B.2 LER No. 270/97-001

Event Description: Unisolable reactor coolant system leak

Date of Event: April 21, 1997

Plant: Oconee 2

B.2.1 Event Summary

An unisolable 45.4 L/min (12-gal/min) leak developed in the reactor coolant system (RCS), high-pressure injection (HPI) nozzle, safe end-to-piping weld downstream of reactor coolant pump (RCP) 2A1 (Ref. 1). Unit 2 was shut down, and personnel removed and inspected the leaking pipe section. The leak was caused by a circumferential crack with through-wall penetration along 1.3 rad (77°) of the outer pipe surface. In addition, the nozzle thermal sleeve was loose and cracked, with portions missing from the end that extends into the RCS flow path. The piping failures were caused by high-cycle thermal fatigue that resulted from the mixing of makeup, warming, and RCS flows. The estimated conditional core damage probability (CCDP) associated with this event is 2.2×10^{-5}.

B.2.2 Event Description

At approximately 2245 on April 21, 1997, with Unit 2 at 100% power, changes were noted in the rate at which the water level in the letdown storage tank (LDST) was decreasing while the reactor building (RB) sump level was increasing. RB radiation monitor alarms followed. At 2300 the RCS leak rate was estimated to be 8.9 L/min (2.36 gal/min). Personnel entered the RB at 0215, determined that a leak did exist, but could not identify the source.

The shutdown of Unit 2 began at 0352, with the intention to reduce power to 15%. Because reducing the power level reduces the radiation levels in the RB, personnel could then perform a more detailed inspection of the leak area with the main turbine remaining on-line. At 0900, a more accurate leak rate calculation was performed with power stabilized at 20%. This calculation indicated that the leak rate had increased from 8.9 L/min (2.36 gal/min) at 2300 to 23.6 L/min (6.25 gal/min) at 0940. By 1048 it had increased above 30.3 L/min (8 gal/min).

At 1217, during another RB entry, the leak location was identified as being in the vicinity of 2HP-127, the block valve closest to the injection nozzle on the 2A1 RCP cold leg (Fig. B.2.1). The decision was made to proceed to cold shutdown. The turbine generator was taken off-line at 1250, and the reactor was tripped at 1448 on April 22, 1997. The leak rate peaked at approximately 45.4 L/min (12 gal/min) at 1750 and then began to decrease as RCS pressure was reduced as the shutdown continued.

The leak was found to be in an unisolable section of piping at the weld between the HPI piping and injection nozzle safe end. The unit was placed in a reduced inventory condition, and the pipe from the safe end to the block valve was cut out for examination (a temporary cap was then welded to the safe end, and the RCS water

level was raised). This examination determined that the leaking weld was caused by a 6.3 rad (360°) inside circumferential crack that penetrated, at a minimum, 24% of the pipe wall. The flaw depth increased and became through-wall over 1.3 rad (77°) of the outer circumference, as shown in Fig. B.2.2. The nozzle thermal sleeve was also found to be loose and cracked, with portions missing from the end that extends into the RCS flow. Cracking (~20% through-wall) was also found in the pipe in the vicinity of the warming line nozzle. Video examination, ultrasonic testing (UT), and radiographic testing (RT) of the welds and thermal sleeves in the other HPI nozzles showed no indications of cracking, loosening, or other signs of degradation.

The licensee concluded that the piping failures were caused by high-cycle thermal fatigue, as a result of thermal mixing of the warming line, makeup flow, and RCS flow. Thermal mixing occurred in the thermal sleeve, safe end, and piping because of varying operational conditions, including low makeup flow through the thermal sleeve. This caused cracking in the pipe, pipe-to-safe end weld, and safe end and contributed to the thermal sleeve failure. Vibration may have also contributed to the final failure once the crack was essentially through the pipe wall.

Following several earlier industry events involving cracked thermal sleeves and nozzle safe ends (described in the following section), Oconee adopted an augmented inspection plan to periodically check piping near the pipe-to-safe end welds (on Units 2 and 3) and the thermal sleeves (on all units). A review of the inspection schedules indicated that these inspections had been performed on Unit 2 in May 1996. However, the licensee determined that the inspection program failed to include UT of the piping near the pipe-to-safe end weld. Because of this, the weld that was cracked and leaking had not been inspected since 1982. The criteria for reviewing safe end radiographs were also poorly defined.

A reassessment of all radiographs of the thermal sleeves performed since 1983 determined that there had been no RT on the thermal sleeves at Unit 1 since 1989. However, a review of radiographs taken between 1983 and 1989 indicated no degradation of any of the thermal sleeves at Unit 1. Unit 1 was shut down on June 14, 1997, and its HPI nozzles and thermal sleeves were examined. The Oconee 1 thermal sleeves are of a different design, utilizing two concentric sleeves instead of the single sleeves used in the nozzles at Units 2 and 3. No unacceptable indications were found in the Unit 1 nozzles and sleeves.

Because the reassessment of Unit 3 radiographs indicated that the 3A1 thermal sleeve was potentially degraded, Unit 3 was shut down for inspection on May 1, 1997. Cladding cracks were found in the 3A1 thermal sleeve. UT of the other Unit 3 nozzles found no rejectable indications. Both the 2A1 and 3A1 nozzles were restored by installation of new safe ends, thermal sleeves, and associated piping.

B.2.3 Additional Event-Related Information

At Oconee, the HPI system provides both normal RCS makeup and RCP seal injection, as well as HPI for small-break loss-of-coolant accident (SLOCA) mitigation. During normal operation, the HPI system A header supplies makeup [typically 57–76 L/min (15–20 gal/min)] from the LDST through each of two lines to the RCS. These lines are equipped with "warming" lines that provide a minimum flow of 11.4 L/min (3 gal/min). The B HPI header is for emergency injection only and has no warming lines.

The injection lines terminate at injection nozzle assemblies located on each of the cold legs downstream of the RCPs. Each nozzle assembly (Fig. B.2.3) consists of an Inconel-clad carbon steel nozzle to which a stainless steel safe end is welded. The HPI piping is welded to the other end of the safe end. Inside the safe end is a stainless steel thermal sleeve, which extends into the RCS flow path. The function of the thermal sleeve is to minimize thermal shock and stresses on the nozzle by transporting the relatively cold HPI water [38 to 49°C (100 to 120°F)] into the main flow path. There it mixes with the 291°C (555°F) RCS cold-leg water. Without the thermal sleeve, the HPI water would directly contact the nozzle, resulting in unacceptable stresses in the nozzle material.

Additional information concerning this event is provided in NRC Information Notice 97-46 (Ref. 2). Problems similar to this event occurred in 1982 at Crystal River 3 and Oconee and in 1988 at Farley and Davis Besse. These events are described in NRC Information Notice 82-09 (Ref. 3), Generic Letter 85-20 (Ref. 4), and NRC Bulletin 88-08 (Ref. 5). Generic Letter 85-20 adopted recommended corrective actions developed by the Babcock and Wilcox (B&W) Owner's Group following the 1982 problems at Crystal River 3 and Oconee.

B.2.4 Modeling Assumptions

This event was modeled as a potential SLOCA at the 2A1 cold-leg HPI nozzle. In the actual event the pipe crack developed slowly and began to leak. This leakage was detected, and the plant was shut down while the injection line remained substantially intact. It is possible, however, that the crack could have developed differently, resulting in catastrophic failure of the injection line before detection.

The probability of such a "rupture before leak", which would result in a LOCA, was developed using service-based piping reliability data developed by the Swedish Nuclear Power Inspectorate (SKI).[6] The probability of pipe rupture represents the likelihood that a defect could have progressed to a rupture. The conditional probability of an HPI line rupture was estimated using data related to thermal-fatigue-induced piping failures included in the recently developed SKI piping failure database.[6,7] The SKI database currently includes over 2300 pipe failure records that represent about 4300 reactor-years of operating experience. For failures due to thermal fatigue, 20 cracks and leaks, but no ruptures, were observed in stainless steel piping 2.5 to 10 cm (1 to 4 in.) in diameter. Using Bayesian statistics with a noninformative prior[a], a conditional probability of rupture of 2.4×10^{-2} was estimated.[b] Because no ruptures have been observed, this estimate may be conservative. However, several thermal fatigue-induced failures also included cyclic fatigue (vibration-induced fatigue) as a contributing factor (as noted in the **Event Description**, this may have been the case in

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[a]The use of a noninformative prior is described on page 5-36 of the *PRA Procedures Guide*, NUREG/CR-2300, January 1983. A number of alternate estimators have been proposed for the case where no failures have been observed. See, for example, Section 5.5 of NUREG/CR-2300 and R. T. Bailey's article "Estimation from Zero-Failure Data" in *Risk Analysis*, Vol. 17, No. 3, June 1997.

[b]An alternative to the "data-driven" model that constitutes the SKI effort is the application of probabilistic fracture mechanics models. These models enable the calculation of failure probabilities assuming that piping is susceptible to anticipated degradation mechanisms especially those that develop over a long period. Reference 6 notes that under a similar set of boundary conditions, the two approaches tend to produce similar (i.e., the same order of magnitude) results.

this event as well). Among the 78 failures, 2 cyclic fatigue-related ruptures have been observed. This results in an estimated conditional probability of 3.2×10^{-2}, approximately the same as the 2.4×10^{-2} estimate for thermally induced fatigue. These values are consistent with the average number of piping failures that are ruptures as estimated in 1981 by Thomas (Ref. 8)[a] and are about a factor of 4 smaller than the leak-before-break probability developed by the Electric Power Research Institute (EPRI) in 1992 (Ref. 9).[b]

The strength of the HPI line piping and the proximity of the 2.5-cm (1-in.) warming line to the leaking weld would be expected to limit pipe movement and hence flow area, if a rupture had occurred (flow would also be limited by the thermal sleeve). This was reflected in the analysis by assuming that the potential break would be a SLOCA instead of the medium-break LOCA normally associated with a 6.4-cm (2.5-in.) break at Oconee.

Flow lost from an HPI line break is unavailable for RCS makeup. Orifices in each injection line provide for flow indication to allow the operators to redirect HPI flow between the two sets of injection lines so that a majority of the flow goes through the intact header into the RCS. The HPI system design also includes cross-connects to allow flow from the center HPI pump to be directed to the intact injection lines if the pump that normally supplies these lines (pump A in the case of a break in the 2A1 injection line) is unavailable. To address a potential HPI line break, the HPI and piggy-back cooling (high-pressure recirculation) fault trees were revised to require flow through the intact header in the event of such a break (a break in header A was modeled). In addition, the potential for the operators to realign pump B to inject through the B header was also added to the model.

The model was also revised to address use of rapid RCS depressurization and low-pressure injection (LPI) in the event that HPI were to fail. The Oconee *Individual Plant Examination* (IPE)[10] states that the emergency operating procedures direct the operators to use secondary heat removal to depressurize the RCS until LPI flow is greater than 380 L/min (100 gal/min) per header. The probability of the operators failing to depressurize the RCS and initiating LPI was assumed to be 0.1, consistent with Ref. 10.

B.2.5 Analysis Results

The CCDP for a postulated SLOCA associated with the leaking 2A1 HPI nozzle weld is estimated to be 2.2×10^{-5}. The dominant sequence, sequence 3 in Fig. B.2.4, involves

- a postulated HPI line break (SLOCA) given the weld leak,
- successful reactor trip and secondary-side cooling,
- successful RCS depressurization to the decay heat removal (DHR) initiation pressure, and
- failure of DHR and piggy-back cooling.

[a]Reference 8 estimated that between 2 and 45% of piping failures were catastrophic, depending on the failure cause. On average, approximately 6% of all failures were estimated to be catastrophic. Unfortunately, piping failures caused by high-cycle fatigue were not separately enumerated. Three percent of low-cycle fatigue failures were estimated to be catastrophic, compared to 20% of vibration-related fatigue failures and 20% of failures associated with "thermal shock."

[b]Reference 9 estimated that the probability of break before leak varied from 0.09 to 0.11, depending on pipe size.

The dominant cut sets involve common-cause failures of the DHR heat exchangers and pumps.

Substantial uncertainty is associated with the CCDP estimated for this event, primarily because of uncertainty in the conditional probability of pipe rupture. In addition to the uncertainty related to zero-event data described in **Modeling Assumptions**, Ref. 6 describes, among others, the following sources of uncertainty: coverage and completeness of the SKI data collection effort, data aggregation and exposure time estimation issues, identification of appropriate reliability attributes (e.g., pipe diameter, piping material) and influence factors (such as design and operating practices), plant-to-plant differences, and in-plant differences. In one probabilistic fracture mechanics study[a] cited in Ref. 6, a 3 orders of magnitude difference existed in the conditional rupture probability for leaking 100 to 800 mm pipe ($10^{-4} \leq p \leq 10^{-1}$), depending on (1) the material, (2) whether the crack was in the base metal, or (3) if the crack was in a weld (as it was for this event). For stainless steel, the conditional probability for weld cracks was about 2 orders of magnitude higher than for cracks in base metal.

Definitions and probabilities for selected basic events are shown in Table B.2.1. The conditional probabilities associated with the highest probability sequences are shown in Table B.2.2. Table B.2.3 lists the sequence logic associated with the sequences listed in Table B.2.2. Table B.2.4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table B.2.5.

During the Unit 3 shutdown to inspect its HPI nozzles and thermal sleeves, two of its three HPI pumps were damaged when they were operated with inadequate net positive suction head (NPSH). This resulted from a drained reference leg in the LDST instrumentation. The impact of the HPI pump failures as well as the potential for a combined RCS leak and HPI pump failure are addressed in the analysis of LER 287/97-003.

B.2.6 References

1. Licensee Event Report 270/97-001, "Unisolable Reactor Coolant Leak due to Inadequate Surveillance Program," May 21, 1997.

2. NRC Information Notice 97-46, "Unisolable Crack in High-Pressure Injection Piping," July 9, 1997.

3. NRC Information Notice 82-09, "Cracking in Piping to Makeup Coolant Lines at B&W Plants," March 31, 1982.

4. NRC Generic Letter 85-20, "Resolution of Generic Issue 69: High Pressure Injection/Makeup Nozzle Cracking in Babcock and Wilcox Plants," November 11, 1985.

5. NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," June 22, 1988 (and supplements 1–3 dated June 24, 1988, August 4, 1988, and April 11, 1989, respectively).

[a] *Probabilistic Pipe Fracture Evaluations for Leak-Rate Detection Applications*, NUREG/CR-6004, 1995.

6. R. Nyman, D. Hegedus, B. Tomic, and B. Lydell, *Reliability of Piping System Components, Framework for Estimating Failure Parameters from Service Data*, SKI Report 97:26, December 1997.

7. Personal communication, B. Lydell (RSA Technologies) and J. Minarick (SAIC), September 25, 1997.

8. H. M. Thomas, "Pipe and Vessel Failure Probability," *Reliability Engineering*, 2:83 (1981).

9. *Pipe Failures in U.S. Commercial Nuclear Power Plants*, EPRI TR-100380, July 1992.

10. Duke Power Company, *Oconee Nuclear Station IPE Submittal Report*, December 1990, p. 5.7-22, Rev 1.

Figure removed during SUNSI review.

Fig. B.2.1. Flow diagram of the emergency core cooling system at Oconee 2 (*Source: Oconee 2 Final Safety Analysis Report*).

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Fig. B.2.2. Through-wall flaw depth in pipe at Oconee 2 (*Source*: NRC Information Notice 97-46, "Unisolable Crack in High-Pressure Injection Piping," July 9, 1997). (TW is through-wall, OD is outside diameter, M/U flow is makeup flow, and RCS is reactor coolant system.)

Figure removed during SUNSI review.

Fig. B.2.3. Thermal sleeve 2A1, 2B1, 3A1, 3B1, and 3B2 (*Source*: NRC Information Notice 97-46, "Unisolable Crack in High-Pressure Injection Piping," July 9, 1997). (S.S. is stainless steel, C.S. is carbon steel, and SW is socket weld.)

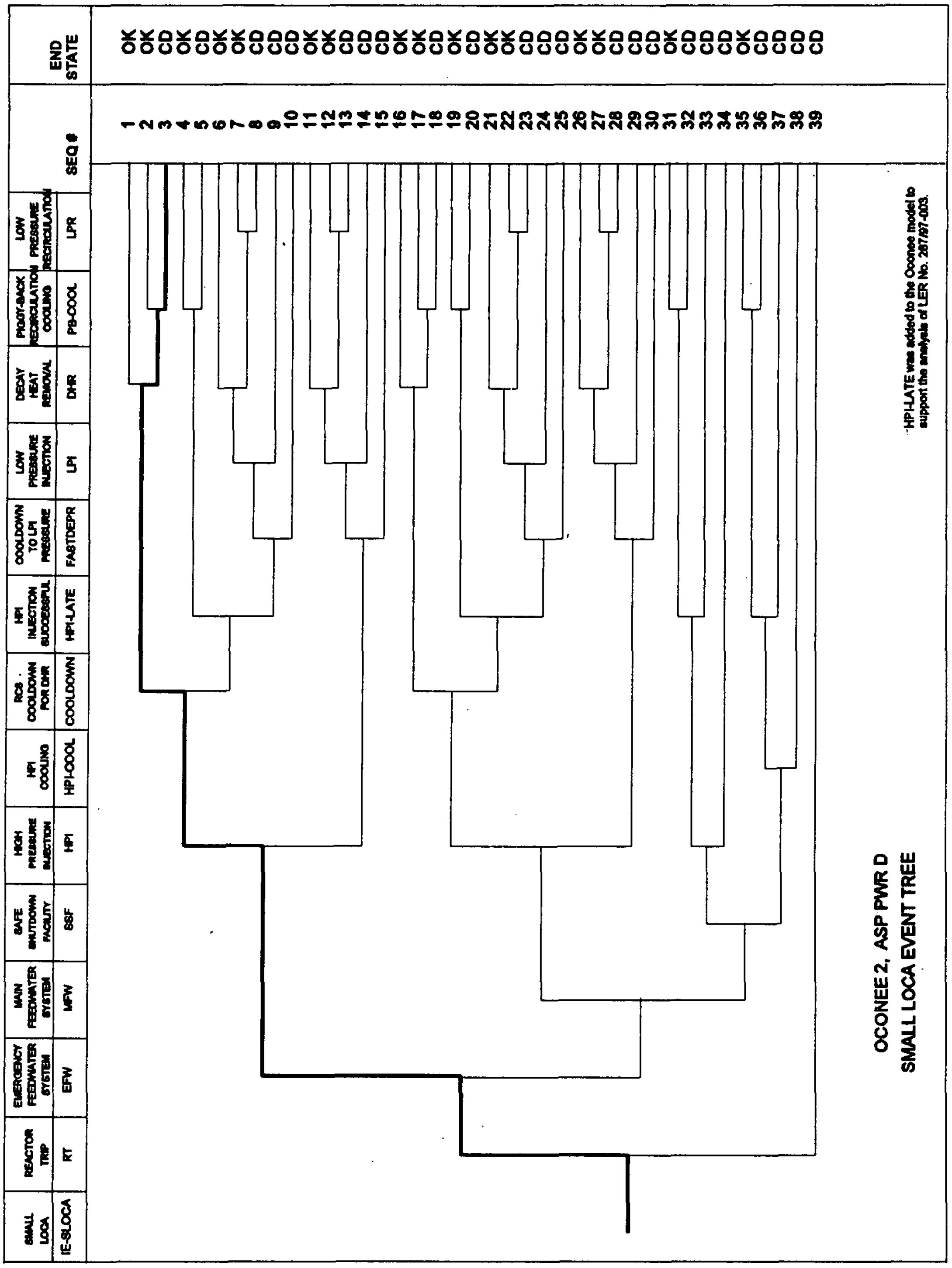

Fig. B.2.4. Dominant core damage sequence for LER No. 270/97-001.

Table B.2.1. Definitions and Probabilities for Selected Basic Events for LER No. 270/97-001

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Initiating Event–Loss of Offsite Power	3.8 E-006	0.0 E+000		Yes
IE-MLOCA	Initiating Event–Medium Loss-of-Coolant Accident	8.2 E-008	0.0 E+000		Yes
IE-SGTR	Initiating Event–Steam Generator Tube Rupture	1.3 E-006	0.0 E+000		Yes
IE-SLOCA	Initiating Event–SLOCA	6.5 E-007	2.4 E-002		Yes
IE-TRANS	Initiating Event–Transient	7.8 E-004	0.0 E+000		Yes
DHR-HTX-CF-ALL	Common-Cause Failure (CCF) of the DHR Heat Exchangers	5.2 E-004	5.2 E-004		No
DHR-MDP-CF-ALL	CCF of all Motor-Driven DHR Pumps	2.1 E-004	2.1 E-004		No
DHR-MOV-CC-SUCA	DHR Suction Motor-Operated Valves (MOVs) LP-1 or LP-2 Fail	6.0 E-003	6.0 E-003		No
HPI-LINE-BREAK	Line Break in HPI Loop A	0.0 E+000	1.0 E+000	TRUE	Yes
HPI-MDP-CF-ABC	CCF (to Run) of the Motor-Driven HPI Pumps	6.1 E-006	6.1 E-006		No
HPI-MDP-CF-START	CCF (to Start) of the Motor-Driven HPI Pumps B and C	1.9 E-004	1.9 E-004		No
HPI-MDP-FC-B	HPI Train B Fails	3.9 E-003	3.9 E-003		No
HPI-MDP-FC-C	HPI Train C Fails	3.9 E-003	3.9 E-003		No
HPI-MOV-CC-409	HPI MOV HP409 Fails to Open	3.0 E-003	3.0 E-003		No
HPI-MOV-CC-SUCA	Isolation Valve in HPI Water Supply Path A Fails	4.2 E-003	4.2 E-003		No
HPI-MOV-CC-SUCB	Isolation Valve in HPI Water Supply Path B Fails	4.2 E-003	4.2 E-003		No
HPI-MOV-CF-SUCT	CCF of HPI Suction Isolation MOVs	2.1 E-004	2.1 E-004		No

Table B.2.1. Definitions and Probabilities for Selected Basic Events for LER No. 270/97-001 (Continued)

Event name	Description	Base probability	Current probability	Type	Modified for this event
HPI-MOV-OO-115	MOV HP115 Fails to Close	3.0 E-003	3.0 E-003		No
HPR-MOV-CF-BWST	CCF of Isolation MOVs for the Borated Water Storage Tank (BWST)	7.7 E-005	7.7 E-005		No
LDST-LVL-LOW	Low Water Level in the LDST Fails the HPI Pumps	4.4 E-003	4.4 E-003		No
LPR-MOV-CF-BWST	CCF of BWST Isolation MOVs	8.6 E-005	8.6 E-005		No
LPR-SMP-FC-SUMP	Failures in the RB Sump	5.0 E-005	5.0 E-005		No
PBC-XHE-XM	Operator Fails to Initiate Piggy-Back Cooling	2.2 E-003	2.2 E-003		No
PCS-VCF-HW	Hardware Failures in the Secondary Systems	3.0 E-003	3.0 E-003		No
PCS-XHE-XM-CDOWN	Operator Fails to Initiate Cooldown	1.0 E-002	1.0 E-002		No
PCS-XHE-XM-FDEPR	Operator Fails to Initiate a Fast Depressurization for LPI	1.0 E-001	1.0 E-001		No

Table B.2.2. Sequence Conditional Probabilities for LER 270/97-001

Event tree name	Sequence number	Conditional core damage probability (CCDP)	Percent contribution
SLOCA	03	1.8 E-005	85.6
SLOCA	15	1.2 E-006	5.5
SLOCA	05	1.1 E-006	5.2
SLOCA	10	4.2 E-007	2.0
Total (all sequences)		**2.2 E-005**	

Table B.2.3. Sequence Logic for Dominant Sequences for LER 270/97-001

Event tree name	Sequence number	Logic
SLOCA	03	/RT, /EFW, /HPI, /COOLDOWN, DHR, PB-COOL
SLOCA	15	/RT, /EFW, HPI, FASTDEPR
SLOCA	05	/RT, /EFW, /HPI, COOLDOWN, /HPI-LATE, PB-COOL
SLOCA	10	/RT, /EFW, /HPI, COOLDOWN, HPI-LATE, FASTDEPR

Table B.2.4. System names for LER 270/97-001

System name	Logic
COOLDOWN	RCS Cooldown to DHR Pressure Using Turbine Bypass Valves, etc.
DHR	No or Insufficient Flow from the DHR System
EFW	No or Insufficient Flow from the Emergency Feedwater System
FASTDEPR	RCS Rapid Cooldown/Depressurization to LPI Pressure Using Turbine Bypass Valves, etc. (HPI Failed)
HPI	No or Insufficient Flow from the HPI System
HPI-LATE	HPI Fails Late
PB-COOL	No or Insufficient Flow from Piggy-Back Cooling
RT	Reactor Fails to Trip During a Transient

Table B.2.5. Conditional Cut Sets for Higher Probability Sequences for LER No. 270/97-001

Cut set number	Percent contribution	CCDP[a]	Cut sets[b]
SLOCA Sequence 03		**1.8 E-005**	
1	67.8	1.3 E-005	DHR-HTX-CF-ALL[c]
2	27.4	5.0 E-006	DHR-MDP-CF-ALL
3	1.7	3.2 E-007	DHR-MOV-CC-SUCA, PBC-XHE-XM
SLOCA Sequence 15		**1.2 E-006**	
1	42.9	5.0 E-007	HPI-MOV-CF-SUCT, PCS-XHE-XM-FDEPR
2	38.8	4.6 E-007	HPI-LINE-BREAK, HPI-MDP-CF-START[d], PCS-XHE-XM-FDEPR
3	3.6	4.3 E-008	HPI-MOV-CC-SUCA, HPI-MOV-CC-SUCB, PCS-XHE-XM-FDEPR
4	3.1	3.7 E-008	HPI-LINE-BREAK, HPI-MDP-FC-B, HPI-MDP-FC-C, PCS-XHE-XM-FDEPR
5	2.4	2.8 E-008	HPI-LINE-BREAK, HPI-MOV-CC-409, HPI-MDP-FC-C, PCS-XHE-XM-FDEPR
6	2.4	2.8 E-008	HPI-LINE-BREAK, HPI-MOV-OO-115, HPI-MDP-FC-C, PCS-XHE-XM-FDEPR
7	1.3	1.5 E-008	HPI-MOV-CF-SUCT, PCS-VCF-HW
8	1.3	1.5 E-008	HPI-MDP-CF-ABC, PCS-XHE-XM-FDEPR
9	1.2	1.4 E-008	HPI-LINE-BREAK, HPI-MDP-CF-START, PCS-VCF-HW
SLOCA Sequence 05		**1.1 E-006**	
1	47.4	5.3 E-007	PCS-XHE-XM-CDOWN, PBC-XHE-XM
2	14.2	1.6 E-007	PCS-VCF-HW, PBC-XHE-XM
3	11.2	1.3 E-007	PCS-XHE-XM-CDOWN, DHR-HTX-CF-ALL
4	4.5	5.0 E-008	PCS-XHE-XM-CDOWN, DHR-MDP-CF-ALL
5	3.4	3.7 E-008	PCS-VCF-HW, DHR-HTX-CF-ALL
6	1.9	2.1 E-008	PCS-XHE-XM-CDOWN, LPR-MOV-CF-BWST

Table B.2.5 Conditional Cut Sets for Higher Probability Sequences for LER No. 270/97-001 (Continued)

Cut set number	Percent contribution	CCDP[a]	Cut sets[b]
7	1.7	1.9 E-008	PCS-XHE-XM-CDOWN, HPR-MOV-CF-BWST
8	1.4	1.5 E-008	PCS-VCF-HW, DHR-MDP-CF-ALL
9	1.1	1.2 E-008	PCS-XHE-XM-CDOWN, LPR-SMP-FC-SUMP
SLOCA Sequence 10		**4.2 E-007**	
1	75.0	3.2 E-007	PCS-VCF-HW, LDST-LVL-LOW
2	25.0	1.1 E-007	PCS-XHE-XM-CDOWN, LDST-LVL-LOW, PCS-XHE-XM-FDEPR
Total (all sequences)		**2.2 E-005**	

[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 cut set. The probability of the initiating events are given in Table B.2.1 and begin with the designator "IE." The probabilities for the basic events also are given in Table B.2.1.

[b]Basic event HPI-LINE-BREAK is a type TRUE event. This type of event is not normally included in the output of the fault tree reduction process, but has been added to aid in understanding the sequences to potential core damage associated with the event. This basic event was added to those cut sets that required HPI-LINE-BREAK for the cut set to occur.

[c]Components in the DHR system are shared with the LPI system. For example, the DHR motor-driven pumps are also the LPI pumps when providing LPI, or piggy-back cooling. Therefore, failure of the DHR pumps results in the failure of both DHR and PB-COOL in this sequence.

[d]This basic event represents the common-cause failure of HPI pumps B and C failing to start. At Oconee, one of the HPI pumps (pump A in this analysis) is always running to provide normal RCS makeup and seal injection; however, its flow is lost due to the HPI line break.