) RHR-1-RV-8708. Relief valve RHR-1-RV-8708 is inside containment and protects RHR discharge piping to reactor coolant system (RCS) hot legs 1 and 2 from exceeding its design pressure rating. The condition was entered at 11:05 am and was exited at 10:56 pm when the plant entered Mode 4. Immediate corrective actions included a repair of the cracked socket weld and installation of a pipe support on RV-8708. In addition, per technical specifications, the associated containment penetration flow path was isolated.

Cause. The root cause of the condition was determined to be containment fan cooler unit (CFCU) vibration that induced a resonant condition in the RHR piping generating stresses above the material endurance limit of the socket weld. Subsequent corrective actions included the replacement of the previously repaired socket weld and one additional socket weld on the RV discharge pipe, the relocation of the previously installed pipe supports, and correction of the condition that caused the CFCU vibrations. Notably, a similar condition (circumferential crack on the socket weld RHR connection RV RHR-1-RV-8708), was documented in LER 275-2013-005 for Diablo Canyon Nuclear Power Plant, Unit 1. In that case, the condition was attributed to low-stress, high-cycle fatigue caused by system vibration.

MODELING

Basis for ASP Analysis/SDP Results. The ASP Program uses Significance Determination Process (SDP) results for degraded conditions when available and applicable. 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.[1]

[1] 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.

In Inspection Report 50-275/15-04 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16035A481), the inspectors documented their review of the events described in License Event Report (LER) 275-2015-001 and 275-2013-005. The inspectors documented a green self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Action," for the licensee's failure to identify the cause and take corrective action to prevent recurrence of a significant condition adverse to quality impacting both trains of the Unit 1 safety-related residual heat removal (RHR) system. Specifically, the licensee failed to identify a definitive cause and implement corrective actions to prevent recurrent failures of the socket weld for RV RHR-1-RV-8708 for both trains of the RHR system.

While the issue associated with LER 275-2015-001 was addressed through the SDP in Inspection Report 50-275/15-04, as of the writing of this report, the LER has not been documented as closed in an inspection report.

An independent ASP analysis was performed for this condition assessment because this event has not yet been directly closed to date.

Analysis Type. The Diablo Canyon Unit 1 & 2 Standardized Plant Analysis Risk (SPAR) model, Revision 8.23, created May 2014, was used for this event analysis. The event was modeled as a condition assessment.

SPAR Model Modifications. Fault tree modifications and basic event additions were made within the BFN Unit 3 SPAR model as described below:

- A new house event, titled LPI-RV-8708 (*Relief Valve 8708 Breaks*) was created to model a break resulting from the circumferential crack on the socket weld connection to RV RHR-1-RV-8708. The normal state of this valve is closed and therefore LPI-RV-8708 was set to FALSE.

- The LPI (*Low Pressure Injection*) fault tree was modified to include the basic house event LPI-RV-8708 (*Relief Valve 8708 Breaks*) under the LPI (*Low Pressure Injection*) "OR" gate.

- The LPR (*Low Pressure Recirculation*) fault tree was modified to include the basic house event LPI-RV-8708 (*Relief Valve 8708 Breaks*) under the LPR-SYS-F (Failure of LPI Systems During Recirculation) "OR" gate.

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

- While the RHR system was declared inoperable in accordance with Technical Specifications, the Diablo Canyon SPAR Report states that:

  > *Success [of RHR] implies the RCS pressure and temperature are within the requirements to allow the RCS hot leg (to LPI pump suction) suction valves to be opened and provide a suction source to the LPI pumps. The dedicated LPI pump train heat exchangers will slowly cool down the reactor. This system requires an operator action to open the RCS hot leg valves which provide the suction source for the pumps. The success criterion is one-of-two LPI pumps providing sufficient flow through their respective heat exchangers*

  Therefore, for this analysis, nominal probabilities for RHR basic events were used which include LPI hot leg suction valves, pumps, and heat exchangers, as well as the operator action to open RCS hot leg valves. Furthermore, most initiating event sequences that reference the RHR top event (fault tree) subsequently reference either low or high pressure recirculation (LPR or HPR fault trees, respectively) implicating RHR in providing heat exchange and suction for LPI, LPR and HPR.

- To ensure conservative bounding risk results, it is assumed that the subject circumferential crack on the socket weld connection to RV RHR-1-RV-8708 resulted in a full pipe rupture. The pipe rupture is assumed to be of sufficient size such that all RHR water is diverted from both cold leg LPI (or LPR) trains through the break until it is isolated manually by operators.

- The time duration of the condition is assumed to be one hour. In an emergency core cooling system mode valve alignment, two motor operated valves are open. These values would have to be closed from the control room during an incident to isolate a break in the hot leg injection line. Based on review of abnormal operating procedures for malfunction of RHR, an upper bound estimate of one hour was ascribed to this action.

- The subject pipe rupture, even if isolated, would result in a loss of low pressure hot leg injection during recirculation. Hot leg recirculation typically occurs approximately 24 hours after low pressure cold leg recirculation is initiated. Therefore, failure of hot leg injection during recirculation was not modeled in this analysis.

- At the time of the event Diablo Canyon Power Plant, Unit 1, was in Mode 3 (Hot Standby) with no components reported as being out of service for test and maintenance. However, this analysis assumes a full break in the hot leg injection line resulting from the circumferential crack on the socket weld connection to RV RHR-1-RV-8708. Thus, basic event LPI-RV-8708 was set to TRUE for this event analysis

- All other safety systems responded as designed.

ANALYSIS RESULTS

CCDP/Rejection Basis. The ∆CDP for this analysis is 1.8×10^{-8} which is below the ASP Program ∆CDP threshold of 1×10^{-6}. Therefore, this event is not a precursor and is screened out of the ASP Program.

Dominant Sequence. The dominant accident sequence is Medium Loss of Coolant Accident (MLOCA) Sequence 02 (∆CDP = 1.7×10^{-8}) that contributes approximately 98% of the total internal events CCDP. Figure 1 in Appendix B illustrates this sequence. 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 events and important component/system failures in MLOCA Sequence 02 are:

- Reactor trip succeeds,
- Offsite electrical power succeeds,
- High pressure injection succeeds
- Auxiliary feedwater succeeds,
- Reactor coolant system cooldown below RHR initial pressure succeeds,
- Low pressure recirculation fails.

REFERENCES

1. Diablo Canyon Power Plant, "LER 275-2015-001-01 – Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld," dated February 11, 2016 (ML16042A470).

2. Diablo Canyon Power Plant, "LER 275-2015-001-00 – Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld," dated March 2, 2015 (ML15061A548).

3. Diablo Canyon Power Plant, "LER 275-2013-005-00 – Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld," dated August 22, 2013 (ML13235A101).

4. U.S. Nuclear Regulatory Commission, "Diablo Canyon Plant – NRC Integrated Inspection Report 05000275/2015004," dated February 4, 2016 (ML16035A481).

5. Nuclear Regulatory Commission, "Diablo Canyon Plant – NRC Integrated Inspection Report 05000275/2014002," dated April 23, 2014 (ML14113A527).

Appendix A: SAPHIRE 8 Worksheet

Summary of Conditional Event Changes

Event	Description	Cond Value	Nominal Value
LPI-RV-8708	Relief Valve 8708 Breaks	True	False

Event Tree Dominant Results

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

EVENT TREE	CCDP	CDP	Δ CDP	DESCRIPTION
MLOCA	1.71E-8	3.85E-11	1.71E-8	Diablo Canyon 1 & 2 Medium Loss of Coolant Accident
LLOCA	2.85E-10	6.18E-13	2.85E-10	Diablo Canyon 1 & 2 Large Loss of Coolant Accident
Total	**1.90E-8**	**1.48E-9**	**1.75E-8**	

Dominant Sequence Results

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

EVENT TREE	SEQUENCE	CCDP	CDP	Δ CDP	DESCRIPTION
MLOCA	02	1.71E-8	3.75E-11	1.71E-8	/RPS, /OEP, /HPI, /AFW, /SSC1, LPR
LLOCA	3	2.85E-10	2.59E-14	2.85E-10	/OEP, /ACC, LPI
Total		**1.90E-8**	**1.48E-9**	**1.75E-8**	

Referenced Fault Trees

Fault Tree	Description
LPI	LOW PRESSURE INJECTION
LPR	LOW PRESSURE RECIRCULATION

Cut Set Report - MLOCA 02

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

#	PROB/FREQ	TOTAL%	CUT SET
Total	1.50E-4	100	Displaying 1 Cut Sets. (1 Original)
1	1.50E-4	100.00	IE-MLOCA,<TRUE>

Cut Set Report – LLOCA 3

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

#	PROB/FREQ	TOTAL%	CUT SET
	1.65E-7	100	Displaying 176 Cut Sets. (176 Original)
1	1.02E-7	61.80	IE-LOOPSC,ACP-CRB-CF-OPSD3
2	4.75E-8	28.84	IE-LOOPSC,RSW-STR-CF-ALL
3	2.60E-9	1.58	IE-LOOPSC,ACP-XHE-XM-EPSXT,EPS-XHE-XL-NR04H,OEP-XHE-XL-NR04HSC,OPR-XHE-XR-CASLT

Referenced Events

Event	Description	Probability
IE-LLOCA	LARGE LOCA	2.50E-6
IE-MLOCA	MEDIUM LOCA	1.50E-4

Appendix B: Key Event Tree

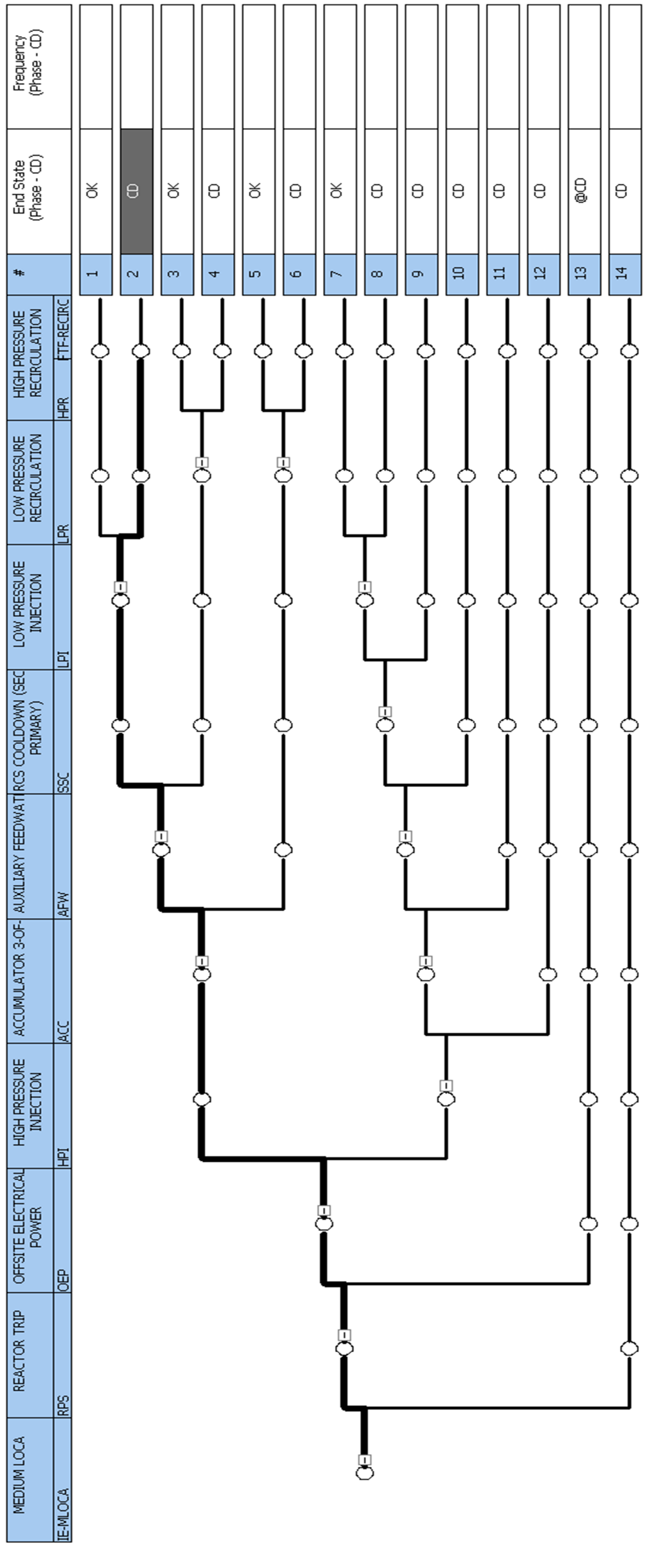

Figure 1: Diablo Canyon MLOCA Event Tree (Sequence 02 Bolded)

Appendix C: SPAR Model Modifications

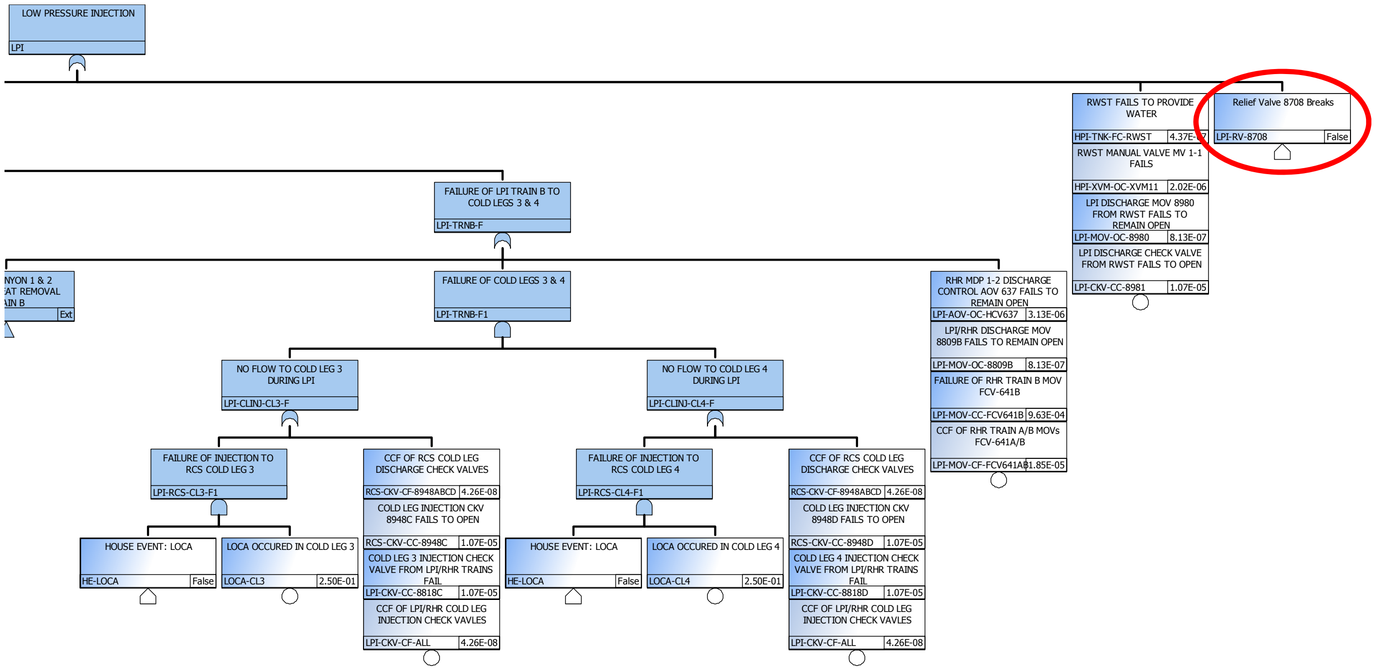

Figure 2: Change to LPI Fault Tree (Partial)

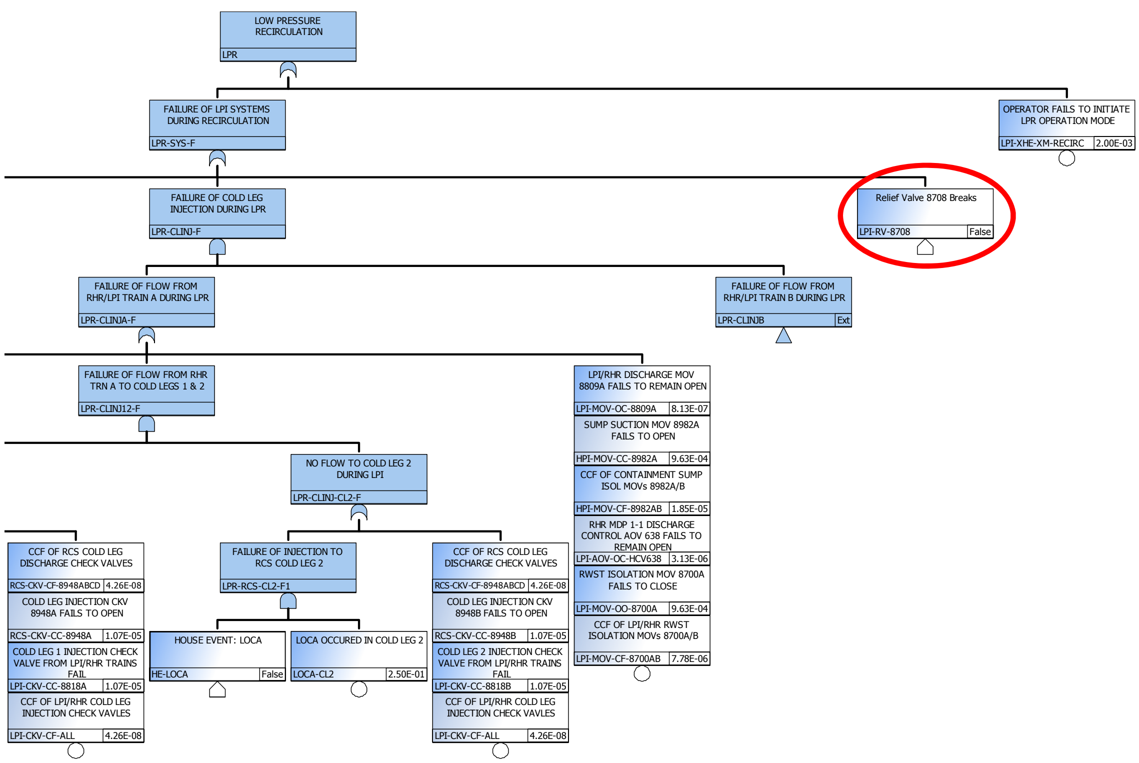

Figure 3: Change to LPR Fault Tree *(Partial)*