Preliminary Precursor Analysis

Accident Sequence Precursor Program --- Office of Nuclear Regulatory Research

Shearon Harris	Inoperable Charging Safety Injection Pump for Extended Period	
Event Date 05/15/1999	LER 400/00-007-00	ΔCDP = 4.2 x 10^{-6}

Condition Summary

On June 19, 2000, during the performance of a scheduled maintenance on the "C" charging safety injection pump (CSIP), the plant staff discovered the pump's outboard thrust bearing shoes were damaged. The "C" CSIP is one of three electrically driven centrifugal pumps used for normal charging evolutions, high pressure safety injection, and feed and bleed operations. Although the pump had performed successfully during normal operations, the pump was declared inoperable on September 4, 2000, due to the likely failure of the pump during operations (e.g., high pressure safety injection mode). Technical Specification (TS) 3.5.2 requires two independent subsystems of the Emergency Core Cooling System (ECCS) to be operable, including one CSIP in each subsystem. Because the inoperable "C" CSIP had been periodically designated as one of the two CSIPs for the ECCS, the TS was violated. (Refs. 1, 2, 3, and 4)

Cause. The plant staff was unable to determine the root cause for this event. However, potential causes identified include partial loss of lubricant flow to the outboard thrust bearing and improper fill and vent of the "C" CSIP. Reference 3 indicates that a lack of lubrication did not exist on the "A" and "B" CSIPs. Therefore, the failure of the "C" CSIP is considered to be an independent failure.

Condition duration. Based on a review of the "C" CSIP operating history and oil samples, it was concluded that the most likely time for the damage to occur was during a pump start that occurred on May 15, 1999. The "C" CSIP is not normally in service and is used to replace the "A" or "B" CSIP should either require extended maintenance.

The time periods following the May 15 pump start in which the "C" CSIP was acting in place of the "A" CSIP and thus designated as one of the two required CSIPs were from May 15 to June 4, 1999, and from November 13 to December 18, 1999 (a total of 55 days). The "C" CSIP was acting in place of the "B" CSIP and also designated as one of the two required CSIPs from January 3 to January 7, 2000 (4 days). The total time that the "C" CSIP was acting as an ECCS pump (in place of "A" or "B" CSIP) was 59 days.

The "C" CSIP also acted in a swing capacity (as a backup to the "A" or "B" CSIP) for a period of 194 days in 1999 and 2000. The plant is not required by the TS to have an operable swing pump.

Enclosure 1

Recovery opportunities. During operations (e.g., high pressure safety injection or feed and bleed) that required the "C" CSIP to operate with flow rates between 200 and 500 gallons per minute, the pump bearing would likely fail due to outboard thrust conditions. Such a failure of the "C" CSIP would not be recoverable by the operators.

Analysis Results

- **Importance**[1]

  The overall risk significance of the "C" CSIP being unavailable as part of the ECCS is determined by subtracting the total nominal core damage probability from the total conditional core damage probability:

Conditional core damage probability (CCDP) =	7.7×10^{-6}
Nominal core damage probability (CDP) =	-3.5×10^{-6}
Importance (ΔCDP = CCDP - CDP) =	4.2×10^{-6}

  The estimated importance (CCDP-CDP) for the condition was 4.2×10^{-6}. This is an increase of 4.2×10^{-6} over the nominal CDP for the 59-day period when the "C" CSIP was unavailable for the ECCS function.

  The Accident Sequence Precursor Program acceptance threshold is an importance of 1×10^{-6}.

- **Dominant sequence**

  The dominant core damage sequence for this condition is a loss of offsite power (LOOP) (Sequence 10). The events and important component failures for the LOOP (shown in Sequence 10, Figure 1) include:

  - a LOOP initiating event,
  - successful reactor trip,
  - success of the emergency power system,
  - success of the auxiliary feedwater system,
  - opening of at least one power-operated relief valve (PORV) or safety relief valve (SRV),
  - failure of open PORVs or SRVs to reseat, and
  - failure of high pressure injection.

- **Results tables**

  - The conditional probability of the dominant sequence is shown in Table 1.
  - The event tree sequence logic for the dominant sequence is provided in Table 2a.
  - The conditional cut sets for the dominant sequence are provided in Table 3.

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[1] Since this condition did not involve an actual initiating event, the parameter of interest is the measure of the incremental increase between the conditional probability for the period in which the condition existed and the nominal probability for the same period but with the condition nonexistent and plant equipment available. This incremental increase or "importance" is determined by subtracting the CDP from the CCDP. This measure is used to assess the risk significance of hardware unavailabilities especially for those cases where the nominal CDP is high with respect to the incremental increase of the conditional probability caused by the hardware unavailability.

Modeling Assumptions

- **Assessment summary**

  This operational event was modeled as an at-power condition assessment with the "C" CSIP inoperable. Two cases were analyzed: (1) the 59 days during which the "C" CSIP was in service as the "A" CSIP or "B" CSIP and (2) the 194 days during which the "C" CSIP was in the swing capacity as a backup to the "A" or "B" CSIP. The results for the second case were negligible.

  During the period "C" CSIP was inoperable, a separate TS violation (LER 400/00-006) occurred due to the failure of an isolation valve between the residual heat removal pump discharge and the CSIP suction header to open. However, an independent assessment determined this event to not be risk significant, and the recovery actions for the isolation valve are independent of the CSIPs.

  Version 2QA of the Standardized Plant Analysis Risk (SPAR) model (dated March 1998) for Shearon Harris (Ref.6) was used in the analysis.

- **Unique system and operational considerations**

  During normal plant operations, one CSIP is continuously operated to maintain the proper water inventory in the reactor coolant system, provide seal water flow to the reactor coolant pumps, and maintain chemistry and purity for the reactor coolant. A second CSIP is maintained in a standby condition in the event that ECCS is required. "A" and "B" CSIPs are the two pumps typically designated for these purposes. The third CSIP (usually "C" CSIP) serves as a backup (swing pump) to "A" and "B" CSIPs.

  For the cases in which the "C" CSIP was in service as the "A" or "B" CSIP, it is assumed that the pump the "C" CSIP is replacing is unavailable (e.g., undergoing extensive maintenance).

- **Basic event probability changes**

  Table 4 provides the basic events that were modified to reflect the event condition being analyzed. The bases for these changes are as follows:

  - The "C" CSIP was in service as the "A" CSIP for a period of 55 days and as the "B" CSIP for a period of 4 days. It is assumed that the "C" CSIP was replacing the "A" CSIP during the entire 59-day period and was the running pump. "B" CSIP is serving as the standby pump. "A" CSIP is assumed to be out of service for maintenance.

  - ***Probability of failure of the "C" CSIP (HPI-MDP-FC-1C).*** This value was set to "TRUE" (i.e., 1.0 failure probability) to reflect the independent failure of the "C" CSIP to provide flow during operations in which outboard thrust conditions exist.

  - ***Probability of failure of the "A" CSIP (HPI-MDP-FC-1A).*** This value was set to "TRUE" (i.e., 1.0 failure probability) to reflect that "A" CSIP is out of service (replaced by the swing "C" CSIP).

- ***Probability of common-cause failure (CCF) to start for "B" CSIP (HPI-MDP-CF-START).*** "A" CSIP and "C" CSIP failures do not involve a CCF. The CCF to start probability (HPI-MDP-CF-START) for the "B" CSIP was updated from the nominal value to 7.8×10^{-5} to account for operating experience reflected in Reference 7.

  The CCF equation ($CCF = \alpha_2 Q_T$) provided in Reference 8 was used to represent the probability for the cut set of "A" and "B" CSIPs suffering a CCF to start. The following values from References 6 and 7 were used: $Q_T = 3.0 \times 10^{-3}$ and $\alpha_2 = 2.6 \times 10^{-2}$.

- ***Probability of CCF to run for "B" CSIP (HPI-MDP-CF-RUN).*** The CCF to run probability (HPI-MDP-CF-RUN) for the "B" CSIP was modified from the nominal value to 2.0×10^{-5} to reflect that one pump ("A" CSIP) was out of service. In addition, it is assumed that the independent failure of the "C" CSIP does not affect the CCF of the system.

  The CCF equation ($CCF = [\alpha_2 + \alpha_3]Q_T$) provided in Appendix E of Reference 8 was used to represent the probability for the cut set of "B" and "C" CSIPs suffering a CCF to run and the probability for the cut set of all three pumps suffering a CCF to run. The following values from References 6 and 7 were used: $Q_T = 7.2 \times 10^{-4}$, $\alpha_2 = 2.0 \times 10^{-2}$, and $\alpha_3 = 8.4 \times 10^{-3}$.

- **Model update**

  The SPAR model for Shearon Harris was updated to account for updates of system/component failure probabilities and initiating event frequencies based on recent operating experience. These updates are independent of the actual event being analyzed. Bases for these updates are described in the footnotes to Table 4.

References

1. LER 400/00-007, Revision 0, *Technical Specifications Violation Due to Inoperable Charging Safety Injection Pump*, October 4, 2000 (ADAMS Accession No. ML003758321).

2. NRC Inspection Report, Shearon Harris Nuclear Power Plant, *NRC Integrated Inspection Report No. 50-400/00-03*, October 30, 2000 (ADAMS Accession No. ML010020381).

3. NRC Special Inspection Report, SDP/EA-00-263, Shearon Harris Nuclear Power Plant, *NRC Special Inspection Report No. 50-400/00-10, Preliminary White Finding*, December 11, 2000 (ADAMS Accession No. ML003776935).

4. NRC Notice of Violation, SDP/EA-00-263, Shearon Harris Nuclear Power Plant, *Final Significance Determination for a White Finding and Notice of Violation (NRC Inspection Report Nos. 50-395/00-03, 50-395/00-10), Shearon Harris Nuclear Power Plant*, February 2, 2001 (ADAMS Accession No. ML010360435).

5. Reserved.

6. M. B. Calley, et al., *Simplified Plant Analysis Risk Model for Shearon Harris (ASP PWR B)*, Rev. 2QA, Idaho National Engineering and Environmental Laboratory, March 1998.

7. F. M. Marshall, et al., *Common-Cause Failure Parameter Estimations*, NUREG/CR-5497, U. S. Nuclear Regulatory Commission, Washington, DC, October 1998.

8. F. M. Marshall, et al., *Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment*, NUREG/CR-5485, U. S. Nuclear Regulatory Commission, Washington, DC, November 1998.

9. G. M. Grant, et al., *Reliability Study: Emergency Diesel Generator Power System, 1987-1993*, NUREG/CR-5500, Vol. 5, U. S. Nuclear Regulatory Commission, Washington, DC, September 1999.

10. J. P. Poloski, et al., *Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995,* NUREG/CR-5750, U. S. Nuclear Regulatory Commission, Washington, DC, February 1999.

11. C. L. Atwood, et al., *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980-1996*, NUREG/CR-5496, U. S. Nuclear Regulatory Commission, Washington, DC, November 1998.

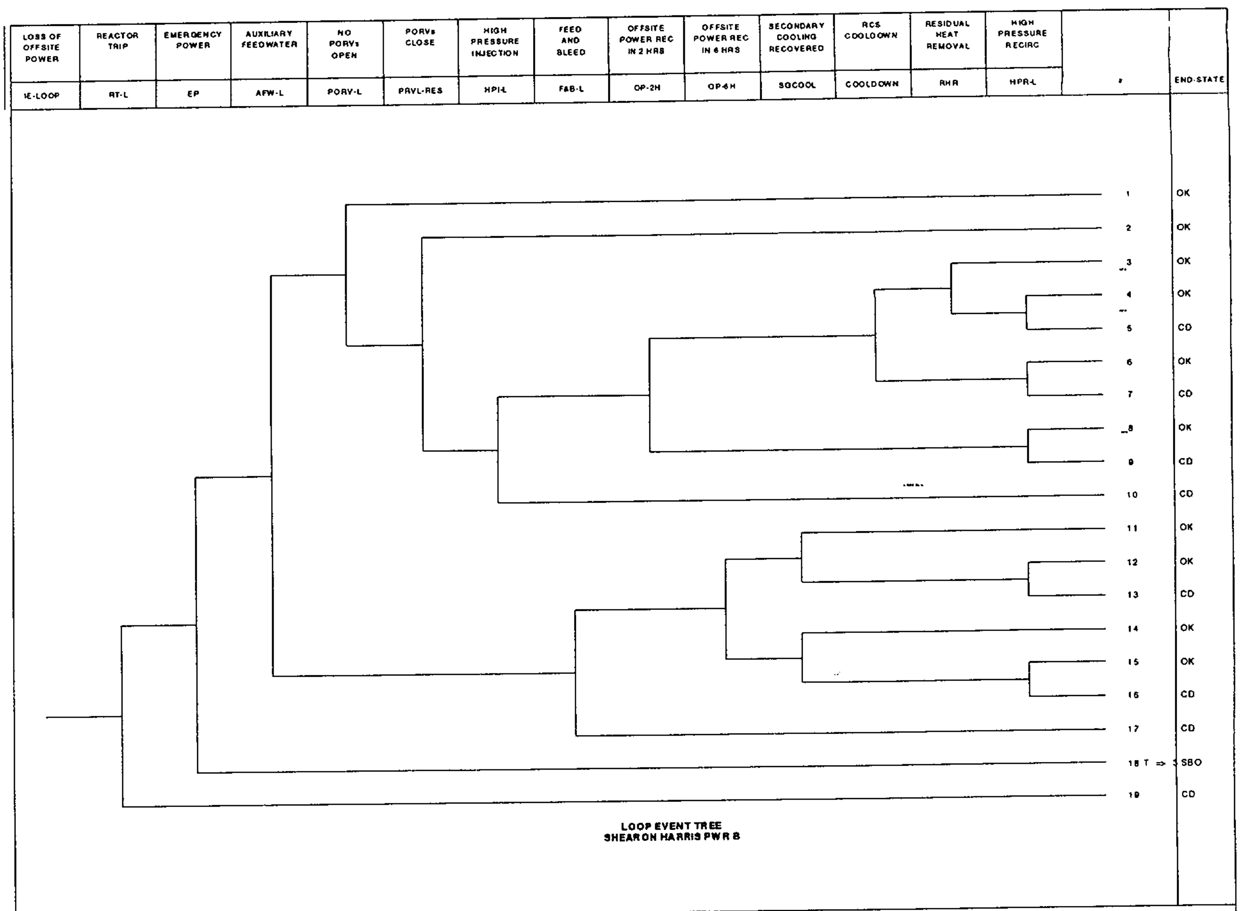

Figure 1. Loss of offsite power even tree. (The dominating sequence is sequence 10.)

Table 1. Conditional probabilities associated with the highest probability sequences.

Event tree name	Sequence no.	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP - CDP)
LOOP	10	3.5E-006	7.3E-008	—
Total (all sequences)[1]		**7.7E-006**	**3.5E-006**	**4.2E-006**

Note:
1. Total CCDP includes all sequences (including those not shown).

Table 2a. Event tree sequence logic for the dominant sequence.

Event tree name	Sequence no.	Logic ("/" denotes success; see Table 2b for fault tree names)
LOOP	10	/RT-L, /EP, /AFW-L, PORV-L, PRVL-RES, HPI-L

Table 2b. Definitions of fault trees listed in Table 2a.

AFW-L	NO OR INSUFFICIENT AUXILIARY FEEDWATER FLOW DURING LOSS OF OFFSITE POWER (LOOP)
EP	EMERGENCY POWER SYSTEM FAILS
HPI-L[1]	NO OR INSUFFICIENT FLOW FROM THE HIGH-PRESSURE INJECTION (HPI) SYSTEM DURING LOOP
PORV-L	PRESSURIZER POWER-OPERATED RELIEF VALVES/SAFETY RELIEF VALVES (PORVS/SRVS) OPEN DURING LOOP
PRVL-RES	PORVS AND BLOCK VALVES OR SRVS FAIL TO RESEAT DURING LOOP
RT-L	REACTOR FAILS TO TRIP DURING LOOP

Note:
1. Fault trees HPI-HD1 and HPI-HD1L were modified to properly represent the normal CSIP configuration. A basic event (HPE-XHE-XA-MDP1B) representing the failure of the operators to align the swing pump was incorrectly located under the gates (HPI-TRNB-F and HPI-TRNB-L) for the B train instead of the C train. The basic event was repositioned under the gates (HPI-TRNC-F and HPI-TRNC-L) for the C train.

Table 3. Conditional cut sets

CCDP	Percent contribution	Minimal cut sets[1]	
		Event Tree: LOOP, Sequence 10	
1.7E-006	49.4	EPS-DGN-FC-1B PPR-SRV-OO-PRV2	PPR-SRV-CO-L LOOP-10-NREC
1.7E-006	49.4	EPS-DGN-FC-1B PPR-SRV-OO-PRV3	PPR-SRV-CO-L LOOP-10-NREC
3.5E-006	**Total**[2]		

Notes:
1. See Table 4 for definitions and probabilities for the basic events.
2. Total CCDP includes all cut sets (including those not shown).

Table 4. Definitions and probabilities for modified or dominant basic events

Event name	Description	Probability/ frequency	Modified
EPS-DGN-CF-ALL	COMMON-CAUSE FAILURE OF DIESEL GENERATORS	7.0E-004	YES[2]
EPS-DGN-FC-1A	DIESEL GENERATOR A FAILS	5.1E-002	YES[3]
EPS-DGN-FC-1B	DIESEL GENERATOR B FAILS	5.1E-002	YES[3]
HPI-MDP-CF-RUN	HIGH-PRESSURE INJECTION (HPI) MOTOR DRIVEN PUMPS (MDPs) COMMON-CAUSE FAILURE TO RUN	2.0E-005	YES[1,6]
HPI-MDP-CF-START	HPI MDPs COMMON-CAUSE FAILURE TO START	7.8E-005	YES[1,7]
HPI-MDP-FC-1A	HPI TRAIN A FAILS	1.0E+000	YES[1]
HPI-MDP-FC-1C	HPI TRAIN C FAILS	1.0E+000	YES[1]
IE-LOOP	LOSS OF OFFSITE POWER (LOOP) INITIATING EVENT	5.9E-06/hr	YES[4]
IE-SGTR	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	8.0E-07/hr	YES[5]
IE-SLOCA	SMALL LOSS OF COOLANT ACCIDENT INITIATING EVENT	3.4E-07/hr	YES[5]
IE-TRAN	TRANSIENT INITIATING EVENT	1.6E-04/hr	YES[5]
LOOP-10-NREC	LOOP SEQUENCE 10 NONRECOVERY PROBABILITY	8.4E-001	NO
PPR-SRV-CO-L	PRESSURIZER POWER-OPERATED RELIEF VALVES/SAFETY RELIEF VALVES (PORVS/SRVS) OPEN DURING LOOP	1.6E-001	NO
PPR-SRV-OO-PRV2	PORV2 FAILS TO RECLOSE AFTER OPENING	3.0E-002	NO
PPR-SRV-OO-PRV3	PORV3 FAILS TO RECLOSE AFTER OPENING	3.0E-002	NO

Notes:

1. Basic event was changed to reflect event being analyzed.
2. Base case model was updated using results from NUREG/CR-5497, Tables 5-2 and 5-5 (Reference 7) and results from NUREG/CR-5500, Vol. 5, Tables C4, C6, and C7 (Ref.9).
3. Base case model was updated using results from NUREG/CR-5500, Vol. 5, Tables C4, C6, and C7 (Ref.9).
4. Base case model was updated using results from NUREG/CR-5750, Table H3 (Ref.10) and NUREG/CR 5496, Table B4 (Ref.11).
5. Base case model was updated using results from NUREG/CR-5750, Table 3-1 (Ref.10).
6. Basic event was changed to reflect event being analyzed. Change based on common-cause guidance provided in NUREG/CR-5485 (Ref.8). Alpha values was obtained from NUREG/CR-5497, Table 11-4 (Ref.7) and Q value from Table 4.1 of Rev. 2QA manual (Ref.6).
7. Basic event was changed to reflect event being analyzed. Change based on common-cause guidance provided in NUREG/CR-5485 (Ref.8). Alpha value was obtained from NUREG/CR-5497, Table 11-1 (Ref.7) and Q value from Table 4.1 of Rev. 2QA manual (Ref.6).