Final Precursor Analysis

Accident Sequence Precursor Program -- Office of Nuclear Regulatory Research

Columbia Generating Station	Reactor Scram During Plant Start-up Due to Reactor Feedwater Pump Trip		
Event Date 06/23/2005	LER 397/05-004	CCDP	1.2 x 10^{-5}

June 8, 2006

Event Summary

Description

On June 23, 2005, Columbia Generating Station was in Mode 1, with the reactor operating at approximately 23 percent power. At 1346 PDT, an automatic scram occurred due to a low water level condition in the reactor vessel. The low reactor water level condition was caused by an inadvertent loss of reactor feedwater pumps (RRW-P-1B) due to a false low suction pressure signal caused by human error during planned maintenance activities. Control room operators entered appropriate Emergency Operating Procedures and stabilized the plant following the reactor scram. Plant systems responded as designed with the exception of Reactor Core Isolation Cooling (RCIC), as discussed below.

The RCIC system was manually started to restore reactor water level and was later manually tripped. The system had to be reset locally due to tripped mechanical overspeed trip linkage. During two subsequent attempts to restart RCIC, the pump tripped on low suction pressure. Operators were then able to successfully start RCIC with the flow controller in manual. A time delay has been added to the RCIC low suction pressure trip to resolve this issue.

Cause

The physical cause of the reactor scram was a loss of feedwater to the reactor vessel. The loss of feedwater was caused by a false low suction pressure signal input to RFW-P-1B. This false signal was generated when an electrician briefly connected a multimeter, set to measure ohms, across two termination points, introducing a voltage that resulted in a rapidly reduced feel flow rate to the reactor vessel. The electrician was removing a bypass of the low suction pressure trip logic using an approved procedure, but did not have a full understanding of the configuration he was restoring, and he connected across the wrong termination points. The bypass had been installed to avoid a potential inadvertent trip of the feed pumps during feedwater heater restoration, following replacement of three condensate system relief valves that had lifted during plant start-up evolutions.

The unanticipated trips of the RCIC system were determined to have been caused by a short-duration low suction pressure condition that was a product of system design. This condition occurred after the pump started, just as the pump discharge pressure exceeded reactor vessel pressure, and water started being pumped into the reactor vessel. As the RCIC discharge check valves rapidly open once pump discharge pressure exceeds reactor vessel pressure, a short-duration negative pressure spike is experienced throughout the system. Normally, the suction pressure of RCIC is maintained high enough by keep-fill pump operation that this negative pressure spike does not actuate the low suction pressure switch. However, shortly after the RCIC pump is shut down following operation, the suction pressure is lower if the keep-fill pump has not operated, and the short-duration negative pressure spike can actuate the low suction pressure trip circuit and cause the RCIC pump to trip.

A review of operating history was performed by the licensee. This review found that there were several time periods where RCIC had operated and been secured, but the keep-fill pump had not operated and pressurized the suction piping. During those time periods it is possible that the RCIC pump would have experienced a low suction pressure trip if called upon to perform its function following a control rod drop accident. Final cause investigation is not complete for this issue. The licensee will issue a supplemental LER following completion of the investigation, if further information of significance warrants.

Condition Duration

As far as one can tell from the available information, the problem with RCIC has existed since the licensing of the plant.

Other Related Conditions or Events During the Condition Period

On June 15, 2005, at 1400, the reactor tripped from 100% power (Reference 2). The trip resulted from a Reactor Protection System (RPS) actuation caused by a failure in the digital electro-hydraulic (DEH) system, which caused the four main turbine throttle valves to spuriously stroke from fully open to fully closed. No specific DEH system failure could be identified, but the three circuit cards providing the control signals to the turbine throttle valves were replaced, as they were identified as the most likely cause of the failure.

This event was complicated by the fact that 19 minutes later, all four throttle valves reopened with no operator action. During the time from the reactor trip to the reopening of the throttle valves, the main turbine failed to trip as designed. Plant operators manually tripped the turbine at 1430, 30 minutes after the initial reactor trip.

Because the CCDP of this event is less than that associated with an uncomplicated loss of main feedwater, no detailed analysis was performed.

Analysis Results

- **Conditional Core Damage Probability (CCDP)**

The point estimate CCDP for the loss-of-feedwater initiating event is 1.2×10^{-5}. To put this value in perspective, the CCDP for a loss of feedwater with no complicating factors is 4.6×10^{-6}.

The uncertainty distribution for CCDP is summarized below.

	5%	Mean	Point Estimate	95%
CCDP (LOMFW)	2.1×10^{-7}	1.3×10^{-5}	1.2×10^{-5}	5.5×10^{-5}

- **Dominant Sequences**

  There are 24 sequences from the LOMFW event tree that contribute to the overall CCDP. There are three dominant accident sequences which collectively contribute about 96% of the CCDP. These sequences are listed in Table 1.

  The three dominant sequences are initiated by a loss of feedwater, with the main condenser remaining available as a heat sink in the top two sequences; however, condensate is assumed to be failed in the SPAR model, and high-pressure injection from either RCIC or High-Pressure Core Spray (HPCS) is required in the short term because of discharge of the safety/relief valves (SRVs) into the suppression pool. In Sequence 5 (52%), both RCIC and HPCS fail, the operators successfully depressurize the reactor to allow low-pressure sources to inject, but all low-pressure injection sources fail, leading to core damage.

  Sequence 6 (27%) is similar to Sequence 5, except that the reactor is not successfully depressurized, leading to core damage at high pressure following the failure of RCIC and HPCS.

  In sequence 12 (17.5%), the main condenser is not available as a heat sink, HPCS functions to provide high-pressure injection, but decay heat removal fails, leading to the need to vent the containment to relieve pressure. However, containment venting is unsuccessful, leading to containment failure and subsequent failure of long-term injection.

- **Results Tables**

  - The conditional probabilities for the dominant sequences are shown in Table 1.
  - The event tree sequence logic for the dominant sequences are presented in Table 2a.
  - Table 2b defines the nomenclature used in Table 2a.
  - The most important cut sets for the dominant sequences are listed in Table 3.

- Definitions and probabilities for modified or dominant basic events are provided in Table 4.

Modeling Assumptions

- **Analysis Type**

  The loss-of-feedwater event was analyzed as an initiating event assessment involving the occurrence of a reactor trip caused by loss of feedwater (IE-LOMFW). The evaluation was performed using the *Initiating Event Assessment* feature of the GEM 7.26 software package. The Columbia model utilized was the Revision 3.31 Columbia SPAR model (Reference 3). The initiating event representing loss of feedwater (IE-LOMFW) was set to TRUE for the assessment.

- **Unique Design Features**

  Columbia is a BWR/5 reactor in a Mark II containment. There are no unique design features relevant to this event.

- **Modeling Assumptions Summary**

  Key modeling assumptions. The key modeling assumptions are listed below and discussed in detail in the following sections. These assumptions are important contributors to the overall risk.

  – **Loss of Feedwater Initiating Event**

  RCIC did not receive an automatic start signal when reactor pressure vessel (RPV) level dropped to the Level 2 set point. As described in Reference 1, this was because of a mismatch in level instrumentation set point tolerances. Reference 1 indicates that Level 2 was not actually reached in the RPV during this event. Operators manually started RCIC during this event. For purposes of this assessment, **it is assumed that RCIC would have received an automatic start signal when RPV level reached the Level 2 set point.**

  During the actual event, the operators had to make several attempts to restart RCIC (after manually securing it). This difficulty was captured in the SPAR model by setting event **RCI-RESTART** to TRUE (1.0).

  The SPAR model contains a dependence of HPCS upon room cooling, such that failure of room cooling, even in the short term, leads to failure of HPCS. This assumption was questioned with the Idaho National Laboratory (INL), and was found to be supported by the modeling in the licensee’s PRA. Therefore, this dependence was retained.

  Similarly, a question was raised about impacts of strainer plugging in the Standby Service Water System (SSW), which acts as a heat sink for HPCS room cooling. INL

indicated that the licensee's PRA shows a common suction strainer for SSW pumps that take suction from Spray Pond "A." This level of detail is not shown in the Final Safety Analysis Report for Columbia. Based on this information, no changes were made to the SSW portion of the SPAR model.

– **Condition Duration**

NA as this is an initiating event assessment.

- **Basic Event Probability Changes**

The loss-of-feedwater initiating event (**IE-LOMFW**) was set to TRUE.

To model the need to restart RCIC, event **RCI-RESTART** was set to TRUE.

To model the failure of RCIC to restart because of the low suction pressure trip, event **RCI-TDP-FS-RSTRT** was set to TRUE.

The HEP for event **RCI-XHE-XL-RSTRT** was set to 0.12, a value derived by the Region IV Senior Reactor Analyst as part of a Phase 3 Significance Determination Process assessment for this event (Ref. 4). The uncertainty distribution for this event was changed to a constrained noninformative prior, the distribution used by SPAR-H to represent epistemic uncertainty.

- **Sensitivity Analyses**

Dependence of HPCS Upon Room Cooling

The dependence of HPCS upon room cooling, with SSW as the room-cooling heat sink was removed from the SPAR model to examine the impact of this assumption on CCDP. This was done by inserting a flag event into the HCS fault tree that would act as a switch to turn on and off the room-cooling dependence. With the dependence on room cooling removed, the CCDP was found to be 4.8×10^{-6}. Thus, the assumption of dependence of HPCS upon room cooling has a significant impact on the results.

Reduced HEP for RCIC Restart

The HEP for failure to restart RCIC (**RCI-XHE-XL-RSTRT**) was lowered by a factor of 10, to 0.012. The resulting CCDP is 5.3×10^{-6}.

Combination

Removing the dependence of HPCS upon room cooling and lowering the HEP for failure to restart RCIC by a factor of 10 reduced the CCDP for this event to 3.1×10^{-6}.

References

1. Columbia Licensee Event Report 397/2005004, August 22, 2005.

2. Columbia Licensee Event Report 397/2005003, August 15, 2005.

3. Columbia_331_Prerelease, obtained from http://saphire.inl.gov.

4. Columbia Integrated Inspection Report 397/2005005, February 13, 2006, ADAMS Accession Number ML0604406270.

Table 1. Conditional core damage probabilities of dominating sequences.

Event tree name	Sequence no.	CCDP[1]	Contribution
LOMFW	05	6.2×10^{-6}	51.7%
LOMFW	06	3.2×10^{-6}	26.7%
LOMFW	12	2.1×10^{-6}	17.5%
Total (all sequences)[2]		**1.2×10^{-5}**	**100%**

1. Values are point estimates.
2. Total CCDP includes all sequences (including those not shown in this table).

Table 2a. Event tree sequence logic for dominating sequences.

Event tree name	Sequence no.	Logic ("/" denotes success; see Table 2b for top event names)			
LOMFW	05	/RPS RCI	/SRV /DEP	/CND LPI	HCS VA
LOMFW	06	/RPS RCI	/SRV DEP	/CND	HCS
LOMFW	12	/RPS SPC	/SRV PCSR	CND CVS	/HCS LI01

Table 2b. Definitions of top events listed in Table 2a.

Top Event	Definition
CND	Main Condenser
CVS	Containment Venting
DE3	Manual Reactor Depress
DEP	Manual Reactor Depress
HCS	HPCS
LC12	LPCI
LCS	LPCS
LI01	Developed Event
LI02	Developed Event
LPI	Low Pressure Injection (LPCS or LPCI)
LVL	Reactor Level Control

Top Event	Definition
NX	Inhibit ADS
P2	Two or more stuck open SRVs
PC2	Power Conversion System
PCSR	Power Conversion System Recovery
PPR	SRV's open
RCI	RCIC
RPS	Reactor Protection System
RRS	Recirc Pump Trio
SLC	Standby Liquid Control
SPC	Suppression Pool Cooling
SRV	SRV's Close
TAF	Transfer Branch ATWS-1
VA	Alternate Rod Injection

Table 3a. Conditional cut sets for the dominant sequences.

CCDP	Percent Contribution	Minimum Cut Sets (of basic events)
Event Tree: LOMFW Sequence 05		
1.6×10^{-6}	26.4	RCI-XHE-XL-RSTRT SSW-STR-CF-1A1B
2.7×10^{-7}	4.4	RCI-XHE-XL-RSTRT SSW-MOV-CF-2AB29
2.3×10^{-7}	3.7	SSW-MDP-TM-1B RCI-XHE-XL-RSTRT SSW-STR-PG-1A
2.3×10^{-7}	3.7	RCI-XHE-XL-RSTRT SSW-MDP-TM-P2 SSW-CKV-CF-1AB
2.2×10^{-7}	3.5	SSW-MDP-CF-START2-2 RCI-XHE-XL-RSTRT SSW-MDP-TM-P2
6.2×10^{-6}	100.0	**Total (all cutsets)**[1]

1. Total Importance includes all cutsets (including those not shown in this table).

CCDP	Percent Contribution	Minimum Cut Sets (of basic events)
Event Tree: LOMFW Sequence 06		
1.2×10^{-6}	37.1	ADS-XHE-XE-MDEPR RCI-XHE-XL-RSTRT SSW-MDP-TM-P2
4.2×10^{-7}	13.0	ADS-XHE-XE-MDEPR HCS-MDP-TM-HPCS RCI-XHE-XL-RSTRT
1.5×10^{-7}	4.6	ADS-XHE-XE-MDEPR RBC-FAN-FS-4 RCI-XHE-XL-RSTRT
1.2×10^{-7}	3.7	ADS-XHE-XE-MDEPR RBC-FAN-TM-4 RCI-XHE-XL-RSTRT
1.2×10^{-7}	3.7	ADS-XHE-XE-MDEPR RCI-XHE-XL-RSTRT SSW-MDP-FS-P2
3.2×10^{-6}	100.0	**Total (all cutsets)**[1]

1. Total Importance includes all cutsets (including those not shown in this table).

CCDP	Percent Contribution	Minimum Cut Sets (of basic events)
Event Tree: LOMFW Sequence 12		
1.4×10^{-6}	66.73	PCS-XHE-XL-LTLMFW FWS-XHE-XM-ERROR SSW-MDP-TM-1B RBC-FAN-FR-12 CFAILED01
1.8×10^{-7}	8.34	PCS-XHE-XL-LTLMFW FWS-XHE-XM-ERROR RBC-FAN-FR-12 SSW-FAN-FS-PRA1B CFAILED01
1.4×10^{-7}	6.67	PCS-XHE-XL-LTLMFW FWS-XHE-XM-ERROR SSW-MDP-FS-1B RBC-FAN-FR-12 CFAILED01
2.1×10^{-6}	100.0	**Total (all cutsets)**[1]

1. Total Importance includes all cutsets (including those not shown in this table).

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

Event Name	Description	Probability/ Frequency (per hour)	Modified
ADS-XHE-XE-MDEPR	OPERATOR FAILS TO DEPRESSURIZE THE REACTOR	5.0E-004	
CFAILED01	CONTAINMENT FAILURE CAUSES LOSS OF HPCS INJEC	3.3E-001	
FWS-XHE-XM-ERROR	OPERATOR FAILS TO ALIGN FIREWATER INJECTION	1.0E+000	
HCS-MDP-FR-HPCS	HPCS PUMP FAILS TO RUN	5.2E-004	
HCS-MDP-FS-HPCS	HPCS PUMP FAILS TO START	1.5E-003	
HCS-MDP-TM-HPCS	HPCS PUMP IS UNAVAILABLE DUE TO MAINTENANCE	7.0E-003	
HCS-MOV-CC-V12	HPCS RECIRC VALVE MOV-12 FAILS TO OPEN	1.0E-003	
HCS-MOV-CC-V15	HPCS SUCTION TRANSFER VALVE FAILS TO OPEN	1.0E-003	
HCS-MOV-CC-V4	HPCS INJECTION VALVE FAILS TO OPEN	1.0E-003	
HCS-XHE-XR-HPCS	OPERATOR FAILS TO RESTORE HPCS	1.0E-003	
OPR-XHE-XE-HPINJ	OPERATOR FAILS TO ALIGN PCS	1.0E-003	
OPR-XHE-XE-LPINJ	OPERATOR FAILS TO START/CONTROL LP INJECTION	5.0E-004	
OPR-XHE-XM-ALPI4	OPERATOR FAILS TO ALIGN ALTERNATE LOW PRESSUR	1.4E-001	
PCS-XHE-XL-LTLMFW	OPERATOR FAILS TO RECOVER PCS IN THE LONG-TER	1.0E+000	
RBC-FAN-FR-12	RBEC-FAN 12 (WMA-AH-53A) FAILS TO RUN	2.2E-004	
RBC-FAN-FS-4	RBEC FAN 4 FAILS TO START	2.5E-003	
RBC-FAN-TM-4	RBEC FAN 4 IS UNAVAILABLE DUE TO MAINTENANCE	2.0E-003	

Event Name	Description	Probability/ Frequency (per hour)	Modified
RBC-XHE-XR-4	OPERATOR FAILS TO RESTORE RBEC FAN 4	1.0E-003	
RCI-XHE-XL-RSTRT	OPERATOR FAILS TO RECOVER RCIC FAILURE TO RES	1.2E-001	YES[1]
RCI-RESTART	RESTART OF RCIC IS REQUIRED	1.0E+000	YES[1]
RCI-TDP-FS-RSTRT	RCIC FAILS TO RESTART GIVEN	1.0E+000	
SSW-CKV-CF-1AB	CCF OF STANDBY SERVICE WATER PUMP DISCHARGE C	9.5E-005	
SSW-FAN-FS-PRA1B	SSW FAN 1B (PRA-FN-1B) FAILS TO START	2.5E-003	
SSW-MDP-CF-START2-2	CCF OF STANDBY SERVICE WATER PUMPS TO START	9.0E-005	
SSW-MDP-FR-1B	STBY SERVICE WATER PUMP 1B FAILS TO RUN	5.2E-004	
SSW-MDP-FR-P2	STBY SERVICE WATER PUMP HPCS-P2 FAILS TO RUN	5.2E-004	
SSW-MDP-FS-1B	STBY SERVICE WATER PUMP 1B FAILS TO START	2.0E-003	
SSW-MDP-FS-P2	STBY SERVICE WATER PUMP HPCS-P2 FAILS TO STAR	2.0E-003	
SSW-MDP-TM-1B	SSW MDP 1B UNAVAILABLE DUE TO TEST AND MAINTE	2.0E-002	
SSW-MDP-TM-P2	SSW MDP HPCS-P2 UNAVAILABLE DUE TO TEST AND M	2.0E-002	
SSW-MOV-CC-12B	STANDBY SERVICE WATER PUMP 1B DISCHARGE MOV 1	1.0E-003	
SSW-MOV-CC-29	HPCS SERVICE WATER PUMP DISCHARGE MOV 29 FAIL	1.0E-003	
SSW-MOV-CC-2B	STANDBY SERVICE WATER PUMP 1B DISCHARGE MOV 2	1.0E-003	
SSW-MOV-CF-12AB	CCF OF STANDBY SERVICE WATER PUMP DISCHARGE M	2.6E-005	
SSW-MOV-CF-2AB	CCF OF STANDBY SERVICE WATER PUMP DISCHARGE M	2.6E-005	
SSW-MOV-CF-2AB29	CCF OF SSW PUMP DISCHARGE MOVS (2A, 2B & 29)	2.3E-006	
SSW-STR-CF-1A1B	STANDBY SERVICE WATER STRAINERS 1A & 1B PLUG	1.4E-005	
SSW-STR-PG-1A	STANDBY SERVICE WATER SUCTION STRAINER FOR PU	9.6E-005	
SSW-XHE-XR-MDP1B	OP FAILS TO RESTORE STBY SERVICE WATER PUMP 1	1.0E-003	
SSW-XHE-XR-MDP2	OP FAILS TO RESTORE STBY SERVICE WATER PUMP H	1.0E-003	

1. Event was set to TRUE to model the condition of the actual event.

Appendix A

Event Tree Figures

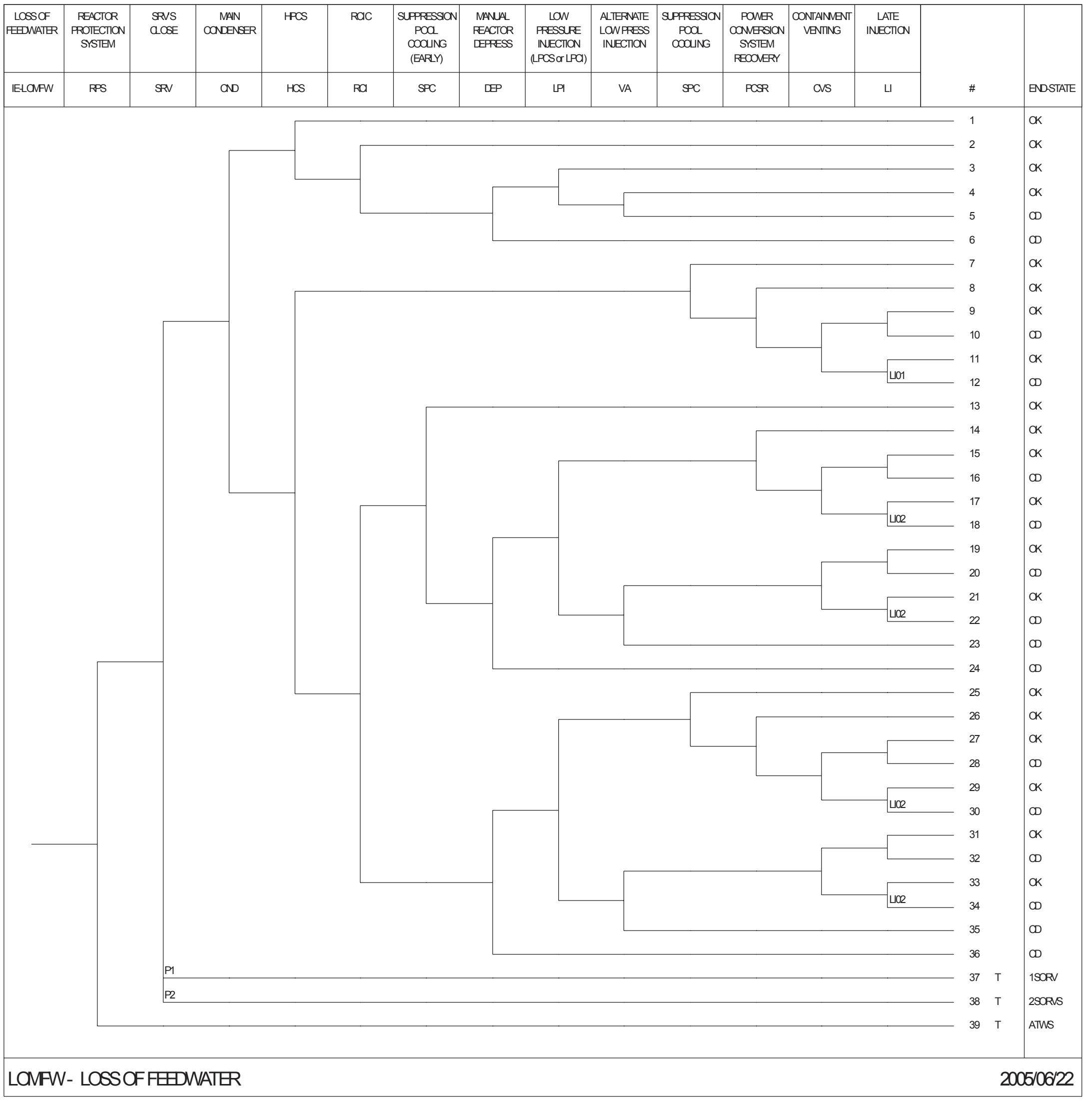

Figure 1 Loss of Feedwater Event Tree