B.6 LER No. 313/96-005

Event Description: Reactor trip and subsequent steam generator dryout

Date of Event: May 19, 1996

Plant: Arkansas Nuclear One, Unit 1

B.6.1 Summary

Arkansas Nuclear One, Unit 1 (ANO 1) was operating at 100% power when the plant experienced an automatic trip on high reactor coolant system (RCS) pressure that resulted from reduced main feedwater (MFW) flow. Following the scram, six of eight main steam safety valves (MSSVs) lifted on the B once-through steam generator (OTSG). One of these safety valves stuck open when pressure was reduced, and about 18 min after the trip, the operators, in accordance with the plant emergency operating procedures, isolated the faulted B OTSG from its MFW source and steam outlet. With the pressure and temperature decreasing on the secondary side of the OTSG, the OTSG was allowed to "dry out" because the RCS temperature was maintained relatively constant. About 5 h and 41 min after the trip, the stuck-open safety valve was gagged closed. After that, the B OTSG was refilled, and the plant was returned to normal hot shutdown conditions. The estimated conditional core damage probability (CCDP) for this event is 5.6×10^{-6}.

B.6.2 Event Description

According to the licensee event report (LER),[1] the plant was operating at 100% power when a degradation of the power supply to the turbine hydraulic control valve for MFW pump (MFWP) A caused a rapid decrease in pump speed. A second decrease in pump speed resulted in the pump going to minimum speed. The integrated control system (ICS) responded to the change in feedwater flow by increasing the speed on MFWP B. The lower heat removal rate resulting from the reduced feedwater flow caused by MFWP A going to minimum speed in turn caused the pressure in the reactor to increase. The increasing pressure then caused the reactor to automatically trip on high pressure, just before the operators attempted to manually trip the reactor. At this time, MFWP A was at minimum speed; the feedwater cross-over valve was closed (normal position) because no MFWP trip signal was present; and MFWP B was at maximum speed, in its "Diagnostic-Manual" mode, and not responding to additional ICS signals. Following the trip, A OTSG had a low water level inventory, and B OTSG had a high water level inventory. However, because of a back pressure wave induced by closing the main turbine stop valves, the sensed water level in the B OTSG indicated low, thereby actuating the emergency feedwater (EFW) system. MFWP B tripped on high discharge pressure ~14 s after the reactor scram, and MFWP A responded to ICS demand signals when its control circuit fault cleared; however, because the demand signal was very high, MFWP A tripped on mechanical overspeed about 37 s after the reactor trip.

Secondary-side steam pressure in the A OTSG remained below the MSSV setpoints because of the reduced inventory in the steam generator that resulted from the lower feedwater flow rate caused by the MFWP A speed decrease; conversely, the high inventory in the B OTSG (caused by MFWP B going to maximum

speed) resulted in a high secondary-side steam pressure. Consequently, six of the eight MSSVs on the B OTSG opened to reduce pressure. These valves opened prior to the A and B MFW pump trips. The operators noted ~64 s after the reactor trip that one of the MSSVs had failed to reclose following the pressure reduction, thus causing an accelerated RCS cooldown rate. Operators manually initiated high pressure injection (HPI) about 6 min after the reactor trip in accordance with the plant's Emergency Operating Procedures (EOPs) when the water level in the pressurizer dropped below 30 in. Following that, also in accordance with EOPs, the faulted B OTSG was isolated from its feedwater source and steam outlet about 18 min after the trip. The secondary side of the OTSG continued to "blow down" through the open MSSV; however, the operators controlled the RCS cooldown rate by maintaining the RCS temperature above 271.1°C (520°F). This blowdown is also referred to as "drying out" the OTSG, and the lack of steam in the OTSG results in the OTSG shell cooling down below the RCS temperature.[2] This tube-to-shell temperature differential is governed by plant Technical Specifications and is limited to 15.6°C (60°F) for ANO 1. During this transient, however, the shell-to-tube temperature differential increased to 23.3°C (74°F). The OTSG vendor, Framatome Technology, Inc., and the licensee both analyzed the 23.3°C (74°F) temperature difference and concluded that no excessive stresses were induced on the OTSG or the reactor pressure vessel.

With the steam header B isolated, the normal supply for sealing steam for the gland seals on the main turbine was not available, and because the backup steam supply (the auxiliary boiler) was also unavailable due to control system problems, sealing steam was eventually lost. As a result, about 35 min after the trip, the vacuum in the main condenser was lost. However, heat removal was still possible after the main condenser became unavailable by discharging steam through an atmospheric dump valve, which the operators did until the auxiliary boiler was available. At this time, the vacuum in the main condenser was reestablished, and the main condenser was again used for heat removal.

Plant maintenance crews successfully gagged closed the MSSV approximately 5 h and 41 min after the reactor trip. Using EFW, operators then began refilling B OTSG and cleared the main steam line isolation signal. The main steam isolation valve for B OTSG was opened about 2 h later and the main feedwater isolation valve about an hour after that. At that time, normal feedwater was established to the OTSG, and the plant was restored to a normal hot shutdown condition.

B.6.3 Additional Event-Related Information

A short circuit in a digital speed sensing probe for MFWP A reduced voltage in the feedwater control system 24-V power supply. This, in turn, decreased control oil pressure for the MFWP turbine steam admission valve, causing the valve to partially close. The closing of the steam admission valve decreased the speed of MFWP A, thereby decreasing feedwater flow. The reduced feedwater flow in turn caused the ICS to demand maximum MFW; however, the MFW control system incorrectly interpreted this as a failure (invalid signal) in the ICS and transferred MFWP B control to the "Diagnostic Manual" mode. This effectively kept MFWP B operating in response to the last valid sensed signal (high demand). When the reactor tripped, the ICS sent a reduction signal to the feedwater control system. However, MFWP B did not respond because it was in "Diagnostic Manual" mode. The feedwater block valves closed in response to the rapid flow reduction signal. As a result, the system pressure increased rapidly, and MFWP B tripped on overpressure.

The MFW system at ANO 1 consists of two variable-speed turbine-driven pumps that take their common suction downstream of feedwater heaters E2A and E2B and discharge to the OTSGs. Either MFWP can discharge to both OTSGs by routing through the normally closed feedwater cross-over valve located before the feedwater flow control valves.[3,4] Typically, these pumps are used to supply feedwater to the OTSGs from about 3% power to full power. The system also has an auxiliary motor-driven pump, which is used to supply feedwater to the OTSGs during plant startup and shutdown below 3% power. The auxiliary pump takes a suction from the MFWP suction header and discharges to the MFWP A discharge header upstream of the cross-over valve. The MFWPs are rated at 60% of the plant's full-load capacity each, and the auxiliary feedwater pump is rated at 5% full-load capacity.

The MSSV that did not reclose failed to reseat because the locking device cotter pin was not engaged with the release nut. This allowed the release nut to travel down the spindle of the valve and block the manual lift top lever from returning to its normal position. This phenomenon has been documented in NRC Information Notice 84-33, as well as in other industry studies. Investigations indicate that either the failure of the cotter pin or the insufficient slot engagement by the cotter pin allows the release nut to rotate down the spindle while the MSSV is lifted. The NRC Augmented Inspection Team (in Sects. 3.2 and 6.3 of Ref. 2) sent to investigate this event found that of the 16 MSSVs at ANO 1:

> . . . one stuck-open, . . . because of a stem-nut utilized to facilitate manual lifting of the valve, not being properly pinned in place so that during lift and/or blowdown of the valve the nut traveled down the stem and contacted the lifting device. This contact precluded the valve from reseating. . . . 6 of the 15 other MSSVs in Unit 1 had less than desirable cotter pin engagement . . .2 of the remaining 9 had marginal (i.e., cotter pin) engagement. (i.e., and the remaining 7 valves had acceptable cotter pin engagement) . . . despite marginal engagement, none of the nuts could be rotated by hand.

Hence, the quotes above show by allowing the release nut to rotate does not prevent the valve from reclosing after it has opened. The NRC's Augmented Inspection Team determined that the licensee's procedures for installing the cotter pins were inadequate. Moreover, the licensee's own inspection (Ref. 1, Section D) found that

> Cotter pins for two other valves were found not engaged in the release nuts. These valves were determined to have been operable since the release nuts could not be rotated due to the cotter pin ends being engaged on the nuts. Six valves had the pins partially engaged at the top end of the release nut slot. Seven valves were found with the cotter pins fully engaged.

Therefore, this was determined to be a singular incident.

B.6.4 Modeling Assumptions

This event was examined as the combination of two individual events. The first is the reactor trip and subsequent loss of main feedwater (LOFW) transient. The second is the potential for a steam generator tube rupture (SGTR) as a result of the "drying out" of B OTSG. The LOFW is a relatively simple and straightforward transient with few complications other than operator burdens in the recovery process. The potential for a SGTR, however, is neither simple nor straightforward. Both events are discussed below.

LOFW

The LOFW transient began with the reactor trip, continued through the subsequent OTSG dryout, and concluded with MFW recovery. The transient concluded when the MFW isolation valve was opened, which according to Ref. 2, Attachment 1, was approximately 9 h and 36 min after the trip. The event was modeled as a high-pressure reactor trip initiating event with subsequent LOFW (IE-TRANS and MFW-SYS-TRIP set to TRUE for this portion of the analysis). MFW was assumed recoverable; however, it should be noted that both MFWPs were tripped (one on mechanical overspeed and the other on high discharge pressure, but the second also had an undetermined failure in its control system) and were not used in lieu of EFW prior to the OTSG isolation. After that, in the long-term after the OTSG isolation, EFW was also used to supply the on-line OTSG. When the isolation was cleared, the B OTSG was refilled using EFW. MFW was not used until the OTSGs were supplied via the startup valves about 9 h and 42 min after the trip (Ref. 2, Attachment 1), and EFW was not secured until almost 10 h after the reactor trip (Ref. 2, Attachment 1). The ANO 1 model used in conjunction with the Integrated Reliability and Risk Analysis System (IRRAS)[5] already includes the motor-driven auxiliary feedwater pump as a supplement to the MFW when the MFWPs have tripped off or have failed.

SGTR

The typical accident analysis for core damage examines loss-of-coolant accidents (LOCAs), of which the small-break LOCA (SLOCA) is a subset. The SGTR, in many aspects, is similar to the SLOCA. The SGTR is examined for its resulting effect on core integrity. According to NUREG-0844 (Ref. 6):

> . . . concerns which were raised relative to steam generator tube degradation stem from the fact that the steam generator tubes are a part of the reactor coolant system (RCS) boundary and that tube failures result in a loss of primary coolant. . . .
>
> The leakage of primary coolant into the secondary has two major safety implications. The first is the potential for direct release of radioactive fission products into the environment, and the second is the loss of cooling water which is needed to prevent core damage. An extended uncontrolled loss of coolant outside containment would result in the depletion of the initial RCS inventory and emergency core cooling system (ECCS) water without the capability to recirculate the water.

The licensee and Framatome Technologies, Inc., examined this event by focusing on the differential temperature (Δt) experienced by the OTSG during the dryout, and they correlated that temperature to a pounds compressive force (Ref. 2, Section 4.2). The maximum Δt occurred approximately 2 h and 44 min after the trip (Ref. 2, Attachment 1). The stresses induced by the corresponding high Δt during the OTSG dryout were probably greater than those produced by the transient-induced differential pressure (Δp); however, if the scope of the analysis follows the increased stress due to the maximum Δt, the underlying assumption still concerns tube integrity, and the analysis will ultimately result in examination of tube rupture or leakage. The analysis will follow the SGTR after that. In the absence of a simple correlation available to reconcile the stresses induced by the temperature increase, the transient was analyzed using the Δp increase rather than the Δt increase.

About 18 min after the plant had shut down, the OTSG was isolated and allowed to blow down through the stuck-open MSSV. While the secondary-side pressure was decreasing, the primary-side (RCS) pressure was kept nearly constant.[2,4] This resulted in the OTSG tubes being exposed to an increasing Δp, which stopped increasing only when the safety valve was gagged closed. The OTSG secondary-side pressure decreased to 0.138 MPa (20 psig),[7] while the primary-side pressure was stabilized near the normal operating pressure[2,7] of 14.72 MPa (2,155 psig).[4] This means that the tubes of the B OTSG were subjected to a maximum Δp of 14.58 MPa (2,135 psid). The licensee has supplied data[8] that indicates that the probability of an SGTR is 1.6×10^{-3} for this Δp.

Following a nominal SGTR, the RCS is depressurized by the operators to below the MSSV setpoint. This allows the MSSVs to close and equalizes pressure between the primary and secondary sides of the SG, which terminates flow through the break. During this event, the high SG Δp (which could have induced the tube rupture) was the result of a stuck-open MSSV. The stuck-open MSSV would have prevented SG and RCS pressures from equalizing, and therefore prevented isolation of the ruptured SG. In this case, the operators would have had to continue cooling down and depressurizing the RCS until the unit could have been placed on the DHR system. Basic events MSS-VCF-HW-ISOL and MSS-XHE-NOREC were set to TRUE (i.e., probability of occurring = 1.0) to reflect the inability to isolate the ruptured SG following the postulated SGTR.

EVENT MODEL

The analyses for LOFW and SGTR were combined to analyze the entire event as follows:

[P(SGTR) × estimated CCDP for SGTR] + {[1 - P(SGTR)] × estimated CCDP for LOFW]}.

B.6.5 Analysis Results

The estimated CCDP for this event is 5.6×10^{-6}. This estimation was derived from the equation given in the previous section and is calculated as follows:

[P(SGTR) × estimated CCDP for SGTR] + {[1 - P(SGTR)] × estimated CCDP for LOFW]},

where

probability of tube rupture	=	0.0016
1 - probability of tube rupture	=	0.9984
CCDP due to SGTR	=	3.1×10^{-3}
CCDP due to LOFW	=	6.37×10^{-7}

The dominant core damage sequence, highlighted as sequence number 3 on Fig. B.6.1, contributes ~54% to the estimated CCDP. The dominant sequence involves the following steps:

- SGTR initiating event occurs,
- the reactor is successfully tripped,
- EFW is successful,
- HPI is successful,
- failure of SG isolation,
- successful depressurization to the DHR initiation pressure, and
- failure of DHR.

The LOFW contributes ~11% to the estimated total CCDP.

Definitions and probabilities for selected basic events are shown in Table B.6.1. The conditional probabilities and sequence logic associated with the highest probability sequences are shown in Table B.6.2. Table B.6.3 describes system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table B.6.4. Because the LOFW transient contributed about 11% to the total CCDP, information regarding the analysis was included in the above tables for clarification, better understanding of the event, and calculation of the total CCDP.

B.6.6 References

1. LER No. 313/96-005, Rev. 0, "Automatic Reactor Trip and Engineered Safety Features Actuations Caused by Failure of a Speed Sensing Probe in the Control Circuitry of a Main Feedwater Pump Turbine and Failure of a Main Steam Safety Valve to Re-Seat," June 18, 1996.

2. NRC Augmented Inspection Team Report No. 50-313, -368/96-19, June 12, 1996.

3. *ANO 1 Probabilistic Risk Assessment - Individual Plant Examination Submittal*, April 1993.

4. *ANO 1 Safety Analysis Report, Amendment 13*, September 25, 1995.

5. U.S. Nuclear Regulatory Commission, *Systems Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE), Version 5.0*, NUREG/CR-6116 (EGG-2716), Volumes 1-10, July 1994.

6. U.S. Nuclear Regulatory Commission, *NRC Integrated Program of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity*, NUREG-0844, September 1988.

7. Personal communication, P. D. O'Reilly, U.S. NRC, with T. Reis, U.S. NRC.

8. D. C. Mims, Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, July 23, 1997.

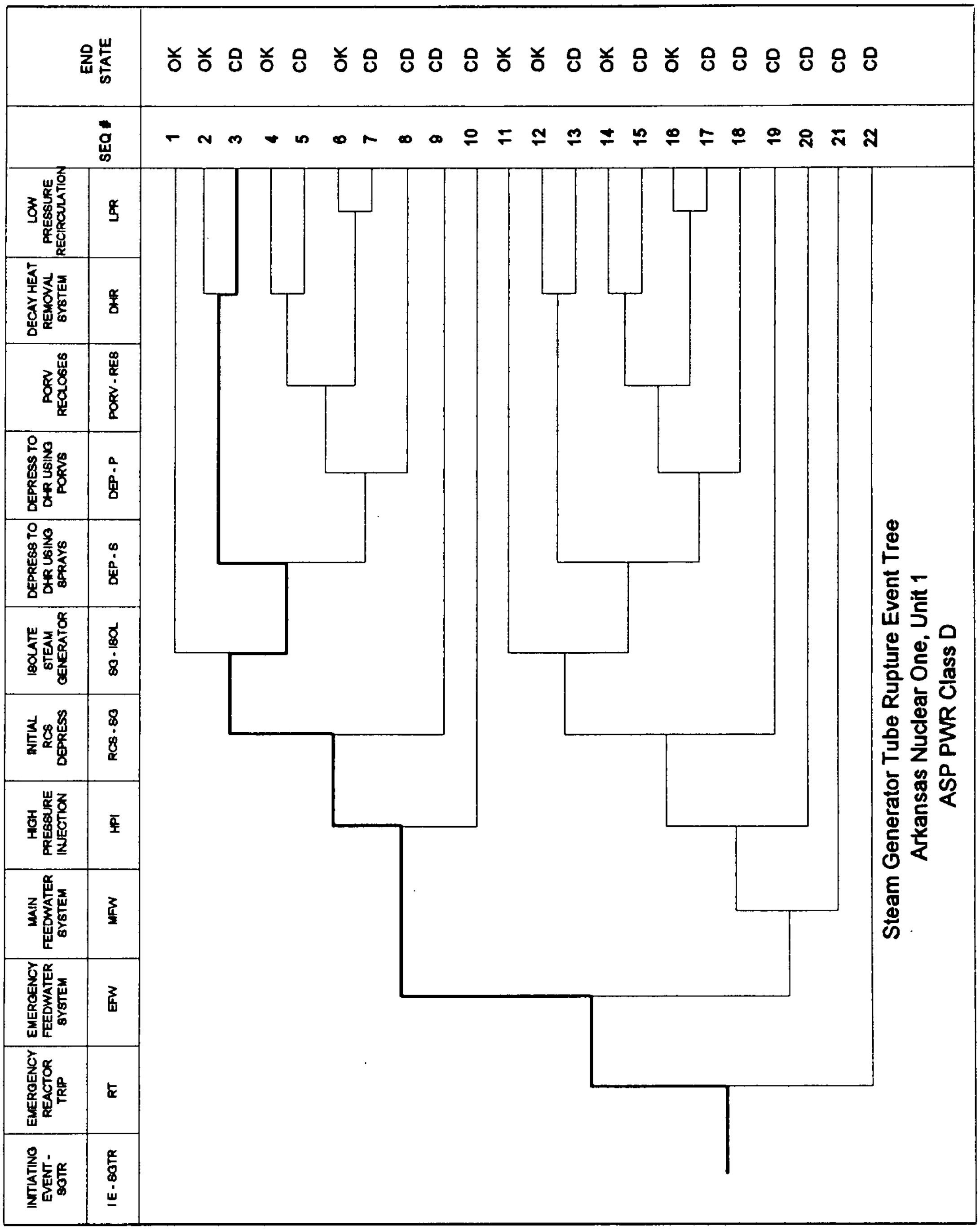

Fig. B.6.1 Dominant core damage sequence for LER No. 313/96-005.

Table B.6.1. Definitions and Probabilities for Selected Basic Events for LER No. 313/96-005

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Initiating Event–Loss of Offsite Power	5.8 E-006	5.8 E-006		No
IE-SGTR	Initiating Event–SGTR	1.6 E-006	1.0 E+000	TRUE	Yes
IE-SLOCA	Initiating Event–SLOCA	1.0 E-006	1.0 E-006		No
IE-TRANS	Initiating Event–Transient (TRANS)	1.3 E-004	1.0 E+000	TRUE	Yes
DHR-MDP-CF-ALL	Common-Cause Failures of DHR Pumps	4.5 E-004	4.5 E-004		No
DHR-MOV-CC-SUC	DHR Suction Path Failures	6.0 E-003	6.0 E-003		No
DHR-MOV-CF-DISCH	Common-Cause Failures of the Discharge MOVs	2.6 E-004	2.6 E-004		No
DHR-XHE-NOREC	Operator Fails to Recover the DHR System	1.0 E-001	1.0 E-001		No
DHR-XHE-XM-DHR	Operator Fails to Initiate the DHR System	1.0 E-003	1.0 E-003		No
EFW-MDP-FC-1A	Failure of the Emergency Feedwater (EFW) Motor-Driven Pump	3.8 E-003	3.8 E-003		No
EFW-MOV-CF-DISAL	EFW Discharge Valves Fail From Common Causes	5.5 E-005	5.5 E-005		No
EFW-TDP-FC-1B	Failure of the EFW Turbine-Driven Pump	3.2 E-002	3.2 E-002		No
EFW-TNK-FC-CST	Failure of the Condensate Storage Tank	1.0 E-004	1.0 E-004		No
EFW-XHE-NOREC	Operator Fails to Recover EFW System	2.6 E-001	2.6 E-001		No
EFW-XHE-XA-CST	Operator Fails to Align a Backup Water Supply	1.0 E-003	1.0 E-003		No

Table B.6.1. Definitions and Probabilities for Selected Basic Events for LER No. 313/96-005 (Continued)

Event name	Description	Base probability	Current probability	Type	Modified for this event
HPI-CKV-OO-MST	Makeup Storage Tank Stop Check Valve Fails to Seat	3.0 E-003	3.0 E-003		No
HPI-MDP-CF-ABC	High Pressure Injection (HPI) Motor-Driven Pumps Fail to Run due to Common Cause	1.2 E-005	1.2 E-005		No
HPI-MDP-FC-1C	HPI Train C Fails	3.9 E-003	3.9 E-003		No
HPI-MOV-CC-SUCA	Train A Suction Isolation Motor-Operated Valve Fails	3.1 E-003	3.1 E-003		No
HPI-MOV-CC-SUCC	Train C Suction Isolation Motor-Operated Valve Fails	3.1 E-003	3.1 E-003		No
HPI-MOV-CF-SUCT	HPI Suction Isolation Motor-Operated Valves Fail due to Common Cause	2.6 E-004	2.6 E-004		No
HPI-XHE-NOREC	Operator Fails to Recover HPI	8.4 E-001	8.4 E-001		No
HPI-XHE-XM-BOR	Operator Fails to Initiate Emergency Boration	1.0 E-003	1.0 E-003		No
HPI-XHE-XM-HPIC	Operator Fails to Initiate HPI Cooling	1.0 E-002	1.0 E-002		No
MFW-SYS-TRIP	Main Feedwater (MFW) System Trips	2.0 E-001	1.0 E+000	TRUE	Yes
MFW-XHE-NOREC	Operator Fails to Recover MFW	1.6 E-002	1.6 E-002		No
MSS-VCF-HW-ISOL	Ruptured SG Isolation Fails	1.0 E-002	1.0 E+000	TRUE	Yes
MSS-XHE-NOREC	Operator Fails to Isolate Ruptured SG	1.0 E-001	1.0 E+000	TRUE	Yes
PCS-ICC-FA-TT	Failure of the Main Turbine to Trip	1.0 E-003	1.0 E-003		No
PCS-PSF-HW	Hardware Failures Causing Failure to Depressurize	1.0 E-005	1.0 E-005		No
PCS-XHE-XM-SG	Operator Fails to Initiate RCS Depressurization	4.0 E-004	4.0 E-004		No

Table B.6.1. Definitions and Probabilities for Selected Basic Events for LER No. 313/96-005 (Continued)

Event name	Description	Base probability	Current probability	Type	Modified for this event
PPR-MOV-OO-BLK	Power-Operated Relief Valve (PORV) Block Valve Fails to Close	4.0 E-003	4.0 E-003		No
PPR-SRV-CC-PORV	PORV Fails to Open on Demand	6.3 E-003	6.3 E-003		No
PPR-SRV-CC-RCS	PORV/SRVs Fail to Limit Reactor Coolant System (RCS) Pressure	4.4 E-004	4.4 E-004		No
PPR-SRV-CO-TRAN	PORV Opens During Transient	8.0 E-002	8.0 E-002		No
PPR-SRV-OO-PORV	PORV Fails to Reclose After Opening	3.0 E-002	3.0 E-002		No
PPR-XHE-NOREC	Operator Fails to Close the Block Valve	1.1 E-002	1.1 E-002		No
RCS-PHN-MODPOOR	Moderator Temp Coefficient not Negative Enough	1.4 E-002	1.4 E-002		No
RCS-XHE-XM-DEPRH	Operator Fails to Depressurize the RCS to RHR Entry Conditions	1.0 E-003	1.0 E-003		No
RPS-NONREC	Nonrecoverable RPS Failures	2.0 E-005	2.0 E-005		No
RPS-REC	Recoverable RPS Failures	4.0 E-005	4.0 E-005		No
RPS-XHE-XM-SCRAM	Operator Fails to Manually Trip the Reactor	1.0 E-002	1.0 E-002		No

Table B.6.2. Sequence Conditional Probabilities for LER No. 313/96-005

Event tree name	Sequence number	Conditional core damage probability (CCDP)	Percent contribution[a]	Logic
SGTR	3	1.6 E-003	53.5	/RT, /EFW, /HPI, /RCS-SG, SGISOL, /DEP-S, DHR
	8	1.0 E-003	31.8	/RT, /EFW, /HPI, /RCS-SG, SGISOL, DEP-S, DEP-P
	9	4.1 E-004	13.0	/RT, /EFW, /HPI, RCS-SG
Subtotal (SGTR)		**3.1 E-003**		
TRANS	21-16	3.1 E-007	49.4	RT, MFW-A, /EFW-ATWS, RCSPRESS
	21-17	2.0 E-007	31.4	RT, MFW-A, EFW-ATWS
	20	8.9 E-008	13.9	/RT, EFW, MFW, HPI-COOL
	21-15	2.0 E-008	3.2	RT, MFW-A, /EFW-ATWS, /RCSPRESS, BORATION
	8	9.9 E-009	1.5	/RT, /EFW, PORV, PORV-RES, HPI
Subtotal (TRANS)		**6.3 E-007**		

[a]Percent contribution to the subtotal CCDP.

Table B.6.3. System Names for LER No. 313/96-005

System name	Description
BORATION	Emergency Boration Fails
DEP-P	Failure to Depressurize the RCS to DHR Using PORVs
DEP-S	Failure to Depressurize the RCS to DHR Using Spray
DHR	No or Insufficient Flow From the DHR System
EFW	No or Insufficient Emergency Feedwater (EFW) System Flow
EFW-ATWS	No or Insufficient EFW System Flow
HPI	No or Insufficient Flow From the High Pressure Injection (HPI) System
HPI-COOL	Failure to Provide HPI Cooling
MFW	Failure of the Main Feedwater System
MFW-A	Failure of the Main Feedwater System During ATWS
PORV	PORV Opens During Transient
PORV-RES	PORV Fails to Reseat
RCS-SG	Failure to Initiate RCS Depressurization
RCSPRESS	Failure to Limit RCS Pressure
RT	Reactor Fails to Trip During Transient
SGISOL	Failure to Isolate SG

Table B.6.4. Conditional Cut Sets for Higher Probability Sequences for LER No. 313/96-005

Cut set number	Percent contribution	Conditional probability[a]	Cut sets[b]
SGTR Sequence 03		1.6 E-003	
1	59.4	1.0 E-003	MSS-VCF-HW-ISOL, MSS-XHE-NOREC, DHR-XHE-XM-DHR
2	35.6	6.0 E-004	MSS-VCF-HW-ISOL, MSS-XHE-NOREC, DHR-MOV-CC-SUC, DHR-XHE-NOREC
3	2.6	4.5 E-005	MSS-VCF-HW-ISOL, MSS-XHE-NOREC, DHR-MDP-CF-ALL, DHR-XHE-NOREC
4	1.5	2.6 E-005	MSS-VCF-HW-ISOL, MSS-XHE-NOREC, DHR-MOV-CF-DISCH, DHR-XHE-NOREC
SGTR Sequence 8		1.0 E-003	
1	100.0	1.0 E-003	MSS-VCF-HW-ISOL, MSS-XHE-NOREC, RCS-XHE-XM-DEPRH
SGTR Sequence 9		4.1 E-004	
1	97.5	4.0 E-004	PCS-XHE-XM-SG
2	2.4	1.0 E-005	PCS-PSF-HW
TRANS Sequence 21-16		3.1 E-007	
1	88.9	2.8 E-007	RPS-NONREC, MFW-SYS-TRIP, RCS-PHN-MODPOOR
2	6.3	2.0 E-008	RPS-NONREC, MFW-SYS-TRIP, PCS-ICC-FA-TT
3	2.7	8.8 E-009	RPS-NONREC, MFW-SYS-TRIP, PPR-SRV-CC-RCS
4	1.7	5.6 E-009	RPS-REC, MFW-SYS-TRIP, RPS-XHE-XM-SCRAM, RPS-PHN-MODPOOR
TRANS Sequence 21-17		2.0 E-007	
1	82.9	1.7 E-007	RPS-NONREC, MFW-SYS-TRIP, EFW-TDP-FC-1B, EFW-XHE-NOREC

Table B.6.4. Conditional Cut Sets for Higher Probability Sequences for LER No. 313/96-005 (Continued)

Cut set number	Percent contribution	Conditional probability[a]	Cut sets[b]
2	10.0	2.0 E-008	RPS-NONREC, MFW-SYS-TRIP, EFW-MDP-FC-1A, EFW-XHE-NOREC
3	2.5	5.2 E-009	RPS-NONREC, MFW-SYS-TRIP, EFW-XHE-XA-CST, EFW-XHE-NOREC
4	1.6	3.3 E-009	RPS-REC, RPS-XHE-XM-SCRAM, MFW-SYS-TRIP, EFW-TDP-FC-1B, EFW-XHE-NOREC
TRANS Sequence 20		8.9 E-008	
1	46.7	4.2 E-008	EFW-XHE-XA-CST, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-HPIC
2	29.4	2.6 E-008	EFW-XHE-XA-CST, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-PORV
3	5.8	5.1 E-009	EFW-MDP-FC-1A, EFW-TDP-FC-1B, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-HPIC
4	4.6	4.1 E-009	EFW-TNK-FC-CST, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-HPIC
5	3.6	3.2 E-009	EFW-MDP-FC-1A, EFW-TDP-FC-1B, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-PORV
6	2.9	2.6 E-009	EFW-TNK-FC-CST, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-PORV
7	2.5	2.3 E-009	EFW-MOV-CF-DISAL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-HPIC
8	1.6	1.4 E-009	EFW-MOV-CF-DISAL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-PORV
9	1.0	9.1 E-010	EFW-XHE-XA-CST, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-MOV-CF-SUCT, HPI-XHE-NOREC
TRANS Sequence 21-15		2.0 E-008	
1	98.0	2.0 E-008	RPS-NONREC, MFW-SYS-TRIP, HPI-XHE-XM-BOR
2	1.9	4.0 E-010	RPS-REC, RPS-XHE-XM-SCRAM, MFW-SYS-TRIP, HPI-XHE-XM-BOR

Table B.6.4. Conditional Cut Sets for Higher Probability Sequences for LER No. 313/96-005 (Continued)

Cut set number	Percent contribution	Conditional probability[a]	Cut sets[b]
TRANS Sequence 8		9.9 E-009	
1	58.8	5.8 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-PORV, PPR-XHE-NOREC, HPI-MOV-CF-SUCT, HPI-XHE-NOREC
2	21.4	2.1 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-PORV, PPR-MOV-OO-BLK, HPI-MOV-CF-SUCT, HPI-XHE-NOREC
3	2.7	2.7 E-010	PPR-SRV-CO-TRAN, PPR-SRV-OO-PORV, PPR-XHE-NOREC, HPI-MDP-CF-ABC, HPI-XHE-NOREC
4	2.7	2.7 E-010	PPR-SRV-CO-TRAN, PPR-SRV-OO-PORV, PPR-XHE-NOREC, HPI-MOV-CC-SUCA, HPI-MD-FC-1C, HPI-XHE-NOREC
5	2.6	2.6 E-010	PPR-SRV-CO-TRAN, PPR-SRV-OO-PORV, PPR-XHE-NOREC, HPI-CKV-OO-MST, HPI-MDP-FC-1C, HPI-XHE-NOREC
6	2.1	2.1 E-010	PPR-SRV-CO-TRAN, PPR-SRV-OO-PORV, PPR-XHE-NOREC, HPI-MOV-CC-SUCA, HPI-MOV-CC-SUCC, HPI-XHE-NOREC
7	2.0	2.1 E-010	PPR-SRV-CO-TRAN, PPR-SRV-OO-PORV, PPR-XHE-NOREC, HPI-CKV-OO-MST, HPI-MOV-CC-SUCC, HPI-XHE-NOREC
Subtotal SGTR		3.1 E-003	
Subtotal TRANS		6.3 E-007	
Total		**3.1 E-003**	

[a]The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cutset. The probabilities of the initiating and basic events are given in Table B.6.1. Initiating events begin with the designator IE.

[b]Basic events MFW-SYS-TRIP, MSS-VCF-HW-ISOL, and MSS-XHE-NOREC are "TRUE" type events, which would not normally be included in the output of a fault tree analysis and reduction program; however, these events have been added to this table to help understand the sequences for potential core damage associated with LER No. 313/96-005.