Final Precursor Analysis

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

Hope Creek Generating Station	Station Service Water Travelling Screen Failed Due to Inadequate Maintenance Instructions	
Event Date: July 1, 2003	**IR:** 50-354-03-06	**ΔCDP = 3×10^{-6}**

Operating Condition Summary

Operating condition Decription. On July 1, 2003, the plant was in Mode 1 (power operation). The Station Service Water system (SSWS) Train A traveling screen failure occurred. Before the traveling screen failure, the licensee replaced the head-shaft key for the traveling screen on June 20, 2003, using a maintenance procedure that existed at that time. During this maitenance, utility maintenance personnel failed to follow instructions for maintenance related to SSWS traveling screens. Later, USNRC Inspectors found and confirmed with the licensee that the maintenance procedure for the SSWS traveling screens did not include appropriate quantitative acceptance criteria to ensure that the travelling screen head-shaft key was installed correctly.

During the failure event of the SSWS Train A traveling screen, other remaining traveling screens did not fail and were operating when the reactor was at power. So, the licensee did not generate and issue a Licensee Event Report (LER) to USNRC.

Because USNRC inspectors found that a violation of 10 CFR PART 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings", Region I generated and issued an inspection report (Reference 1). Office of Enforcement issued a final significance determination process (SDP) finding letter (Reference 2) for the same event.

Cause. During a maintenance event prior to failure, maintenance workers did not use adequate maintenance procedure and cut the key too short. Inadequate maintenance procedure for traveling screen resulted in a failure of the SSWS Train A traveling screen.

Condition duration. USNRC inspectors documented all applicable maintenance-related failure events for the SSWS Train A traveling screen during a one year period prior to July 1, 2003 failure event (Reference 1). A systematic accounting of these related-failure events and durations that were observed during a one year period prior to July 1, 2003, was conducted. Based on this, a total condition duration of 504 hours is estimated.

Related event. None.

Recovery opportunity. Given an initiating event (e.g., loss of offsite power), SSWS Train A traveling screen failure would have been non-recoverable.

Other related conditions or events. A review of the licensee-issued LERs during a one year period prior to the failure event indicates that more than two safety relief valve were significantly leaking beyond the limits of plant technical specifications for a longer period. These SRVs might not have been seated adequately after lifting for pressure relief during a transient or a LOOP event.

Analysis Results

- **Importance**[1]

  The risk significance of the failed SSWS Train A traveling screen due to inadequate maintenance procedures for a condition duration of 504-Hours was determined by subtracting the nominal core damage probability (point estimate) from the conditional core damage probability (point estimate):

  Conditional core damage probability (CCDP) = 4.0E-6
  Nominal core damage probability (CDP) = 5.2E-7
  Importance (ΔCDP = CCDP - CDP) = 3.4E-6

  The estimated importance (CCDP-CDP) for the operating condition was 3.4E-6.

  A uncertainty analysis was conducted for the operating condition. The mean estimates for CCDP, CDP, and importance were 4.5E-06, 5.8E-07, and 3.9E-06 respectively.

- **Dominant sequence**

  Loss of offsite power (LOOP) event followed by successful reactor scram, failures of the Emergency Power system, successful closure of safety relief valves, successful Reactor Core Isolation Cooling system, and failure of offsite AC power recovery in 5 hours.

  Sequence LOOP 25-13; Importance was estimated to be 2.1E-6. The events and important component failures in this sequence were as follows:

  - Initiating event (LOOP)
  - Mitigating system failures and successes
    - successful reactor scram
    - failures of the Emergency Power system,
    - successful closure of safety relief valves,

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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 change 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 change 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 operating conditions where the nominal CDP is high with respect to the incremental change of the conditional probability caused by the hardware unavailability.

- successful Reactor Core Isolation Cooling system, and
- failure of offsite power recovery in 5 hours failures.

- Onset of potential core damage

Paths for dominant sequence LOOP 25-13 is shown Figure 1.

- **Results tables**

– Table 1 provides the conditional probabilities for 2 dominant sequences.
– Table 2a provides the event tree sequence logic for the dominant sequences listed in Table 1.
_ Table 2b provides the definitions of fault trees used in event tree logic listed in Table 2a.
– Table 3 provides the conditional (CCDP) cut sets for 2 dominant sequences.
– Table 4 provides the definitions and probabilities for added basic events and condition-affected basis events.

Modeling Assumptions

- **Assessment summary**

Assessment type -

A total of four repeated maintenance-related failure events were observed during a period between July 1, 2002 and July 1, 2003. Duration estimates for these four separate maintenance-related events are: 1. 5/26/2003 and 5/30/2003, 2. 06/20/03 and 06/26/03, 3. 06/28/03 and 07/01/03, and 4. 07/01/03 and 07/09/03. The total period for these four separate events is estimated to be 504 hours. Therefore, the operating condition was modeled as an at-power condition assessment with the SSWS Train A travelling screen failure for a total period 504-hour period.

Condition modeling and related assumptions-

1. Although no maintenance was performed on traveling screens for other SSWS trains, a similar screen failure could have occurred due to the same cause of maintenance failure. Modeling of this failure-dependent maitenacne and testing of other similar trains is beyond the scope of this precursor analysis report.

2. Screen failure events may not be recoverable within 6 hours of an initiating event.

Model use - The Revision 3.11 Standardized Plant Analysis Risk (SPAR) model for Hope Creek Nuclear Generating Station (Reference 4) was used for this condition assessment.

The Revision 3.11 SPAR model includes event trees for general transients, loss of feedwater, loss of offsite power, and loss of condenser heat removal, small loss-of-coolant accident, loss of a support system (e.g., service water, one Train DC bus), and other LOCA events (medium LOCA, large LOCA, interfacing system LOCAs). The Revision 3.11 SPAR plant model includes fault tree models for many frontline systems (e.g., the HPCI system, the RCIC system, the RHR system) and other support systems (e.g., Station Service Water system) modeled in the event trees and fault trees, including the suppression pool cooling function and the containment venting function. This version of the plant model also reflects recent operating experience (e.g., reduced LOOP frequency estimate, reduced transient frequency, reduced EDG maintenance unavailabilities, reduced EDG common cause failure probabilities). It is noted that Revision 3.11 plant model update for Hope Creek SPAR 3 plant models did not include models for other initiating events such as loss of instrument air (IE-LOIA) and loss of single AC bus (IE-LOACBUS). These initiating events would increase the total importance risk measure for the operating condition that is being evaluated.

- Model update to Revision 3.01 SPAR model

  No model change was performed to Revision 3.11 SPAR model for Hope Creek (December 31, 2004 Version).

- Basic event probability changes

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

  SSW-TSA-PG-TRN1A (SWS TRAIN A TRAVELLING SCREEN FAILURE)- This was set to TRUE to reflect the operating condition. IR concluded that the travelling screen failure was not recoverible.

  IE-LOSWS (loss of standby service water system (SSWS) iniitiating event)- This was set to a value of 6E-3 per year to reflect the fact that the initiating event might have increased during the condition period. This value was estimated based on the ratio of (unavailability of one train of the SSWS to unavailability of all trains of the SSWS) and the baseline initiating event frequency IE-LOSWS.

- Sensitivity analyses -

  A total of three repeated maintenance failure events were observed during a period between July 1, 2002 and July 1, 2003. The third maintenance failure was closely observed during a period of 219 hours. Region I inspectors conducted an evaluation of this failure duration and issued a final WHITE SDP letter to the licensee. So, one sensitivity analysis was conducted in evaluating the impact of the third condition duration of 219 hours. The impact of this third duration on the importance (increase in core damage frequency) risk measure was found to be 1.5E-6.

- Uncertainty analysis and range for total importance due to operating condition-

  The parameter estimates and the uncertainties regarding the numerical estimates of the parameters used in the model (parameter uncertainty) are calculated. These data and uncertainty distributions are then propagated through the modified version of the Revision 3.11 SPAR model for Hope Creek (Reference 4) to produce uncertainty estimates.

Uncertainty analysis of the operating condition along with parameters was performed using the SAPHIRE code (Version 7.22). Default distribution types for applicable initiating events (e.g. loss of offsite power) and basic events for components were documented in the Revision 3.11 SPAR model for Hope Creek. These uncertainty estimates and uncertainty estimates for condition-affected basic events were used in estimating mean condition-CDP values and mean condition-CCDP values. Other statistical values such as point estimates, 5% estimates, and 95% estimates were also calculated for CDP and CCDP analysis cases. Estimated statistical values for the operating condition are shown in Table 5.

References

1. USNRC, Region I, “HOPE CREEK NUCLEAR GENERATING STATION, NRC INSPECTION REPORT 50-354/2003-006” dated April 20, 2004. ADAMS ACCESSION NUMBER ML 041120166.

2 USNRC, Region I, “HOPE CREEK NUCLEAR GENERATING STATION, NRC INTEGRATED INSPECTION REPORT 50-354/2003-006” dated February 11, 2004. ADAMS ACCESSION NUMBER ML040420422.

3 USNRC Office of Enforcement, “FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING (NRC INSPECTION REPORT 50-354/2003-006) (HOPE CREEK NUCLEAR GENERATING STATION) - EA-04-086” dated May 10, 2004. ADAMS ACCESSION NUMBER ML041320205

4 Richard Gregg, et al., “Standardized Plant Analysis Risk (SPAR) Model for Hope Creek Nuclear Generating Station” by Idaho National Engineering and Environmental Laboratory, 12/31/2004.

Table 1. Conditional probabilities (point values) for dominant sequences

Event tree name	Sequence no.	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP - CDP)[2]
LOOP	25-13	2.2E-6	1.3E-7	2.1E-6
LOSWS	36	4.3E-7	2.9E-8	4.1E-7
Total (all sequences)[1]		4.0E-6	5.2E-7	3.4E-6

Notes:
1. Total CCDP and CDP includes all sequences (including those not shown in this table).
2. Importance is calculated using the total CCDP and total CDP from all sequences of all applicable event trees. Sequence level importance measures are not additive.

Table 2a. Event tree sequence logic for dominant sequences

Event tree name	Sequence No.	Logic ("/" denotes success; see Table 2b for top event names)
LOOP	25-13	(/RPS)*(EPS)*(/SRV)*((/RCI1)*(OEP-5HR)
LOSWS	36	(/RPS)*(/SRV)*(SSWR)*(/HPI01)*(/DEP)*(CVS)*(LI02)

Table 2b. Definitions of fault trees used in event tree logic listed in Table 2a

LOSWS	LOSS OF SERVICE WATER
LOOP	LOSS OF OFFSITE POWER
RPS	REACTOR SHUTDOWN FAILS
EPS	ELECTRICAL POWER SYSTEM FAILS
SRV	SRV'S CLOSE
RCI1	REACTOR CORE ISOLATION COOLING SYSTEM FAILS
OEP-5HR	OFFSITE POWER RECOVERY IN 5 HOURS FAILS
SSWR	OPERATOR FAILS TO RECOVER FROM A LOSS OF SERVICE WATER
HPI01	HIGH PRESSURE INJECTION IS UNAVAILABLE (LOSWS-FTF FLAGS)
DEP	MANUAL REACTOR DEPRESSURIZATION FAILS
LI02	LATE INJECTION IS UNAVAILABLE (CONTAINMENT FAILED)
CVS	CONTAINMENT (SUPPRESSION POOL) VENTING

Notes:
1. "/" Iindicates that top event is a success event in the event tree logic.

Table 3a. CCDP cut sets for LOOP Sequence 25-13

CCDP	Percent contribution	Minimal cut sets[1]
Event Tree: LOOP, Sequence 25-13		
1.036E-6	**45.53**	**SSW-TSA-CF-ALL * ACP-XHE-XE-GTRB * OEP-XHE-XL-NR05H**
9.781E-8	**4.23**	**ACP-XHE-XE-GTRB * EPS-DGN-FR-DGB * EPS-DGN-FR-DGC * OEP-XHE-XL-NR05H**
2.200E-6	Total[2]	

Table 3b. CCDP cut sets for LOSWS Sequence 36

CCDP	Percent contribution	Minimal cut sets[1]
Event Tree: LOSWS Sequence 36		
1.151E-7	**27.04**	**CVS-XHE-XE-VENT * SSW-XHE-XL-NOREC**
1.036E-7	**24.34**	**SSW-XHE-XL-NOREC * CVS-AOV-CC-N2LIN**
4.300E-7	Total[2]	

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

Table 4 - Definitions and probabilities for added basic events and condition-affected basis events

Basic event name	Description	Added to base Model	Probability	Modified to reflect condition	Note
SSW-TSA-CF-ALL	SWS TRAVELING SCREENS FAIL FROM COMMON CAUSE	NO	4.800E-3	NO	
ACP-XHE-XE-GTRB	OPERATOR FAILS TO START AND ALIGHN GAS TURBINE	NO	6.000E-1	NO	
OEP-XHE-XL-NR05H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 5 HOURS	NO	1.900E-1	NO	
SSW-XHE-XL-NOREC	OPERATOR FAILS TO RECOVER FROM A LOSS OF SERVICE WATER	NO	3.400E-1	NO	
EPS-DGN-FR-DGB	DIESEL GENERATOR B FAILS TO RUN	NO	2.117E-2	NO	
EPS-DGN-FR-DGC	DIESEL GENERATOR C FAILS TO RUN	NO	2.117E-2	NO	
CVS-XHE-XE-VENT	OPERATOR FAILS TO VENT CONTAINMENT	NO	1.000E-3	NO	
CVS-AOV-CC-N2LIN	NITROGEN SUPPLY LINE FAILS	NO	9.000E-4	MO	
SSW-TSA-PG-TRN1A	SWS TRAIN A TRAVELING SCREEN	NO	TRUE	YES	1
LOSWS	LOSS OF SERVICE WATER SYSTEM INITIATOR	NO	6.000E-3	YES	1

NOTE:

1. Basic event probability was changed to reflect the operating condition. Bases for change was documented in Basic event probability changes section of this report. Changes were made in accordance with SAPHIRE/SPAR 3 common cause failure model assumptions and parameters.

Table 5 - Uncertainty estimates for the operating condition

Plant: Hope Creek Nuclear Generating Station
IR ID: 50-354/2003-006
SDP: EA-04-086
LER ID : None.

Analysis type = Monte Carlo
Samples = 10000; Seeds = 97453

Initiating event (IE)	IE ID	Point estimate	mean estimate	5% estimate	50% estimate	95% estimate
All internal initiating events	CCDP for 1 year	6.873E-05	7.933E-05	3.823E-06	3.393E-05	2.711E-04
All internal initiating events	CDP for 1 year	9.110E-06	1.020E-05	5.998E-07	4.353E-06	3.737E-05
All internal initiating events	CCDP for 504 hours	3.954E-06	4.564E-06	2.200E-07	1.952E-06	1.560E-05
All internal initiating events	CDP for 504 hours	5.241E-07	5.868E-07	3.451E-08	2.504E-07	2.150E-06
All internal initiating events	Importance for 504 hours	3.430E-06	3.977E-06	1.854E-07	1.702E-06	1.345E-05

Figure 1 - Hope Creek Nuclear Generating Station - LOOP Event Tree Showing Sequence 25-13

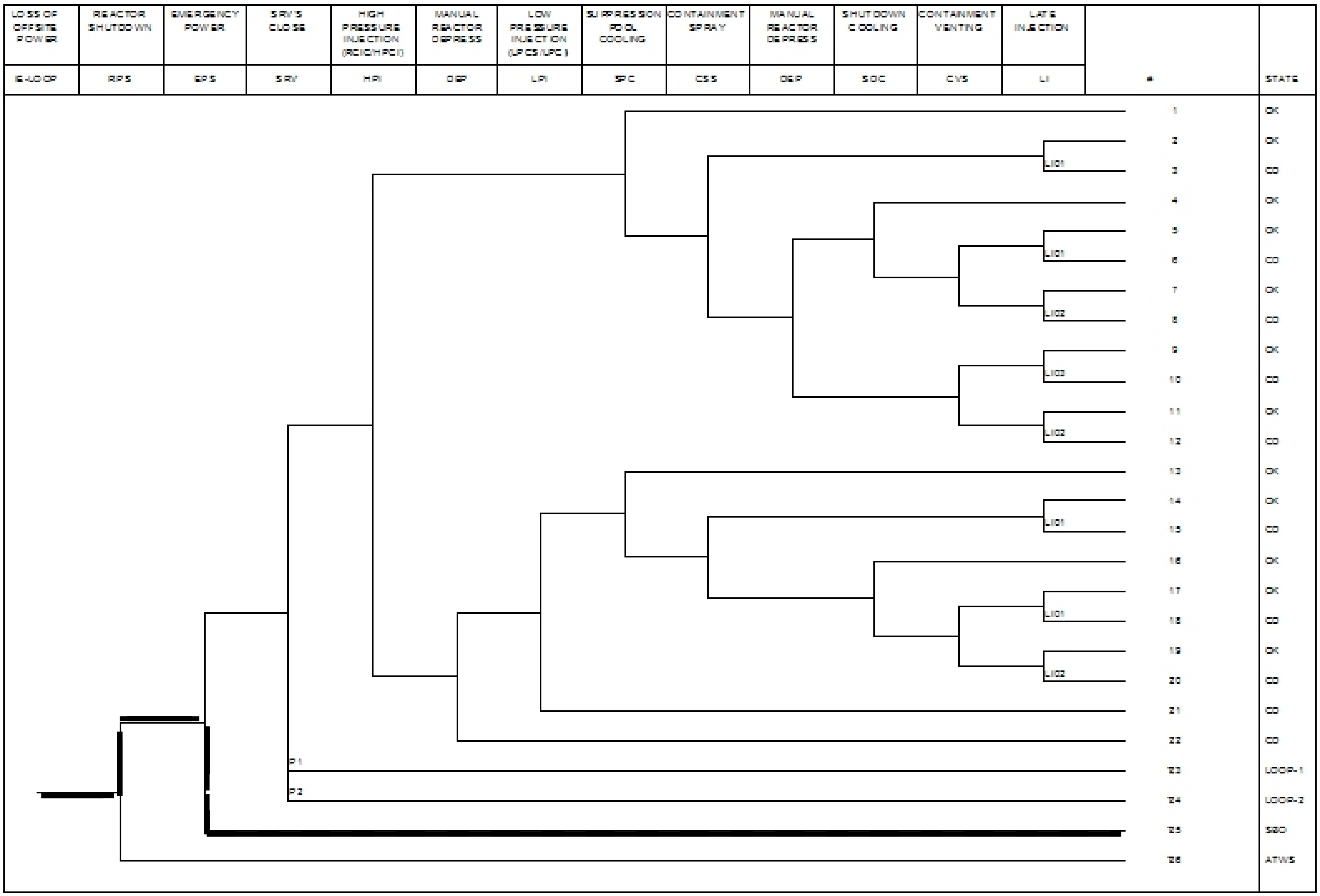

Figure 1 - Hope Creek Nuclear Generating Station - LOOP Event Tree Showing Sequence 25-13 (Continued.)