C.2 LER Nos. 213/94-004, -005, -007, -013; IR 213/94-03

Event Description: Power-Operated Relief Valves and Vital 480-V ac Bus Degraded

Date of Event: February 16 and 19, 1994

Plant: Haddam Neck

C.2.1 Event Summary

On February 16, 1994, testing revealed that one of two feeds to motor control center-5 (MCC-5) could jam and fail to close when demanded. MCC-5 supplies power to a number of vital components in both safety system trains. During testing on February 19, 1994, it was discovered that air operators for the pressurizer power-operated relief valves (PORVs) were experiencing control air leaks and that the PORVs could not be operated properly from their safety-grade control air supply. Investigation revealed that repairs to fix a prior PORV failure were made incorrectly during the previous refueling outage. The PORV diaphragms were not seated correctly and were coated with a lubricant rather than a required sealant. Substantial air leaks resulted, and the PORVs could not be opened more than 50%. The combined conditional core damage probability estimated for these events is 1.4×10^{-4}.

C.2.2 Event Description

During a maintenance outage, an operability surveillance test was performed on the pressurizer PORVs on February 19, 1994. This test revealed that both PORV air operators had leaking diaphragms (LER 213/94-005). The PORV diaphragms had been replaced during the 1993 refueling outage following a diaphragm leak in one of the two PORVs (LER 213/93-007).

Surveillance testing of the PORVs in May 1993 revealed that one valve was experiencing leakage from its diaphragm assembly (LER 213/93-007). This leak, in conjunction with failure of the associated air pressure regulator, resulted in excessive air consumption. Had the system been demanded, operator action to isolate the leaking PORV would have been required to ensure an adequate long-term supply of control air to the other PORV. Repairs to the system, including replacement of the PORV diaphragms, were completed prior to the end of the 1993 refueling outage.

The design of the new diaphragms varied somewhat from the original ones, which may have contributed to the difficulties experienced during the replacement process. Errors were made during the replacement, including the use of a lubricant instead of a sealant around the diaphragm's bolt circle. This allowed the diaphragm to extrude out between the sections of its housing, creating a pathway for air leakage. An NRC inspection team report related to this event (50-213/94-03, April 7, 1994) indicates that both valves could only be opened about 50% during testing. The LER for the event indicates that two safety functions were potentially compromised by the PORV failures: feed-and-bleed cooling and high-pressure safety injection (HPSI) makeup during certain small-break loss-of-coolant accidents (LOCAs).

The HPSI pumps at Haddam Neck do not develop sufficient discharge head to force adequate flow for feed-and-bleed cooling through the pressurizer safety valves. Accordingly, the operators must be able to open a PORV for feed-and-bleed cooling to succeed. Air is supplied to the PORVs from the containment air compressors. The containment air compressors, which are located within the containment building, are not rated for the environmental conditions that could occur during feed-and-bleed cooling, and the compressors could be expected to fail under such conditions. The PORVs are provided with safety-related control air accumulators that maintain a reserve supply of control air in the event of compressor failure, but these accumulators were inadequate to operate the PORVs during the time that the air-operator diaphragms were damaged. LER 213/94-005 reported that air leakage would have resulted in the eventual loss of air and closure of the PORVs for feed-and-bleed conditions. As a result of their incorrect installation, the PORV air-operator diaphragms were damaged and subject to leakage from some unknown time after they were were replaced during the 1993 refueling outage until the condition was discovered on February 19, 1994.

LER 213/94-005 also identified a concern related to the provision of HPSI minimum flow protection by the PORVs. During small-break LOCA sequences, the HPSI minimum flow recirculation line to the refueling water storage tank is isolated, and minimum flow protection is provided by opening the PORVs. With the PORVs inoperable, this protection would not be provided, and the HPSI pumps would be subject to damage if reactor coolant system (RCS) pressure remained above the HPSI pump shutoff head. The LER indicates that an alternate strategy of using charging flow would be successful in maintaining the RCS filled for small break sizes that would not be large enough to ensure minimum necessary HPSI flow.

LER 213/94-004 reports that, during a period of time overlapping the PORV unavailability, the automatic bus transfer (ABT) circuit for MCC-5 failed when tested. At the time of the event, MCC-5 supplied many pieces of important equipment in both trains, including equipment that may have been required for successful operation of HPSI, low-pressure safety injection (LPSI), recirculation, long-term cooling, containment spray, RCS loop isolation, one PORV block valve, emergency boration, feedwater isolation, reactor coolant pump (RCP) seal cooling, service water, control air, and the closed cooling water system. Subsequent to this event, modifications were made to reduce the dependency upon MCC-5.

MCC-5 can be supplied from either 480-V ac bus 5 (emergency train A) or bus 6 (emergency train B). Normally, the alignment is aligned such that bus 5 is the preferred supply and bus 6 is the alternate supply. At the time of the event, if the preferred supply was lost, the ABT system aligned MCC-5 to the alternate bus. If power was restored to the preferred bus, the ABT would realign back to the preferred bus. During a test of the ABT system, bus 5 was deenergized. As designed, the breaker supplying MCC-5 from bus 5 opened, and the supply breaker from bus 6 automatically closed to restore power. When bus 5 was reenergized, MCC-5 automatically realigned itself to bus 5. During the second part of the test, the preferred power source selector switch (PPSSS) for the ABT was moved to make bus 6 the preferred power supply and bus 5 the alternate. When the PPSSS was moved to the bus 6 position, the bus 5 supply breaker opened as expected, but the bus 6 supply breaker failed to automatically close, deenergizing MCC-5.

Subsequent investigation revealed that a mechanical defect in the MCC-5 feeder breaker from bus 6 prevented it from closing. With bus 6 still energized and selected as the preferred power source to MCC-5, the bus 5 supply to MCC-5 was prevented from closing by the ABT system logic. NRC inspection report 50-213/94-03 indicated that the likely cause of the failure of breaker 11C, the feeder from bus 6, was mispositioning of a breaker component ("snap ring") during maintenance. The snap ring being improperly located would cause the 11C breaker to have intermittent failures. Vibrations of the breaker would cause the trip to occur at times and not to occur at other times. This condition would result in intermittent failures of the MCC-5 ABT. The fraction of time that the breaker may have operated is unknown.

LER 213/94-013 reported the failure of a HPSI common header relief valve discovered during testing on May 5, 1994. While the actual lift pressure for the valve was not stated, it was reported that it did not lift during operation of the A pump, which developed a discharge pressure of about 1460 psig; but the valve did lift prematurely during operation of the B pump, which developed about 1510 psig. Leakage flow back to the refueling water storage tank (RWST) through this valve was limited to a maximum of 35 gpm. The condition is reported to have existed from the time that the B pump was overhauled in 1993 until discovery on May 5, 1994.

LER 213/94-007 reported the discovery that the chemical volume and control system (CVCS) pump common header discharge relief valve minimum lift set point was 2653 psig. The maximum charging pump discharge pressure under accident conditions was estimated to be about 2658 psig. Maximum flow through this relief valve is 30 gpm, which would be directed to radwaste drain tanks. Since CVCS is utilized to provide high-pressure recirculation, this represents a potential diversionary flow path from the CVCS during recirculation.

C.2.3 Additional Event-Related Information

The description of this event and the modeling assumptions are based on the plant status at the time of the event. Subsequent design changes have been made to reduce the likelihood and risk of future failures, such as elimination of the PPSSS MCC-5 ABT. Some plant modifications initiated after the June 1993 MCC-5 bus transfer failure that were complete at the time of this event included shifting the power supply from MCC-5 to MCC-12 for one residual heat

removal (RHR) to charging pump suction valve, the A charging pump main lube oil pump, and one PORV block valve. The power supply to PORV PR-AOV-570 was also shifted to another source.

C.2.4 Modeling Assumptions

C.2.4.1 General Modeling Issues

The NRC inspection report related to this event (Ref. 5) indicates that the PORVs are required to remain operable for 30 h and provide a total of four valve strokes during feed-and-bleed scenarios. The measured control air leak rate was such that, during an actual event involving loss of the containment control air compressors and PORV demand, the PORV control air accumulators would have been depleted within minutes. Although the valves were able to partially open during testing, the valves would not be able to stay open for the required duration. Further, the containment air compressors are not rated for the containment environment that is expected after initiation of feed-and-bleed cooling. Therefore, the event was modeled as an unavailability of the PORVs for feed-and-bleed cooling. The PORVs would still be functional for overpressure protection of the reactor coolant system.

LER 213/94-005 indicates that the last successful operation of the PORVs was during an outage in May and June of 1993 following installation of the new diaphragms. It further indicates that the likely cause of the PORV failure was incorrect installation of the air-operator diaphragms during the 1993 outage. It was, therefore, assumed that the PORVs were inoperable for feed-and-bleed cooling from July 1993 until the leakage was discovered on February 19, 1994. This was modeled by setting the PORVs as failed for feed-and-bleed conditions at the appropriate places in the model.

The defect, which led to the intermittent failure of the bus 6 feeder breaker, was presumed to have existed from the time of the previous failure during the June 1993 refueling outage until the time of this event in February 1994. The interval analyzed was the period from July 21, 1993, until February 19, 1994, a period of 234 days (4728 h). Although the failure mechanism was intermittent, the fraction of time the component would have operated is unknown. It was assumed the breaker was failed throughout the period of interest. This is a conservative assumption.

The potential loss of HPSI minimum flow protection was not modeled because alternate means, such as the charging system, were available for RCS makeup in event of a small-small-break LOCA.

The potential failure of the HPSI relief valve during operation of train B in recirculation mode after a small-break LOCA was not modeled because, according to information from the LER, it would probably reseat following initiation of sump recirculation with secondary side cooling available. In any event, the maximum potential loss estimated for this pathway during a 24-h demand would be about 50,000 gal, which would still leave adequate sump inventory.

Potential failure of the CVCS relief valve was not modeled because LER 213/94-007 indicated that expected losses would be much less than the maximum relief valve flow rate of 30 gpm. The potential total diversion within a 24-h mission time is less than for the HPSI relief valve and would not affect system operability.

This analysis is structured similarly to the analysis of LER 213/93-007 and Augmented Inspection Team (AIT) Report 213/93-80 provided in the 1993 Accident Sequence Precursor (ASP) Program Annual Report (NUREG/CR-4674, ORNL/NOAC-232, Vols. 19 and 20). That analysis also dealt with failures of PORV control air system components coincident with inoperability of the MCC-5 ABT. Minor modifications to the 1993 analysis were required to adapt the approach to the current event. Those modifications are noted.

Challenge Rate for Pressurizer PORVs and Safety Relief Valves (SRVs)

The PORV block valves are maintained in a closed position at Haddam Neck, and at least one is dependent on MCC-5. Further, the PORVs are assumed failed in this analysis due to the diaphragm air leaks. Therefore, the PORV/SRV challenge rate applies solely to the SRVs after a loss-of-offsite power (LOOP) with MCC-5 unavailable. Since the PORV block valves are normally closed, it was assumed that the lift rate for SRVs is the same as when both the PORVs and SRVs are available. Therefore, this value was not modified.

PORV/SRV Reseat of Challenged Pressurizer PORVs and SRVs

It was assumed that the failure to reseat probability for the SRVs is the same as for the PORVs. The nonrecovery value was set to 1.0 because the safety valves do not have block valves.

Feed-and-Bleed

Feed-and-bleed requires the operation of HPI or the charging pumps, the high-pressure recirculation system (HPR), and the pressurizer PORVs. One HPI or charging pump and one PORV are required for success. Because the PORVs would not remain open for the required duration, feed-and-bleed was assumed inoperable.

C.2.4.2 Transient and Small-Break LOCA Sequences

Two cases were used to model the effects of the failed PORVs during transient and small-break LOCA conditions.

In the first case (IRRAS case 1A), the transient initiating probability was set to 1.0 (true), and the PORVs were failed (set to true). All other initiator probabilities were set to 0 (ignore).

The probability of a transient during the vulnerability period was calculated as follows:

$$1 - \exp(-\lambda t) = 1 - \exp\,[-(1.85 \times 10^{-4}/\mathrm{h}) \times (4728\ \mathrm{h})] = 0.58\ .$$

In the second case (IRRAS case 1B), the small-break LOCA initiating probability was set to 1.0 (true), and the PORVs were failed (set to true). All other initiator probabilities were set to 0 (ignore).

The probability of a small-break LOCA during the vulnerability period was calculated as follows:

$$1 - \exp(-\lambda t) = 1 - \exp\,[-(1.0 \times 10^{-6}/\mathrm{h}) \times (4728\ \mathrm{h})] = 4.7 \times 10^{-3}\ .$$

The initiating event probabilities and IRRAS case conditional probabilities were used to calculate the core damage probabilities from these initiators (see Table C.2.3 on p. C.2-11).

C.2.4.3 LOOP Sequences

To address the potential loss of MCC-5 and the failed PORVs following a postulated LOOP, a conditioning event tree was used. This event tree characterized potential plant conditions involving emergency diesel generator (EDG) success and failure, short-term (30-min) LOOP recovery, and short-term MCC-5 recovery. The event tree, shown in Figure C.2.1, includes the conditioning sequences shown in Table C.2.1.

Table C.2.1. Sequences for conditioning event tree in Figure C.2.1

Sequence	Description
1	Initial emergency power (EP) success with short-term recovery of offsite power and MCC-5 following the postulated LOOP. This is similar to a loss of feedwater but with a higher probability of a transient-induced LOCA, because the SRVs would lift (if necessary) as a result of the inoperable PORVs. Feed-and-bleed is failed in IRRAS Case 2.1 because of the inoperable PORVs.
2	Initial EP success and short-term recovery of offsite power but with MCC-5 not recovered at 30 min. This is similar to sequence 1, but with the potential for an RCP seal LOCA if MCC-5 is not recovered at 1 h. HPI is assumed unavailable if MCC-5 is not recovered ~0.5 h following a seal LOCA. HPI following a stuck-open SRV and feed-and-bleed are also assumed to be unavailable in IRRAS Case 2.2 since MCC-5 is unavailable at 30 min.
3	LOOP with EP initially successful, MCC-5 recovered, and feed-and-bleed unavailable (IRRAS Case 2.1). Higher probability of a transient-induced LOCA.
4	LOOP with EP initially successful but neither MCC-5 nor offsite power recovered at 30 min. There is a higher potential for an RCP seal LOCA if MCC-5 is not recovered. There is also a higher probability of a transient-induced LOCA. HPI following a stuck-open relief valve and feed-and-bleed are also assumed to be unavailable (IRRAS Case 2.2) since MCC-5 is unavailable at 30 min.
5	Station blackout.
6	Anticipated transient without scram.

LOOP Initiating Event Probability

The probability of a LOOP during the vulnerability period was calculated as follows:

$$1 - \exp(-\lambda t) = 1 - \exp[-(1.6 \times 10^{-5}/\text{h}) \times (4728\ \text{h})] = 7.3 \times 10^{-2}.$$

The vulnerability period was estimated at 4728 h. This is the operational time between plant restart in 1993 and discovery of the PORV problem in February 1994.

Failure to Trip Probability

The failure to trip probability was not modified for this event. The value from the IRRAS model for Haddam Neck is 2×10^{-5}. This includes RPS hardware failures and subsequent operator recovery of the system.

Emergency Power

The probability for emergency power failure was not modified. This probability (2.3×10^{-3}) includes operator recovery following postulated EDG failures.

LOOP Recovery in the First 30 min

The probability for failure to recover the LOOP in the first 30 min was based on LOOP recovery models described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11. These models are based on the results of the data contained in NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*.

MCC-5 Failure and Restoration

Based on the condition of breaker 11C (feeder from bus 6 to MCC-5) and the unpredictability of its observed failures, breaker 11C was assumed to be failed in this analysis. In addition to the failure of breaker 11C, one additional failure

must occur for MCC-5 to lose power. Either breaker 9C (feeder from bus 5) must fail to reclose, or EDG A must fail to start and run.

After a LOOP with the PPSSS in the bus 5 (normal) position, power would be lost to buses 5 and 6. Two cases could then occur.

- Bus 5 is re-energized before bus 6. In this case, breaker 9C will attempt to reclose. If 9C fails to close, the ABT will automatically try to close breaker 11C once bus 6 is energized. However, since breaker 11C is assumed to be failed, manual operator action is required to restore power to MCC-5.
- Bus 6 is re-energized before bus 5. In this case, breaker 11C will attempt to close. Assuming 11C fails to close, the ABT will attempt to automatically reclose breaker 9C after power is restored to bus 5. If breaker 9C fails to reclose, manual operator action is required to restore power to MCC-5. Data collected by the licensee on EDG performance indicate that the time to rated speed and voltage for both of the EDGs was essentially the same. This would mean that bus 6 would reach rated voltage first about 50% of the time. (Circuit timing delays may affect this value somewhat but would have little impact on the analysis results.) Assuming that breaker 11C will fail to close on demand, and a beta factor of 0.1 for breaker 9C since it was subject to the same maintenance procedures as the failed 11C breaker, the probability of failure of the ABT given a LOOP is:

$$[p(5before6) \times p(9C|11C)] + [p(\overline{5before6}) \times \{p(EDG\,A) + p(9C|11C)\}] =$$

$$[p(5before6) \times p(9C|11C)] + p(\overline{5before6}) \times p(EDG\,A) + p(\overline{5before6}) \times p(9C|11C) =$$

$$p(9C|11C) + [p(\overline{5before6}) \times p(EDG\,A)] = 0.1 + (0.5 \times 0.05) = 0.125.*$$

The licensee performed a detailed analysis of MCC-5 failure probabilities. Their assessment indicates that the probability that MCC-5 fails to supply power is 0.059 for LOOP events. However, this assumed a nominal failure rate for breaker 11C.

To recover MCC-5 following a failure of the ABT, an operator must proceed to MCC-5, diagnose the situation, and manually close one of the MCC-5 feeder breakers. During the June 1993 event, an operator took 4 min to complete this action. However, the operator was already stationed at the selector switch, was immediately aware of the ABT failure, and had a minimum of other distractions and stresses. Similarly, during one of two ABT test failures on February 16, 1994, operators took approximately 3 min to repower MCC-5 from bus 5. The time required to repower MCC-5 during the second event is not known.

Following a postulated LOOP with the failure of MCC-5, additional delays would be introduced, including detection time, delays for the control room to contact an auxiliary operator and describe the problem, and operator transit time. Unavailability of power on MCC-5 is not directly addressed in procedure E-0, "Reactor Trip or Safety Injection," until step 16. A median value was used in the analysis; this assumes 6 min for diagnosis and transit time and the observed ~4 min for recovery at the equipment. A 6-min diagnosis and transit time is considered possible because of the proximity of MCC-5 to the control room. [The 10-min median value is somewhat longer than the licensee's estimate of 5 to 6 min (2- to 3-min diagnostic time, 1-min transit, and 2 min to operate breakers) and somewhat shorter than a 16-min value that can be estimated based on a distribution of transit times in response to a faulted EDG (another important component) included in "Electric Power Recovery Models," J. W. Reed and K. N. Fleming, *Proceedings of the International Topical Meeting on Probabilistic Safety Assessment*, PSA'93, January 26–29, 1993.]

The probability of not recovering MCC-5 was estimated by assuming that the 10-min period was the median of a lognormal distribution with an error factor of 3.2 (see Dougherty and Fragola, *Human Reliability Analysis*, John Wiley

* For situations where offsite power is recovered within 30 min, the probability for MCC-5 failure is 0.1.

and Sons, New York, 1988, Chap. 10). This is the error factor for time-reliability correlations (TRCs) for actions without hesitancy, which is considered appropriate based on the recognized importance of MCC-5. Three primary time intervals for MCC-5 recovery were considered in this analysis. These intervals and the associated MCC-5 nonrecovery probabilities are shown in Table C.2.2.

Table C.2.2. MCC-5 nonrecovery values

Time interval (h)	p(MCC-5 not recovered)
0.5	6.0×10^{-2}
1.0	5.6×10^{-3}
1.5	9.5×10^{-4}

For the conditioning event tree, the probabilities of MCC-5 failure followed by failure to recover MCC-5 were determined as follows:

$$p(\text{MCC-5 failed and not recovered} \mid \text{LOOP recovered within first 30 min}) = 0.1 \times (6.0 \times 10^{-2}) = 6.0 \times 10^{-3}$$

$$p(\text{MCC-5 failed and not recovered} \mid \text{LOOP not recovered within first 30 min}) = 0.125 \times (6.0 \times 10^{-2}) = 7.5 \times 10^{-3}\,.$$

Calculations for LOOP Conditioning Sequences 1 and 3 (IRRAS Case 2.1)

To reflect the conditions assumed in Figure C.2.1 for these sequences, the IRRAS model for Haddam Neck was evaluated with the LOOP initiator set to 1.0, both PORVs failed for feed-and-bleed only (set to true) and nonrecoverable for LOOP conditions (see General Modeling considerations for a discussion of the PORV operability), emergency power successful (basic events for both EDGs set to false), and short-term LOOP nonrecovery set to 1.0. Potential EDG failures are addressed in the conditioning event tree (Figure C.2.1). Other initiators were ignored for this calculation.

Calculations for LOOP Conditioning Sequences 2 and 4 (IRRAS Case 2.2)

For these two sequences, two calculations were performed. In the first, the IRRAS model was evaluated with the LOOP initiator set to 1.0, both PORVs failed for feed-and-bleed only (set to true) and nonrecoverable under LOOP conditions, emergency power successful (basic events for both EDGs set to false), short-term LOOP nonrecovery set to 1.0, and both HPI pumps failed (set to True) and nonrecoverable for both HPI and HPR. HPI is assumed unavailable if MCC-5 is not recovered ~0.5 h following a seal LOCA. HPI following a stuck-open SRV and feed-and-bleed are also assumed to be unavailable since MCC-5 is unavailable at 30 min. Potential EDG failures are addressed in the conditioning event tree on Figure C.2.1.

In addition to the IRRAS calculation, an event tree was developed to address the possibility of a seal LOCA (Figure C.2.2). This tree was quantified as follows.

MCC-5 Recovered Before Seal LOCA and Seal LOCA Probabilities

Operator action is required to recover either means of RCP seal cooling (seal injection and thermal barrier cooling) following a LOOP and the loss of MCC-5. Component cooling water, which provides thermal barrier cooling, is lost following the LOOP due to the loss of instrument air. The charging pumps, which provide seal injection, also trip following a LOOP due to an automatic tripping feature that had recently been installed. During the 1993 ABT failure event, because the main lube oil pumps for the charging pumps were powered from MCC-5, the charging pumps could

not be restarted without first recovering MCC-5 or aligning the alternate lube oil pumps. After the 1993 event, the power supply to A charging pump lube oil pump was realigned to MCC-12. However, instrument air is powered from MCC-5, and loss of MCC-5 would cause the charging flow control valve to go wide open, depriving RCP seals of flow. Operator actions would be required to either recover MCC-5 or to throttle charging flow and restore seal injection.

The potential impact of an RCP seal LOCA following loss of MCC-5 but with emergency power available was addressed in the event tree model shown in Figure C.2.2. This model is applicable to sequences involving emergency power and auxiliary feedwater (AFW) success with the SRV closed. In this model, MCC-5 must be recovered or the charging system must be restarted and realigned to prevent an RCP seal LOCA. Given that a seal LOCA has occurred, HPI and HPR are required to prevent core damage. Recovery of HPI requires recovery of MCC-5 or the charging system.

To simplify the analysis, an RCP seal LOCA was assumed likely in nonblackout sequences if MCC-5 or the charging system are not recovered at 1 h. The probability of not recovering MCC-5/charging system at 1 h, given that they were not recovered at 0.5 h (this probability is addressed in a conditioning event tree branch), was estimated to be

$$p(\text{MCC-5 recovered at 1 h} \mid \text{MCC-5 not recovered at 0.5h}) = (5.6 \times 10^{-3}/6.0 \times 10^{-2}) = 9.3 \times 10^{-2}.$$

The probability of seal LOCA occurring at this time was assumed to be 0.7, consistent with other ASP analyses.

HPI High-Pressure Injection

Following the loss of MCC-5, the HPI system is lost (Haddam Neck IPE Table B-1, System/Function 2, Sump Recirculation). Restoration of power to MCC-5 is required to regain HPI function. The charging pumps are also unavailable following a loss of MCC-5 until they are realigned and restarted.

For a stuck-open SRV, the probability of HPI failure, given that MCC-5 was not recovered at 0.5 h, was assumed to be 1.0. For an RCP seal LOCA with emergency power initially available, the failure probability for HPI was estimated to be 0.17:

$$p(\text{MCC-5 not recovered 0.5 h after a potential seal LOCA} \mid \text{MCC-5 not recovered at 1.0 h})^*$$

$$= 9.5 \times 10^{-4}/5.6 \times 10^{-3}$$

$$= 0.17\,.$$

HPR High-Pressure Recirculation

The failure probability for HPR was determined by using the system failure probability from the IRRAS model for Haddam Neck.

Calculations for LOOP Conditioning Sequence 5

The event tree model used to address potential seal LOCAs following a station blackout is shown in Figure C.2.3. This model utilizes the same assumptions regarding the onset of a seal LOCA and recovery of HPI as the nonblackout case.

AFW Auxiliary Feedwater

Normal AFW flow control is dependent on MCC-5. However, flow control is also possible using the hydraulically powered turbine steam admission valves. AFW flow is controlled using these valves during startup and shutdown, so operators are familiar with their use. Therefore, nominal AFW response was assumed following the postulated loss of MCC-5.

* Onset of seal LOCA assumed at 1 h—see MCC-5 failure and restoration.

MCC-5 Vulnerable to Failure When Power Restored

Following restoration of power, MCC-5 is vulnerable to failure if breaker 9C fails to operate. The failure probability of breaker 9C is assumed to be 0.1 (the same as the beta factor) since the breaker was exposed to the same maintenance practices that led to the failure of breaker 11C.

AC Power and MCC-5 Recovered in 1 h

For blackout sequences, both ac power and MCC-5 (or charging) must be recovered to prevent an RCP seal LOCA. The probability of not recovering both in 1 h (the time at which RCP seal LOCAs are assumed to begin) is estimated to be 0.17 based on a convolution approach.

When MCC-5 is not vulnerable to failure when power is restored, the probability of failing to recover ac power is estimated to be 0.12 based on LOOP recovery models described in ORNL/NRC/LTR-89/11.

Seal LOCA Probability

As discussed above for the event tree in Figure C.2.2, the probability of an RCP seal LOCA occurring at 1 h was assumed to be 0.7, consistent with other ASP analyses.

HPI High-Pressure Injection

Following the loss of MCC-5, HPI is lost (Haddam Neck IPE Table B-1, System/Function 2, Sump Recirculation). Restoration of power to MCC-5 is required to regain HPI function. The charging pumps are also unavailable following a loss of MCC-5 until they are realigned and restarted.

For an RCP seal LOCA following a station blackout, HPI recovery requires the recovery of both AC power and MCC-5 (or charging). The probability of failing to recover either of these, given they were not recovered at 1 h, is estimated to be 0.57. This value was approximated as:

p(offsite power not recovered at 1.5 h | offsite power not recovered at 1 h, 0.47) +

p(MCC-5 not recovered at 1.5 h | MCC-5 not recovered at 1 h, 0.17) .

AC Power Recovered in 6 h (Prior to Battery Depletion)

The probability of failing to recover offsite power before battery depletion at 6 h was estimated to be 0.037, based on LOOP recovery models described in ORNL/NRC/LTR-89/11. These models are based on the results of the data contained in NUREG-1032. The probabilities of ac recovery at 6 h, given it was not recovered at 1 h, were calculated as follows:

p(offsite power recovered at 6 h | offsite power not recovered at 1 h and MCC-5 vulnerable to failure when power is restored)

= 0.037/0.17

= 0.22,

p(offsite power recovered at 6 h | offsite power not recovered at 1 h and MCC-5 NOT vulnerable to failure when power is restored)

= 0.037/0.12

= 0.31.

C.2.4.4 Core Damage Probability Calculation

Calculations were structured to parallel the similar precursor analysis of MCC-5 potential unavailability coincident with PORV failures, which was performed in 1993 for AIT 213/93-80, LERs 213/93-006 and -007 (see NUREG/CR-4674, ORNL/NOAC-232, Vols. 19 and 20).

The impact of the failed PORVs on feed-and-bleed following postulated transients and small-break LOCAs was assessed by setting the PORVs to true (failed) in the model and calculating the associated conditional core damage probability given the initiator. This value was then multiplied by the probability that those initiators would occur during the time interval between startup in July 1993 and discovery of the PORV failure in February 1994.

To address the loss of MCC-5 and the failed PORVs following a postulated LOOP, a conditioning event tree was used. This event tree characterizes potential plant conditions involving EDG success and failure, short-term (30 min) LOOP recovery, and short-term MCC-5 recovery. The event tree is shown in Figure C.2.3.

Table C.2.3 provides the relevant branch and conditioning sequence probabilities and identifies the calculation or IRRAS case associated with each sequence. Specific model probability modifications are indicated in the tables of selected basic events that are included with this analysis.

The conditional probabilities estimated in calculations 1A (feed-and-bleed unavailable during transients), 1B (feed-and-bleed unavailable following a small-break LOCA), 2-1 (conditioning sequences 1 and 3), 2-2 (conditioning sequences 2 and 4), Figure C.2.2 (seal LOCA for nonblackout sequences), and Figure C.2.3 (station blackout) were combined with the probabilities of such sequences occurring in the observation period to estimate the conditional probability for the combined event.

The sum of the probabilities for the sequences is 1.4×10^{-4}.

For operational events involving unavailabilities such as this event, the ASP Program estimates the core damage probability for the event by calculating the probability of core damage during the unavailability period conditioned on the failures observed during the event and subtracting a base-case probability for the same period, assuming plant equipment performs nominally. Because a conditioning event tree was used to analyze some of the sequences associated with a postulated LOOP, the computer code was not used to perform this differential calculation. Instead, the calculation program was used to calculate the probability of core damage given the conditions observed during the event and a postulated initiating event. This probability was then multiplied by the probability of the initiator during the unavailability period. The nominal core damage probability was estimated in the same way. For this analysis, the nominal core damage probability for the period analyzed was found to be small and was neglected.

Table C.2.3. Summary of conditional core damage probabilities

Sequence	p(sequence)	p(cd | sequence)	p(cd)	% Contribution
1A Transient	5.8×10^{-1}	5.2×10^{-5} (IRRAS Case 1A)	3.0×10^{-5}	21.0
1B Small-break LOCA	4.7×10^{-3}	6.6×10^{-4} (IRRAS Case 1B)	3.1×10^{-6}	2.2
2.1 LOOP*	5.5×10^{-2}	7.9×10^{-4} (IRRAS Case 2-1)	4.3×10^{-5}	30.0
2.2 LOOP*	3.3×10^{-4}	5.9×10^{-2} (IRRAS Case 2-2)	1.9×10^{-5}	13.3
		1.1×10^{-2} (seal LOCA, Fig. C.2.2.)	3.6×10^{-6}	2.5
2.3 LOOP*	1.7×10^{-2}	7.9×10^{-4} (IRRAS Case 2-1)	1.3×10^{-5}	9.1
2.4 LOOP*	1.3×10^{-4}	5.9×10^{-2} (IRRAS Case 2-2)	7.7×10^{-6}	5.4
		1.1×10^{-2} (seal LOCA, Fig. C.2.2.)	1.4×10^{-6}	1.0
2.5 LOOP*	1.7×10^{-4}	1.3×10^{-1} (blackout, Fig. C.2.3.)	2.2×10^{-5}	15.4
		Total	1.4×10^{-4}	100

*See Table C.2.2 for a description of the LOOP sequences.

C.2.5 Analysis Results

The conditional core damage probability estimated for the combined event is 1.4×10^{-4}. Postulated LOOPs (Cases 2.1 through 2.5) contribute approximately 77% of the core damage probability. The dominant sequence, shown in Figure C.2.4, which contributes about 30% of the total, involves a postulated LOOP, emergency power success, recovery of ac power and MCC-5, and failure of AFW and feed-and-bleed cooling. Selected basic event probabilities, sequence probabilities, system names, and conditional cut sets for each of the IRRAS cases are shown in Tables C.2.4 through C.2.19.

C.2.6 References

1. LER 213/94-004, Rev. 1, "Automatic 480 Volt Bus Transfer Failure Due to Circuit Breaker Malfunction," May 26, 1994.
2. LER 213/94-005, "Pressurizer PORVs Failed to Fully Stroke Open During Testing," March 18, 1994.
3. LER 213/94-007, "Potential for Radiological Release During Post-LOCA Sump Recirculation," April 5, 1994.
4. LER 213/94-013, "HPSI Pump Discharge Relief Valve Setpoint Found Low," June 3, 1994.
5. NRC Inspection Report 213/94-03, April 7, 1994.

4. LER 213/94-013, "HPSI Pump Discharge Relief Valve Setpoint Found Low," June 3, 1994.
5. NRC Inspection Report 213/94-03, April 7, 1994.

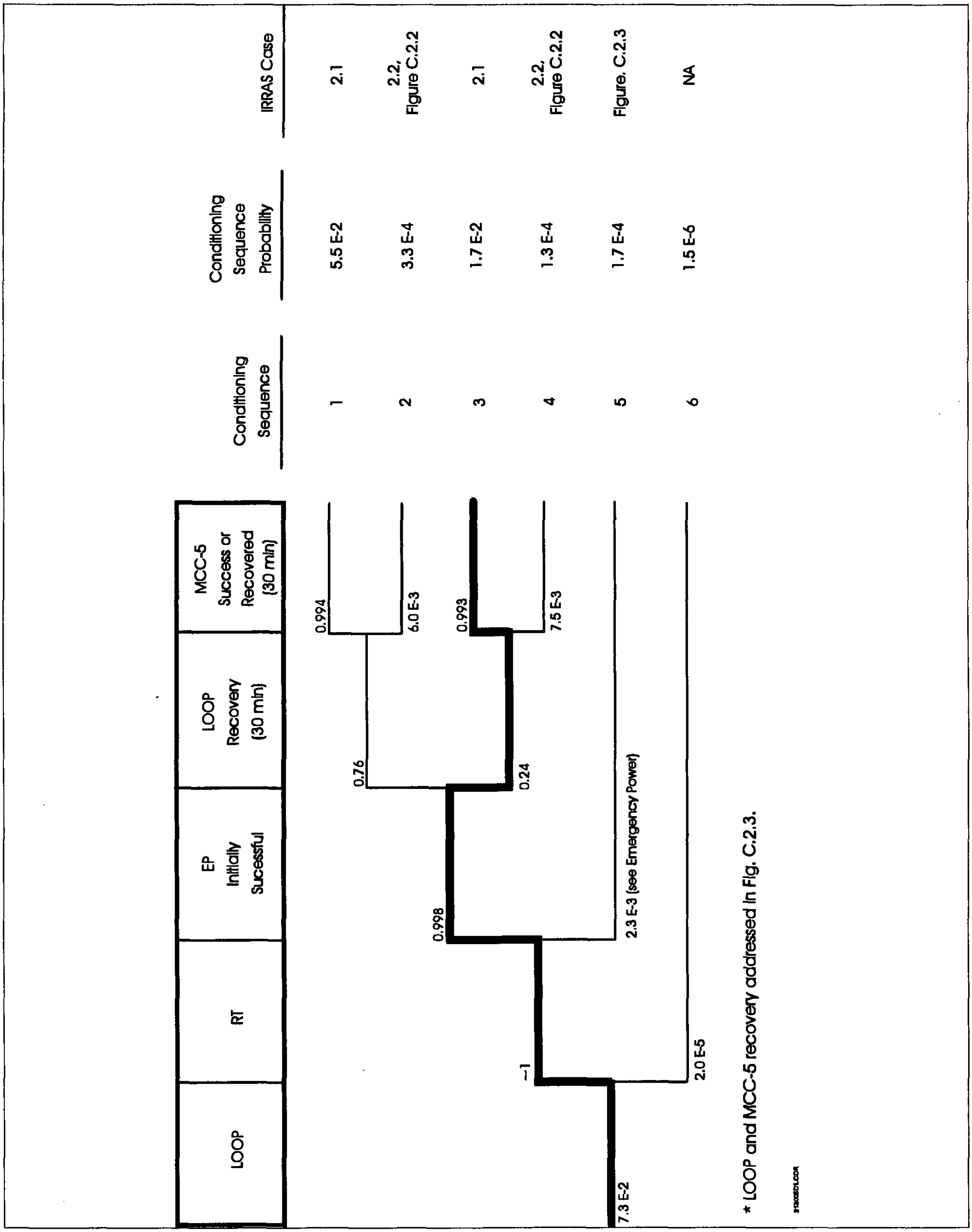

Figure C.2.1. Conditioning event tree for postulated LOOP.

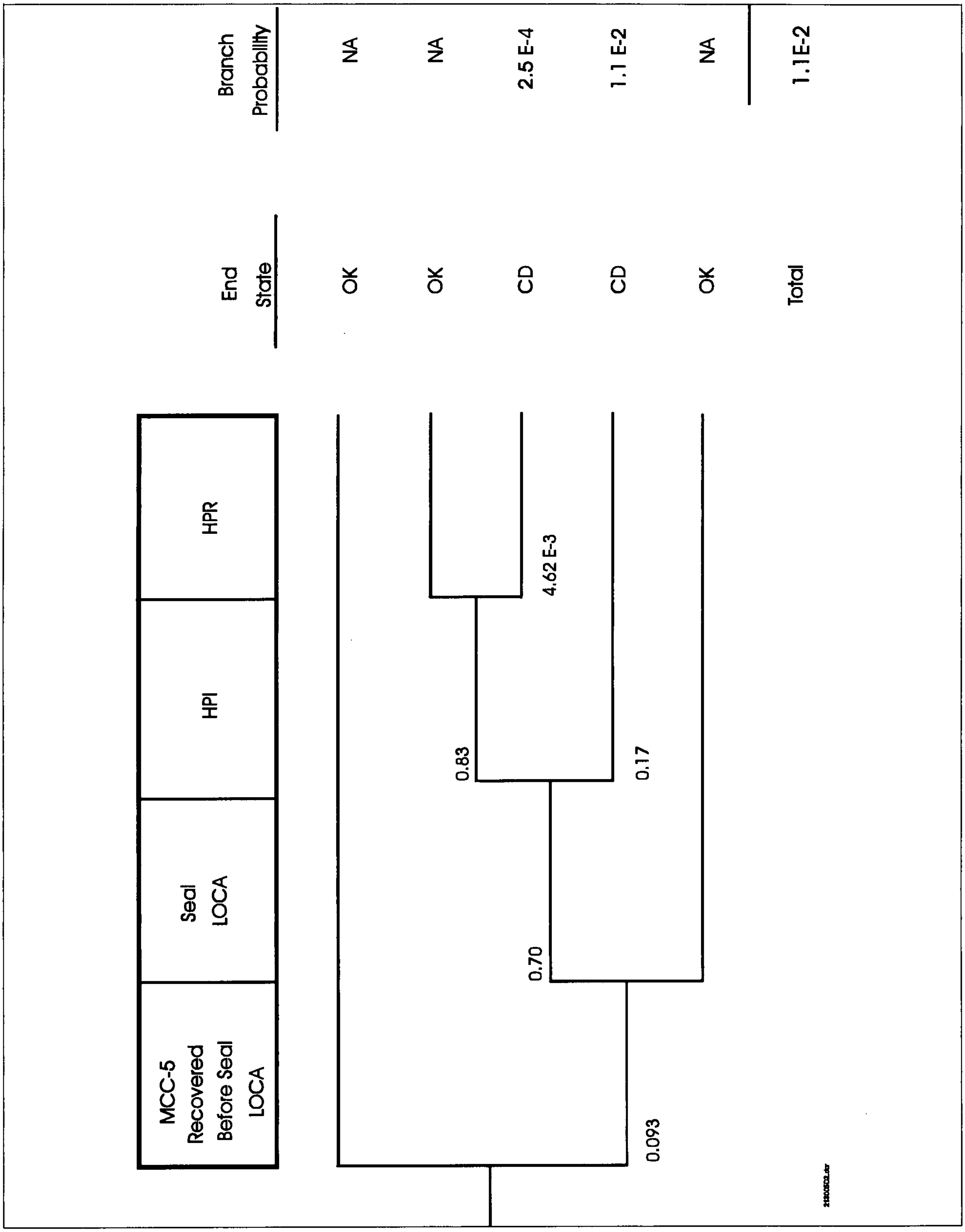

Figure C.2.2. Event tree for RCP seal LOCA (non-blackout sequences).

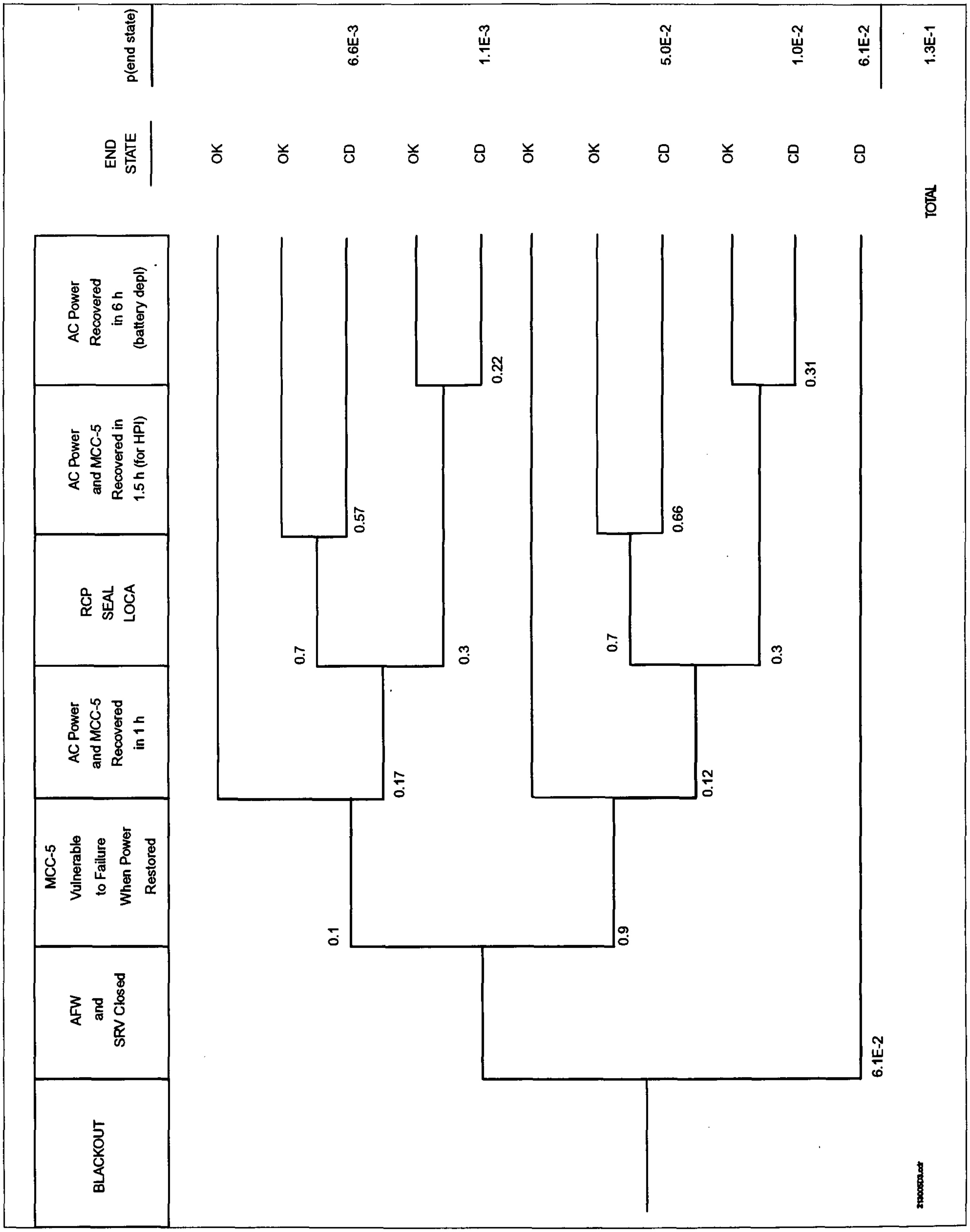

Figure C.2.3. Event tree model for blackout sequences.

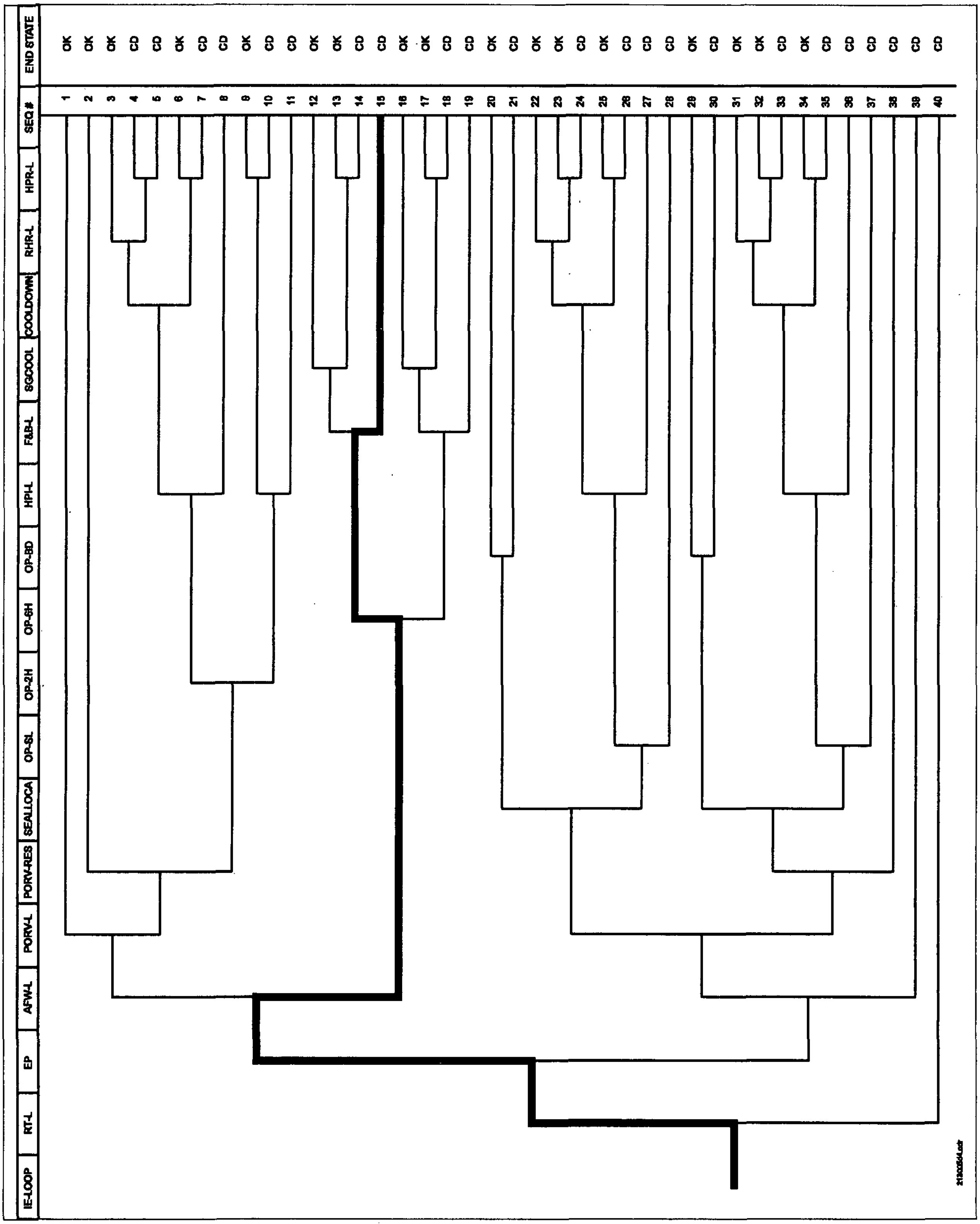

Figure C.2.4. Dominant core damage sequence for LER Nos. 213/94-004, -005, -007, -013 ; Inspection Report 213/94-03.

Table C.2.4. Selected basic events for Case 1A, PORVs unavailable during transients

Event name	Description	Base probability	Current probability	Type	Modified for this event
AFW-TDP-FC-1A	Failure of Turbine Driven Pump 1A	3.3E-002	3.3E-002		N
AFW-TDP-FC-1B	Failure of Turbine Driven Pump 1B	3.3E-002	3.3E-002		N
AFW-TDP-CF-AB	Common Cause Failures of Turbine Driven Pumps	1.4E-003	1.4E-003		N
AFW-XHE-NOREC	Operator Fails to Recover AFW System	2.6E-001	2.6E-001		N
AFW-XHE-NOREC-A	Operator Fails to Recover AFW System	2.6E-001	2.6E-001		N
AFW-XHE-RWSS-A	Operator Fails to Align Backup Water Source During ATWS	4.0E-002	4.0E-002		N
IE-LOOP	Loss-of-Offsite Power Initiating Event	8.5E-006	0.0E+000	IGNORE	Y
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.63E-006	0.0E+000	IGNORE	Y
IE-SLOCA	Small LOCA Initiating Event	1.0E-006	0.0E+000	IGNORE	Y
IE-TRANS	Transient Initiating Event	5.3E-004	1.0E+000	TRUE	Y
MFW-SYS-TRIP	Main Feedwater System Trips	2.0E-001	2.0E-001		N
MFW-XHE-NOREC	Operator Fails To Recover Main Feedwater	3.4E-001	3.4E-001		N
PPR-SRV-CC-PRV1	PORV 1 Fails To Open On Demand	6.3E-003	1.0E+000	TRUE	Y
PPR-SRV-CC-PRV2	PORV 2 Fails To Open On Demand	6.3E-003	1.0E+000	TRUE	Y
RPS-VCF-FO	Reactor Trip System Fails	6.0E-005	6.0E-005		N
RPS-XHE-XM-SCRAM	Operator Fails to Manually Trip The Reactor	3.4E-001	3.4E-001		N

Table C.2.5. Sequence probabilities for Case 1A, PORVs unavailable during transients

Event tree name	Sequence name	Frequency	% Contribution	Logic
TRANS	20	5.1E-005	98.3	/RT, AFW, MFW, F&B
TRANS	21-8	5.8E-007	1.1	RT, /RCSPRESS, AFW-ATWS
Total (all sequences)		5.2E-005	100.0	

Table C.2.6. System names for Case 1A, PORVs unavailable during transients

System name	Description
AFW	No or Insufficient AFW Flow
AFW-ATWS	No or Insufficient AFW Flow Following ATWS
F&B	Failure to Provide Feed-and-Bleed Cooling
MFW	Failure of the Main Feedwater System
RCSPRESS	Failure to Limit RCS Pressure to <3200 psi
RT	Reactor Fails to Trip During Transient

Table C.2.7. Conditional cut sets for higher probability sequences for Case 1A

Cut set No.	% Contribution	Frequency	Cut sets
TRANS Sequence: 20		5.1E-005	
1	48.0	2.4E-005	AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, AFW-TDP-CF-AB
2	37.4	1.9E-005	AFW-XHE-NOREC, AFW-TDP-FC-1B, MFW-SYS-TRIP, MFW-XHE-NOREC, AFW-TDP-FC-1A
TRANS Sequence: 21-8		5.8E-007	
1	36.0	2.1E-007	RPS-VCF-FO, RPS-XHE-XM-SCRAM, AFW-XHE-NOREC-A, AFW-XHE-RWSS-A
2	29.7	1.7E-007	RPS-VCF-FO, RPS-XHE-XM-SCRAM, AFW-XHE-NOREC-A, AFW-TDP-FC-1A
3	29.7	1.7E-007	RPS-VCF-FO, RPS-XHE-XM-SCRAM, AFW-XHE-NOREC-A, AFW-TDP-FC-1B
Total (all sequences)		5.2E-005	

Table C.2.8. Selected basic events for Case 1B PORVs unavailable following a small-break LOCA

Event name	Description	Base probability	Current probability	Type	Modified for this event
AFW-TDP-FC-1A	Failure of Turbine Driven Pump 1A	3.3E-002	3.3E-002		N
AFW-TDP-FC-1B	Failure of Turbine Driven Pump 1B	3.3E-002	3.3E-002		N
AFW-TDP-CF-AB	Common Cause Failures of Turbine Driven Pumps	1.4E-003	1.4E-003		N
AFW-XHE-NOREC	Operator Fails to Recover AFW System	2.6E-001	2.6E-001		N
HPI-MOV-OC-SUC	Suction MOV From RWST Fails	4.0E-005	4.0E-005		N
HPI-XHE-NOREC	Operator Fails to Recover the HPI System	8.4E-001	8.4E-001		N
HPR-XHE-NOREC	Operator Fails to Recover the HPR System	1.0E-000	1.0E-000		N
IE-LOOP	Loss-of-Offsite Power Initiating Event	8.5E-006	0.0E+000	IGNORE	Y
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.63E-006	0.0E+000	IGNORE	Y
IE-SLOCA	Small LOCA Initiating Event	1.0E-006	1.0E+000	TRUE	Y
IE-TRANS	Transient Initiating Event	5.3E-004	0.0E+000	IGNORE	Y
MFW-SYS-TRIP	Main Feedwater System Trips	2.0E-001			N
MFW-XHE-NOREC	Operator Fails to Recover Main Feedwater	3.4E-001			N
PPR-SRV-CC-PRV1	PORV 1 Fails to Open on Demand	6.3E-003	1.0E+000	TRUE	Y
PPR-SRV-CC-PRV2	PORV 2 Fails to Open on Demand	6.3E-003	1.0E+000	TRUE	Y
RHR-HTX-CF-AB	Failure of Heat Exchangers Due to Common Cause	1.4E-005	1.4E-005		N
RHR-MDP-CF-ALL	RHR Motor Driven Pumps Fails Due to Common Cause	4.5E-004	4.5E-004		N
RHR-XHE-NOREC	Operator Fails to Recover the RHR System	1.0+000	1.0+000		N
RPS-VCF-FO	Reactor Trip System Fails	6.0E-005	6.0E-005		N
RPS-XHE-XM-SCRAM	Operator Fails to Manually Trip the Reactor	3.4E-001	3.4E-001		N

Table C.2.9. Sequence probabilities for Case 1B PORVs unavailable following a small-break

Event tree name	Sequence name	Frequency	% Contribution	Logic
SLOCA	03	5.3E-004	80.4	/RT, /AFW, /HPI, /COOLDOWN, RHR, HPR
SLOCA	20	5.1E-005	7.7	/RT, AFW, MFW, F&B
SLOCA	06	3.6E-005	5.5	/RT, /AFW, HPI
SLOCA	21	2.0E-005	3.0	RT
SLOCA	05	1.8E-005	2.7	/RT, /AFW, /HPI, COOLDOWN, HPR
Total (all sequences)		7.1E-004		

Table C.2.10. System names for Case 1B PORVs unavailable following a small-break LOCA

System name	Description
AFW	No or Insufficient AFW Flow
COOLDOWN	RCS Cooldown to RHR Pressure using TBVs, etc.
F&B	Failure to Provide Feed-and-Bleed Cooling
HPI	No or Insufficient Flow From the HPI System
HPR	No or Insufficient HPR Flow
MFW	Failure of the Main Feedwater System
RHR	No or Insufficient Flow From the RHR System
RT	Reactor Fails to Trip During Transient

Table C.2.11. Conditional cut sets for higher probability sequences for Case 1B

Cut set No.	% Contribution	Frequency	Cut sets
SLOCA Sequence: 3		5.3E-004	
1	83.9	4.5E-004	HPR-XHE-NOREC, RHR-MDP-CF-ALL, RHR-XHE-NOREC
SLOCA Sequence: 20		5.1E-005	
1	48.0	2.4E-005	AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, AFW-TDP-CF-AB
2	37.4	1.9E-005	AFW-XHE-NOREC, AFW-TDP-FC-1B, MFW-SYS-TRIP, MFW-XHE-NOREC, AFW-TDP-FC-1A
SLOCA Sequence: 06		3.6E-005	
1	92.1	3.3E-005	HPI-XHE-NOREC, HPI-MOV-OC-SUC
SLOCA Sequence: 21		2.0E-005	
1	100.0	2.0E-005	RPS-VCF-FO, RPS-XHE-XM-SCRAM
Total (all sequences)		7.1E-004	

Table C.2.12. Selected basic events for Case 2-1
LOOP conditioning sequences 1 and 3

Event name	Description	Base probability	Current probability	Type	Modified for this event
AFW-TDP-FC-1A	Failure of Turbine Driven Pump 1A	3.3E-002	3.3E-002		N
AFW-TDP-FC-1B	Failure of Turbine Driven Pump 1B	3.3E-002	3.3E-002		N
AFW-TDP-CF-AB	Common Cause Failures of Turbine Driven Pumps	1.4E-003	1.4E-003		N
AFW-TNK-FC-PWST	Primary Water Storage Tank Fails	4.1E-005	4.1E-005		N
AFW-XHE-NOREC-L	Operator Fails to Recover AFW System During Blackout	2.6E-001	2.6E-001		N
EPS-DGN-CF-ALL	Common Cause Failure of Diesel Generators	1.3E-003	+0.0E+000	FALSE	Y
EPS-DGN-FC-1A	Diesel Generator A Fails	4.2E-002	+0.0E+000	FALSE	Y
EPS-DGN-FC-1B	Diesel Generator B Fails	4.2E-002	+0.0E+000	FALSE	Y
IE-LOOP	Loss-of-Offsite Power Initiating Event	8.5E-006	1.0E+000	TRUE	Y
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.63E-006	0.0E+000	IGNORE	Y
IE-SLOCA	Small LOCA Initiating Event	1.0E-006	0.0E+000	IGNORE	Y
IE-TRANS	Transient Initiating Event	5.3E-004	0.0E+000	IGNORE	Y

Table C.2.12. Selected basic events for Case 2-1 LOOP conditioning sequences 1 and 3 (cont.)

Event name	Description	Base probability	Current probability	Type	Modified for this event
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 hrs	2.2E-001	+0.0E+000		Y
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6 hrs	6.7E-002	+0.0E+000		Y
PPR-SRV-CC-PRV1	PORV 1 Fails to Open on Demand	6.3E-003	1.0E+000	TRUE	Y
PPR-SRV-CC-PRV2	PORV 2 Fails to Open on Demand	6.3E-003	1.0E+000	TRUE	Y
PPR-SRV-00-PRV1	PORV 1 Fails to Reclose After Opening	3.0E-002	3.0E-002		N
PPR-SRV-00-PRV2	PORV 2 Fails to Reclose After Opening	3.0E-002	3.0E-002		N
PPR-XHE-NOREC-L	Operator Fails to Close Block Valves During Loop	1.1E-002	1.0E+000	TRUE	Y

Table C.2.13. Sequence probabilities for Case 2-1 LOOP conditioning sequences 1 and 3

Event tree name	Sequence name	Frequency	% Contribution	Logic
LOOP	15	7.5E-004	95.5	/RT-L, /EP, AFW-L, /OP-6H, F&B-L
Total (all sequences)		7.9E-004	100.0	

Table C.2.14. System names for Case 2-1 LOOP conditioning sequences 1 and 3

System name	Description
AFW-L	No or Insufficient AFW Flow During LOOP
EP	Failure of Both Trains of Emergency Power
F&B-L	Failure of Feed-and-Bleed Cooling During LOOP
OP-6H	Operator Fails to Recover Offsite Power Within 6 hrs
RT-L	Reactor Fails to Trip During LOOP

Table C.2.15. Conditional cut sets for higher probability sequences for Case 2-1

Cut set No.	% Contribution	Frequency	Cut sets
LOOP Sequence: 15		7.5E-004	
1	48.1	3.6E-004	AFW-XHE-NOREC-L, AFW-TDP-CF-AB
2	37.4	2.8E-004	AFW-TDP-FC-1A, AFW-TDP-FC-1B, AFW-XHE-NOREC-L
Total (all sequences)		7.9E-004	

Table C.2.16. Selected basic events for Case 2-2
LOOP conditioning sequences 2 and 4

Event name	Description	Base probability	Current probability	Type	Modified for this event
AFW-TDP-FC-1A	Failure of Turbine Driven Pump 1A	3.3E-002	3.3E-002		N
AFW-TDP-FC-1B	Failure of Turbine Driven Pump 1B	3.3E-002	3.3E-002		N
AFW-TDP-CF-AB	Common Cause Failures of Turbine Driven Pumps	1.4E-003	1.4E-003		N
AFW-XHE-NOREC-L	Operator Fails to Recover AFW System During Blackout	2.6E-001	2.6E-001		N
EPS-DGN-CF-ALL	Common Cause Failure of Diesel Generators	1.3E-003	+0.0E+000	FALSE	Y
EPS-DGN-FC-1A	Diesel Generator A Fails	4.2E-002	+0.0E+000	FALSE	Y
EPS-DGN-FC-1B	Diesel Generator B Fails	4.2E-002	+0.0E+000	FALSE	Y
HPI-MDP-FC-1A	HPI Motor Driven Pump 1A Fails	3.9E-003	1.0E+000	TRUE	Y
HPI-MDP-FC-1B	HPI Motor Driven Pump 1B Fails	3.9E-003	1.0E+000	TRUE	Y
HPI-XHE-NOREC-L	Operator Fails to Recover the HPI System	8.4E-001	1.0E+000	TRUE	Y
HPR-XHE-NOREC-L	Operator Fails to Recover the HPR System	1.0E-003	1.0E+000		
IE-LOOP	Loss-of-Offsite Power Initiating Event	8.5E-006	1.0E+000	TRUE	Y
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.63E-006	0.0E+000	IGNORE	Y
IE-SLOCA	Small LOCA Initiating Event	1.0E-006	0.0E+000	IGNORE	Y

Table C.2.16. Selected basic events for Case 2-2
LOOP conditioning sequences 2 and 4 (cont.)

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-TRANS	Transient Initiating Event	5.3E-004	0.0E+000	IGNORE	Y
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 hrs	2.2E-001	2.7E-001		Y
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6 hrs	6.7E-002	3.7E-002		Y
PPR-SRV-CC-PRV1	PORV 1 Fails to Open on Demand	6.3E-003	1.0E+000	TRUE	Y
PPR-SRV-CC-PRV2	PORV 2 Fails to Open on Demand	6.3E-003	1.0E+000	TRUE	Y
PPR-SRV-OO-PRV1	PORV 1 Fails to Reclose After Opening	3.0E-002	3.0E-002		N
PPR-SRV-OO-PRV2	PORV 2 Fails to Reclose After Opening	3.0E-002	3.0E-002		N
PPR-XHE-NOREC-L	Operator Fails to CLose Block Valves During LOOP	1.1E-002	1.0E+000	TRUE	Y

Table C.2.17. Sequence probabilities for Case 2-2
LOOP conditioning sequences 2 and 4

Event tree name	Sequence name	Frequency	% Contribution	Logic
LOOP	08	5.9E-002	98.7	/RT-L, /EP, /AFW-L, PORV-L, PRVL-RES, /OP-2H, HPI-L
LOOP	15	7.5E-004	1.2	/RT-L, /EP, AFW-L, /OP-6H, F&B-L
Total (all sequences)		5.9E-002	100.0	

Table C.2.18. System names for Case 2-2
LOOP conditioning sequences 2 and 4

System name	Description
AFW-L	No or Insufficient AFW Flow During LOOP
EP	Failure of Both Trains of Emergency Power
F&B-L	Failure of Feed-and-Bleed Cooling During LOOP
HPI-L	No or Insufficient Flow From the HPI System During LOOP
OP-2H	Operator Fails to Recover Offsite Power Within 2 hrs
OP-6H	Operator Fails to Recover Offsite Power Within 6 hrs
PORV-L	PORVs Open During LOOP
PRVL-RES	PORVs and Block Valves Fail to Reseat (EP Successful)
RT-L	Reactor Fails to Trip During LOOP

Table C.2.19. Conditional cut sets for higher probability sequences for Case 2-2

Cut set No.	% Contribution	Frequency	Cut sets
LOOP Sequence: 08		5.9E-002	
1	50.0	3.0E-002	PPR-SRV-OO-PRV1
2	50.0	3.0E-002	PPR-SRV-OO-PRV2
LOOP Sequence: 15		7.5E-004	
1	48.1	3.6E-004	AFW-XHE-NOREC-L, AFW-TDP-CF-AB
2	37.4	2.8E-004	AFW-TDP-FC-1A, AFW-TDP-FC-1B, AFW-XHE-NOREC-L
Total (all sequences)		5.9E-002	