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ComEd

August 30, 1996

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Attention: Document Control Desk

Subject: Quad Cities Station Units 1 and 2, Response to NRC Review of Individual Plant Examination Submittal - Internal Events
NRC Docket Nos. 50-254 and 50-265

References:
(a) M.J. Vonk Letter to Thomas E. Murley, dated December 13, 1993

(b) J.C. Schrage Letters to NRR, dated October 3, 1994 and December 13, 1994

(c) T.W. Simpkin Letter to NRR, dated September 1, 1995

(d) R.M. Pulsifer Letter to D.L. Farrar, dated November 9, 1995

(e) B. Rybak Letter to NRR, dated December 15, 1995

(f) J. Stephen Perry Letter to NRR, dated March 5, 1996

(g) E.S. Kraft Letter to NRR, dated March 15, 1996

(h) J.B. Hosmer Letter to NRR, dated June 28, 1996

Reference (d) provides the NRC Staff Evaluation Report (SER), including Technical Evaluation Reports, of the Quad Cities Individual Plant Examination (IPE) for internal events. The Staff raised a number of concerns and stated that it "could not reach the conclusion that ComEd has met the intent of Generic Letter 88-20."

In response to similar concerns expressed by the NRC on the Zion Station IPE, ComEd defined a resolution process that would allow ComEd and the NRC to come to closure on the identified issues. In accordance with that process, ComEd developed a "modified IPE" for Zion and Dresden by making changes to the base IPEs. These modified IPEs and additional analyses on other issues raised in the SERs for Zion and Dresden were submitted to the NRC in References (c) and (h).

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In accordance with that same process, ComEd has developed a "modified Quad Cities IPE" to address the Staff's concerns and to allow the Staff to conclude that ComEd meets the intent of Generic Letter 88-20 for Quad Cities.

Specifically, the following changes were made to the base IPE as discussed in References (e) and (f):

1. Including nine (9) new support system based initiators. Specifically, adding:

    a) A Loss of Service Water initiator;
    b) Four (4) initiators for loss of major (4 kV) AC buses;
    c) A Loss of DC Bus initiator;
    d) A Loss of Instrument Air initiator; and
    e) Two (2) initiators for loss of HVAC caused by 1) a loss of a 480 VAC bus or 2) loss of a 480 VAC Motor Control Center (MCC).

2. Modifying the common cause failure (CCF) factors;

3. Revising the success criteria for ATWS to include the Operator Action to inhibit ADS in the ATWS Event Tree;

4. Modifying the human error probabilities (HEPs) for significant operator actions by employing an alternative Human Reliability Analysis (HRA) methodology and changing mission times for standby instruments to reflect a plant-specific assessment of pre-initiator concerns involving instrument calibration;

5. Incorporating updated plant-specific equipment failure rates and unavailabilities for key (risk significant) systems. These updated data are derived from the information provided to the NRC in Reference (g).

A detailed discussion of these modifications to the base Quad Cities IPE is provided in Enclosure 1, including a discussion of the integrated effect on the IPE results. The modified Quad Cities IPE Core Damage Frequency is 2.2E-06/year. With respect to changes in the risk profile:

- The contribution of the Dual Unit Loss of Offsite Power initiator to total core damage frequency increased by about 80% from the base IPE. This increase was due mainly to higher failure probabilities for diesels, HPCI, and RCIC.
- Two new special initiators, Loss of Service Water and Loss of 125 VDC Bus 1B-1, each contribute about 3% of the new total CDF.
- The contribution of the ATWS event doubled due to adding the operator action to inhibit ADS to the event tree.
- The contribution of the General Transient initiator to total CDF increased by a factor of 5 due to human error probability and equipment failure probability changes.

Enclosure 2 provides a discussion of other issues raised in the Staff Evaluation Report (SER) that, upon ComEd evaluation, did not result in modifications to the IPE. Technical justification for the success criteria for HVAC and information on the Peer review performed for the modified IPE are contained in the enclosure.

For your convenience, Enclosure 3 provides revised pages to the base Quad Cities IPE Submittal Report (Reference (a)) that incorporate the IPE modifications discussed above.

In conclusion, ComEd believes that this submittal of the modified Quad Cities IPE is responsive to the comments and concerns raised in the Staff Evaluation Report and associated discussions with the NRC Staff. ComEd also believes that these issues have been resolved by this transmittal. Our expectation is that this modified Quad Cities IPE model will be accepted by the NRC as "meeting the intent of Generic Letter 88-20" without any special restrictions on its potential use in the full scope of PRA applications.

Sincerely,

John Hosmer

John B. Hosmer
Engineering Vice President

Enclosures: (1) Modifications Made to Quad Cities IPE
(2) Response to Other Issues Raised by SER
(3) Revised Pages for Base Quad Cities IPE Submittal Report

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ComEd

Quad Cities

Individual Plant Examination (IPE)

Response to NRC Staff Evaluation Report and Modified Quad Cities IPE

August 1996

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ENCLOSURE 1

MODIFICATIONS MADE TO QUAD CITIES IPE

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MODIFICATION 1	**Support System Based Special Initiating Events**
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MODIFICATION 1: Support System Based Special Initiating Events

The NRC Staff Evaluation Report (SER) raised some concerns related to the screening of potential special initiating events. Specifically, it contains the following statements:

> *The licensee identified and modeled initiating events that include general plant transients, loss-of-coolant accidents, inadvertently open relief valve, interfacing systems loss-of-coolant accident, loss of offsite power at both units and a single unit, and internal flooding. However, the staff could not determine whether sufficient examination of failures from support systems such as instrument air, heating, ventilation, and air conditioning (HVAC), service water, and alternating current (AC) and direct current (DC) power was performed.*

In the original IPE, a number of potential support system based initiators (support system failure events that would result in a scram) were reviewed and screened from further consideration because of the following judgments:

- The response of the plant to the special initiator was judged to be similar to the response modeled for an existing initiator (such as General Transient or single-unit Loss of Offsite Power)

and

- The frequency of the potential initiator was judged to be much lower than the existing initiator

Because special initiators were all eliminated by this screening process, no support system based initiating events were included in the original IPE for Quad Cities.

As noted in the above excerpt from the SER, the staff indicated a concern with screening the following as initiators:

- Loss of AC power
- Loss of DC power
- Loss of instrument air
- Loss of service water, and
- Loss of HVAC

For the Modified IPE, a qualitative evaluation of these potential special initiating events was performed. As a result of the qualitative evaluations detailed below, nine support system based special initiating events were addressed separately from other initiators in the Modified IPE:

Loss of 4KV AC Bus:

Loss of 4 kV Bus 11
Loss of 4 kV Bus 12
Loss of 4 kV Bus 13
Loss of 4 kV Bus 14

Loss of DC:

Loss of 125 VDC Reserve Bus 1B-1

Loss of Instrument Air

Loss of Service Water

Loss of HVAC:

Loss of 480V Bus 18

Loss of 480V MCC 18-2

Loss of AC

A thorough review of the impact of loss of AC power was conducted for the ComEd response to the 1994 NRC RAI on the Quad Cities IPE. The review considered loss of single AC buses or motor control centers (MCCs) as potential initiating events and assumed that the unit was at full power with the normal feedwater/condensate system configuration when the loss of AC occurred. (Single and dual unit loss of offsite power initiating events were already included in the IPE and are judged to be the most likely cause of loss of multiple AC buses. For this reason, loss of multiple AC buses is judged to be bounded by the loss of offsite power events and were not reviewed as special initiating events.)

The impact of loss of single AC buses was reviewed by station personnel at the time of the RAI response to determine whether loss of a single AC bus would lead to a reactor trip. The conclusions of this review were that three cases existed on each unit for which loss of a single AC bus would lead to a reactor trip:

Case 1 Loss of Bus 13 (Unit 1) or Bus 23 (Unit 2)

Loss of 4 kV bus 13/23 would result in a loss of two of the three circulating water pumps for the affected unit, and station procedures would require a manual scram if the bus and one of the lost circulating water pumps could not be restored immediately.

Case 2   Loss of 480 VAC MCC 18-2 (Unit 1) or MCC 28-2 (Unit 2)

Loss of 480 VAC MCC 18-2/28-2 would result in loss of all six outboard MSIV Room coolers for the affected unit. If the MCC and coolers were not restored promptly, loss of cooling in this area would eventually result in a Group I isolation (from the leak detection temperature switches) and subsequent reactor scram.

Case 3   Loss of 480 VAC Bus 18 (Unit 1) or Bus 28 (Unit 2)

Loss of 480 VAC Bus 18/28 would result in loss of 480 VAC MCC 18-2/28-2 and therefore, as discussed in Case 2, would lead to a reactor scram.

The review conducted at the time of the RAI response concluded that loss of any of the other 480 VAC buses or MCCs would not result in a reactor scram.

The review conducted at the time of the RAI response concluded that loss of any one of the remaining 4 kV buses (e.g., Unit 1 Buses 11, 12, 13-1, 14, or 14-1 and Unit 2 Buses 21, 22, 23-1, 24, or 24-2) would not cause a reactor scram. Buses 13-1/23-1 and 14-1/24-1 have no normally operating equipment loaded to them that would lead to a reactor scram. The review considered that loss of one of 4 kV Buses 11/21, 12/22, and 14/24 would cause loss of one or more condensate or feedwater pumps, and would also cause loss of one of the unit's reactor recirculation pumps. The rapid drop in reactor power due to the drop in reactor recirculation flow would largely mitigate the drop in feedwater flow. However, since that time, the issue has been reconsidered, and it appears prudent to conclude that a scram from such a severe power and feedwater transient could occur and this has some non-zero probability. For this reason, the loss of Buses 11/21, 12/22, and 14/24 were also included as special initiators in the Modified IPE. These are outlined below.

<u>Case 4</u> Loss of Bus 11 (Unit 1) or Bus 21 (Unit 2)

The loss of Bus 11/21 would result in the loss of one reactor recirculation pump and one reactor feedwater pump which are directly fed from the bus. The drop in feedwater flow to the reactor would be offset by a rapid drop in reactor power due to the drop in recirculation flow. In most cases, the flow drops are expected to offset each other, and the reactor would continue to run at reduced power. However, given the severity of this transient, it is reasonable to expect that a reactor scram could result with a non-zero probability.

<u>Case 5</u> Loss of Bus 12 (Unit 1) or Bus 22 (Unit 2)

The loss of Bus 12/22 would result in the loss of one reactor recirculation pump and one reactor feedwater pump which are directly fed from the bus, a situation identical to the loss of Bus 11/21 described above.

<u>Case 6</u> Loss of Bus 14 (Unit 1) or Bus 24 (Unit 2)

The loss of Bus 14/24 would result in the loss of one reactor recirculation pump and either one or two condensate/condensate booster pumps, depending on the initial lineup. The condensate pump trip would likely cause both reactor feedwater pumps to trip on low suction pressure. However, the standby reactor feedwater pump would start. The recirculation pump is lost due to a loss of oil pressure to the recirculation motor-generator set. As with the loss of buses 11 or 12, the drop in feedwater flow to the reactor would be offset by a rapid decrease in reactor power due to the drop in recirculation flow. In most cases, the flow drops are expected to offset each other, and the reactor would continue to run at reduced power. However, given the severity of this transient, it is reasonable to expect that a reactor scram could result with a non-zero probability.

A related issue that was addressed in the ComEd response to the RAI dealt with potential loss of electrical switchgear because of water spray from nearby water bearing lines. Walkdowns were performed to support a response to this concern, and one bus and one MCC that could cause a reactor trip if deenergized were found to be vulnerable to water spray. For the Modified IPE, further walkdowns were conducted to evaluate the potential loss of three additional buses per unit. A failure rate was calculated for each of these pieces of switchgear based on the geometry of the potential spray source, based on component leakage rates, and based on the assumption that a water spray on the switchgear would cause it to deenergize.

The initiator frequencies for loss of AC events were estimated from a fault tree probability that was extrapolated over a one-year time frame. Appropriate contributions from water spray were added to the fault tree-derived frequencies. Where the water spray contributions differed between Unit One and Unit Two, the higher of the two values was used. The third and final factor in the loss of AC event frequency estimates was a probability of scram factor on loss of the MCC or bus. As was discussed above, for the loss of Bus 13, loss of Bus 18, and loss of MCC 18-2, the probability of scram was 1.0. For a loss of the other 4KV buses, the probability of scram was estimated to be 0.1 due to the power and flow transient. This estimate is based on engineering judgment and plant experience. The initiating event frequencies and CDF contributions are shown in Table 1-1. The highest importance loss of AC special initiator, loss of Bus 13, contributed 0.24% of the total CDF in the Modified IPE. None of the other loss of AC special initiators contributed more than 0.1 percent.

Loss of DC

The loss of either division of DC power at 250 VDC, 125 VDC, and 24/48 VDC were examined for situations that would lead to a reactor scram. This investigation was conducted by review of normal and abnormal operating procedures, by review of electrical key diagrams and schematics, and by discussions with Operating and System Engineering personnel. One situation was identified that could lead to a scram from a loss of one division of 125 VDC.

> Loss of either division of 125 VDC power will result in one of the two recirculation pumps coasting down because of a loss of oil pressure to the motor-generator (M-G) set. One of the two normally running reactor feedwater pumps will receive a trip signal on a false indication of low lubricating oil pressure (the logic is deenergize to trip). However, for two of the three possible combinations of running reactor feedwater pumps for full power operations, the pump cannot trip because there is no 125 VDC logic power available to energize the trip coil on the pump's circuit breaker. The final result is a relatively mild decrease in reactor power transient due to the loss of one recirculation pump.

The result from the third of the three possible combinations of running reactor feedwater pumps is different. The "C" reactor feedwater pump can be fed from either Bus 11, which uses Division I 125 VDC control power, or Bus 12, which uses Division II 125 VDC control power. However, the low lubricating oil pressure trip logic for the "C" pump is always fed from the Division II Reserve 125 VDC Bus 1B-1. When the "C" pump is used, it is always powered from the 4KV bus not supplying power to the other feedwater pump in use.

When the "C" reactor feedwater pump is powered from Bus 11, the circuit breaker trip coil is fed from Division I Main 125 VDC Bus 1A-2. A loss of the reserve (Division II) 125 VDC supply will result in a trip signal to the "C" pump on a false low lubricating oil pressure signal, and the pump will trip because the circuit breaker trip coil still has power. The result is a loss of the "C" reactor feedwater pump and the "B" recirculation pump. This is a situation similar to the loss of 4KV Bus 11 or 12, and could also lead to a reactor scram with a non-zero probability due to the severity of power and flow transient.

The initiator frequency for the loss of 125 VDC Reserve Bus 1B-1 event was estimated from the original IPE fault tree probability for loss of this bus by extrapolating the probability over a one-year time frame. Water spray was not a factor because there are no water lines in the vicinity of this bus. For a loss of the 125 VDC Reserve Bus, the probability of scram was estimated to be 0.1 due to the power and flow transient from loss of one recirculation pump M-G set and the "C" reactor feedwater pump. From operating records for the last several cycles on each unit, it was estimated that the "C" reactor feedwater pump would be in operation from the Division I AC bus (the conditions for a trip to occur on loss of 125 VDC) 33% of the time. The resulting initiating event frequency is

1.01E-03/yr as shown in Table 1-1. The CDF attributable to a loss of 125 VDC Bus 1B-1 is 3.43% of the total CDF in the Modified IPE.

Loss of Service Water

Loss of service water would result in loss of cooling for numerous components and would lead to a turbine trip and reactor scram. Some of the vital equipment cooled by service water is:

- Generator stator water coolers
- Generator hydrogen coolers
- Instrument air compressors
- Recirculation Pump M-G set oil coolers
- Turbine oil coolers
- Reactor building closed cooling water (RBCCW) heat exchangers
- Turbine building closed cooling water (TBCCW) heat exchangers
- MSIV room coolers

Although the loss of service water would lead to a reactor trip very quickly due to the loss of stator cooling, the resulting event resembles a general transient without feedwater or the condenser being available. The initiating event frequency was made up of three separate components. The first part was based on the fault tree probability that was extrapolated over a one year time frame. The second component was due to a spontaneous rupture of a large diameter service water pipe. The third component was due to a loss of instrument air during the three week time period that could lead to a rapid clogging of the service water strainers (see discussion in the following section). The frequency for this third component was estimated by multiplying the loss of instrument air annual frequency by the fraction 3/52. The resulting total frequency is

9.10E-03/yr. The CDF attributable to a loss of service water was nearly 3% of the total CDF in the Modified IPE. These values are shown in Table 1-1.

Loss of Instrument Air

Loss of instrument air would result in a Group I isolation and reactor scram. Under most circumstances, the response of the unit would be the same as a general transient with the main condenser unavailable. A detailed review of dependencies identified for the IPE, however, revealed that operation of service water strainers depends on instrument air and that during rare occasions (i.e., during "shad" runs on the Mississippi River which can occur during about three weeks out of the year), loss of the service water strainers would lead to loss of service water quite quickly. During this three week period, the loss of instrument air is considered to be a loss of service water initiator.

The loss of instrument air initiator frequency was estimated from the fault tree probability by extrapolating it over a one year time frame. This frequency was then reduced by the factor 49/52, the fraction of time that the loss of instrument air would not also lead to a loss of service water because of rapid strainer clogging. The resultant loss of instrument air initiator frequency is 1.43E-02/yr. The CDF attributable to a loss of instrument air was less than 0.2% of the total CDF for the Modified IPE. These values are shown in Table 1-1.

Loss of HVAC

Loss of HVAC in the important plant areas was reviewed and confirmed previous conclusions that the only area of concern as a special initiating event was the outboard MSIV room. Loss of the six coolers for this area (as discussed above for the loss of 480 VAC MCC 18-2/28-2) would lead to a Group I isolation from high area temperature and reactor scram. Because these coolers normally run when the unit is at power and because there are several coolers, the most likely cause for loss of cooling to this area is judged to be loss of AC power to the coolers. All of the MSIV room coolers for a unit are powered from the same MCC: MCC 18-2 for Unit 1 and MCC 28-2 for Unit 2. Therefore, the scenario of a scram due to loss of outboard MSIV room coolers is judged to be addressed adequately by the loss of Bus 18 and loss of MCC 18-2 special initiating events.

Other plant areas considered but discounted with respect to potential loss of HVAC special initiating events are as follows:

AC Electrical Switchgear

At some plants, loss of cooling to electrical switchgear rooms is a concern. At Quad Cities, however, the large AC switchgear (and most DC switchgear) is located in open areas of the turbine or reactor building. These areas are normally supplied with ventilation but do not have specific area coolers. Operating experience has shown that loss of ventilation in the open areas of the turbine building during warm times of the year is easily mitigated by opening outer doors of the building, especially the large trackway doors. Extended loss of ventilation to the reactor building could result in a controlled unit shutdown due to equipment qualification concerns, but the concerns would not be expected to lead to a reactor scram.

DC Electrical Equipment

Some important DC electrical equipment (battery chargers, main battery buses, and turbine building distribution panels) is located in compartments below the battery rooms which are supplied with ventilation air from the east turbine building supply fans. Loss of HVAC to these compartments could potentially lead to overheating of equipment, but would require loss of the supply fans which is annunciated in the Control Room. Furthermore, high temperatures in these DC electrical equipment compartments would be identified during once-per-shift operator rounds or, in extreme cases, by equipment alarms. Overheating could be addressed by opening the compartment doors to the turbine building and positioning temporary fans, as necessary. Therefore, a loss of HVAC to these rooms is not expected to result in a reactor scram.

Battery Rooms

The battery rooms have their own self-contained HVAC units. The battery rooms are checked once per shift and their HVAC units are checked daily on operator rounds. Loss of HVAC for a battery room would result in a slow temperature transient and can impact the operability of the battery. If temporary cooling or heating could not be established, the result could be a controlled shutdown of a unit. Such an event would not be expected to lead to a reactor scram.

Main Control Room and Auxiliary Electric Equipment Room

The Main Control Room and the Auxiliary Electrical Equipment Room are normally cooled by the "A" train of HVAC. If the "A" train fails, a redundant "B" HVAC train will automatically start and take on the cooling loads in these two areas. Complete loss of cooling to these areas can be addressed by opening doors to the service building and use of temporary fans, as necessary. These actions are proceduralized and the equipment is pre-staged as a part of the Station's response to station blackout requirements. Loss of HVAC to these two areas would therefore not be expected to lead to a reactor scram.

Feedwater Pumps and Recirculation Pump M-G Sets

The reactor feedwater pump motors and reactor recirculation pump motor-generator (M-G) sets are supplied with ventilation. Loss of ventilation to a single reactor feedwater pump motor or a single recirculation pump M-G set would not be expected to lead to a reactor scram. In the unlikely occurrence of concurrent loss of ventilation to multiple pumps or both of a unit's reactor recirculation M-G sets, a scram could result. The unit's response in such a circumstance would be that modeled for the general transient initiating event, however, and the frequency of such an event is judged to be much less than the general transient initiating event frequency used in the IPE. For these reasons, a special initiating event of loss of ventilation to the reactor feedwater pump motors or the reactor recirculation M-G sets is not warranted.

Emergency Diesel Generator Rooms

Ventilation of the diesel generator rooms is necessary during warm times of the year due to equipment panels in the rooms. Overheating could potentially make a diesel generator inoperable but would not lead to a reactor scram.

Safe Shutdown Makeup Pump Room

The Safe Shutdown Makeup Pump (SSMP) room has a separate cooling system to cool the pump and SSMP electrical distribution system. Failure of the cooling system could threaten pump operability in the event of extended operation with the room doors closed but would not cause a reactor trip.

For these reasons, loss of HVAC for all areas except the outboard MSIV room was screened out from further consideration as a special initiator for the

Modified IPE. As discussed above, loss of the outboard MSIV room coolers is addressed by the loss of Bus 18 and loss of MCC 18-2 special initiating events.

TABLE 1-1
QUANTITATIVE RESULTS FOR SPECIAL INITIATING EVENTS

Initiating Event	Initiating Event Frequency	Core Damage Frequency (CDF) Contribution	Percent of Total CDF*
General Transient	3.87E+00/yr	2.34E-07/yr	10.80%
Loss of Bus 11	2.75E-04/yr	3.30E-10/yr	0.02%
Loss of Bus 12	2.05E-04/yr	3.07E-10/yr	0.01%
Loss of Bus 13	4.65E-03/yr	5.19E-09/yr	0.24%
Loss of Bus 14	5.88E-04/yr	8.15E-10/yr	0.04%
Loss of Bus 18	4.18E-04/yr	6.79E-10/yr	0.03%
Loss of MCC 18-2	1.34E-03/yr	4.09E-10/yr	0.02%
Loss of 125 VDC Bus 1B-1	1.01E-03/yr	7.44E-08/yr	3.43%
Loss of Service Water	9.10E-03/yr	6.49E-08/yr	2.99%
Loss of Instrument Air	1.43E-02/yr	4.03E-09/yr	0.19%

The General Transient Event is shown for comparison purposes.
* Total CDF for the Quad Cities Modified IPE is 2.2E-06/yr.

MODIFICATION 2: Common Cause Failures

The NRC's Staff Evaluation Report (SER) raised concerns regarding the treatment of common cause factors. Specifically, page 3 of the SER states that:

> *Regarding the common cause failure (CCF) analysis, the factors used for CCF quantification are significantly lower than generic boiling water reactor CCF factors. Several beta factors are factors of 3-to-20 lower than typically seen for comparable components (e.g., motor operated valves, diesel generators, pumps). Further, based on the findings of the Quad Cities diagnostic evaluation team, the staff believes that the beta factors may have been underestimated for some components (e.g., motor-operated valves and ERVs). The licensee did not provide a sufficient basis for the use of these low beta factors.*

"Common cause" describes multiple failures of functionally identical components due to a single, shared cause. Common Cause Analysis (CCA) evaluates the effects of these dependencies that may affect the ability of a system to prevent or mitigate a severe accident.

The Quad Cities CCA modeled common cause failures at the basic event level, employing the Multiple Greek Letter (MGL) method as defined in NUREG/CR-4780, "Procedures for Treating Common Cause Failure in Safety and Reliability Studies." The MGL method uses the parameters beta (β), gamma (γ), and delta (δ), defined as follows:

β conditional probability that the common cause of a component failure will be shared by one or more additional components

γ conditional probability that a common cause failure of two components will be shared by one or more additional components

δ conditional probability that a common cause failure of three components will be shared by one or more additional components

The MGL method allows one to continue defining parameters as far as desired. Consideration of more than three parameters generally reduces conservatism but can become unwieldy. The Quad Cities CCA assumed that common cause groups with greater than four components were adequately represented by the three-parameter analysis without being overly conservative.

The evaluation of Quad Cities failure data (January 1985 through December 1991) indicated that there had been no common cause events at the Quad Cities site applicable to current maintenance and operating practices. As a result, to more realistically model current experience at Quad Cities, a Quad Cities-specific evaluation of common cause failure events was performed. Quad Cities-specific common cause parameters were developed for components that had data available, including the following:

- Circuit Breakers
- Check Valves
- Diesel Generators
- Motor-operated Valves
- Fans
- Low-Head Pumps (Core Spray)
- HVAC Chillers
- Relief Valves
- Pumps (Fire Diesel, SLC, CRD, DG Cooling Water, FW)

A generic common cause failure database was developed from EPRI NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," supplemented with events from the September 1990 EPRI draft report, "A Database of Common Cause Events for Risk and Reliability Evaluations."

A four-member expert judgment panel reviewed data from the generic common cause failure database for applicability to ComEd plants for the original IPE. The Individual Plant Evaluation Partnership (IPEP) and ComEd each provided two members to the panel, one representing the Quad Cities IPE and one representing the Byron IPE. The members from IPEP were common cause data specialists and the members from ComEd were plant experts. The Quad Cities plant expert had previously worked as an SRO and was familiar with current plant practices and procedures as well as those in practice at the time of the events in the generic common cause failure database.

The common cause failure database was reviewed for events applicable to Quad Cities. The expert panel came to a consensus opinion on each generic common cause event's applicability to Quad Cities based upon current Quad Cities system configuration and maintenance and operating practices. The detailed analysis of the common cause events is contained in the following document:

> Westinghouse Calculation Note CN-COA-92-470-RO, "Common Cause MGL Factors for Braidwood, Byron, LaSalle, and Quad Cities IPEs," Westinghouse Proprietary Class 2C, 1993.

Examples (not exhaustive) of reasons for expert panel screening of events from the common cause databases discussed above as being not applicable to Quad Cities included the following:

- Single failure events with little to no evidence that a common cause failure existed.
- Multiple failure events due to independent causes rather than common causes.
- Events that would be prevented by procedures in place at Quad Cities.
- Events with significant time periods between failures.
- Events that could not have occurred at Quad Cities due to differences in equipment configuration or condition.
- Events with failures that would have been prevented by post-maintenance testing.
- Events that do not have a significant impact in analyses for PRA applications.
- Events involving easily recognized and recoverable failures.
- Events having failure modes that are not applicable to success criteria for Quad Cities.
- Events occurring prior to commercial operation, detected during startup testing, and not applicable to a mature plant.
- Events involving known common cause mechanisms addressed by specific programs in place at Quad Cities.

In response to the concern expressed in the SER with several beta factors being "lower than typically seen," ComEd committed in the 12/15/95 response to the SER to impose a minimum value of 1.0E-02 for beta factors in all categories. Three beta factors were adjusted as a result of imposing this minimum value; two for diesel generator components and one for pumps (Fire Diesel, SLC, and CRD). Table 2-1 shows the resulting revised MGL parameters used in the Modified Quad Cities IPE.

An average common cause component group was quantified from a composite of all the common cause failures for all components in the database ("ALL" in

Table 2-1). Use of the parameters calculated for this average common cause group was extended to components that have no history of common cause failure, but judged by the analyst to have some potential for common cause failure. The common cause contribution for the following components was calculated using the average MGL values:

- Relays, including contacts and coils
- Switches, including temperature, level, and pressure switches
- Dampers
- Explosive valves
- Solenoid-operated valves
- Strainers and filters
- Stop check valves
- Timing relays

In general, the components included in this list were judged to be less complex than the components in the database and thought to have less potential for common cause failure mechanisms. Therefore, assignment of the average common cause parameters is judged to be realistic.

The "All" category of MGL factors was also applied to pumps that have four components in the system. Pumps in this category are RHR, RHR Service Water, Condensate, and non-safety related Service Water.

The same equations discussed in Section 4.4.3 of the original IPE Submittal Report were used to calculate the MGL factors for the Modified Quad Cities IPE. These are shown in Table 2-2. Table 2-3 shows the revised component-specific MGL factors that were derived from the revised MGL parameters given in Table 2-1.

In the Front End Analysis Technical Evaluation Report (TER) attached to the SER, Table II-3 compared some β factors from the original Quad Cities IPE Submittal Report to β factors from several other sources. Due to a lack of clarity in the referenced table (Table 4.4.3-1) in the original IPE Submittal, the analyst appears to have misinterpreted the β factors for several pieces of equipment. For example, the TER lists the β factor for RHR pumps and service water pumps as 0.009, when in fact the value used in the original (and Modified) Quad Cities IPEs was 0.021. For Core Spray Pumps, the value in the TER table was shown as 0.009, but the value actually used was 0.081. Table 2-3, below, specifies which MGL factors were applied to specific pumps to eliminate this ambiguity.

Changes to the MGL factors as shown in Table 2-3 resulted in changes to 14 common cause basic events in the Modified IPE. The affected systems were diesel generators, SLC pumps, CRD pumps, and diesel fire pumps.

In addition, changes to the failure probabilities of several pieces of equipment as described in the writeup for Modification 5 also resulted in adjusting 70 associated common cause basic event failure probabilities. Also, eight of the above 14 common cause basic events were affected by both an MGL factor change and a change to the associated failure probability. In summary, a total of 84 common cause failure basic event probabilities were adjusted for the Modified IPE.

TABLE 2-1
MGL PARAMETERS FOR QUAD CITIES MODIFIED IPE
From Table 4.4.3-1 in Original IPE Submittal

CC Component Group	Four Component System			Three-Component System		Two-Component System
	β	γ	δ	β	γ	β
Circuit Breakers	7.2E-02	9.9E-01	8.6E-01	5.6E-02	9.3E-01	3.9E-02
Check Valves	4.8E-02	6.8E-01	3.7E-01	4.3E-02	4.4E-01	3.1E-02
Diesel Generators	(a)	(a)	(a)	1.0E-02*	(b)	1.0E-02*
MOV	1.9E-02	5.4E-01	5.2E-01	1.6E-02	4.3E-01	1.1E-02
Fans	5.8E-02	(b)	7.8E-01	5.8E-02	8.5E-01	5.4E-02
Low-Head Pumps	(a)	(a)	(a)	(a)	(a)	8.1E-02
HVAC Chiller	(a)	(a)	(a)	2.4E-02	1.3E-01	1.3E-02
Safety-Relief Valves	3.2E-01	5.4E-01	(b)	2.7E-01	6.4E-01	2.2E-01
Pumps	(a)	(a)	(a)	1.3E-02	3.5E-01	1.0E-02*
"ALL"	2.1E-02	6.8E-01	6.4E-01	1.9E-02	5.8E-01	1.5E-02

Notes: (a) MGL values not calculated for this size system.

(b) Value not calculated. The average value for all component failures ("ALL") given in this table is used as the value for these components.

* New *minimum* value.

TABLE 2-2
EQUATIONS FOR GENERATING MGL FACTORS FROM β, γ, δ
From Table 4.4.3-2 in Original IPE Submittal

COMPONENTS IN SYSTEM (m)	NUMBER OF FAILED COMPONENTS (k)		
	2	3	4
2	β	-	-
3	$1/2 \times \beta$	$\beta \times \gamma$	-
4	$1/3 \times \beta$	$1/3 \times \beta \times \gamma$	$\beta \times \gamma \times \delta$

TABLE 2-3
MGL FACTORS FOR QUAD CITIES MODIFIED IPE
From Table 4.4.3-3 in Original IPE Submittal

COMPONENT - TYPE [Application]	COMPONENTS IN SYSTEM (m)	NUMBER OF FAILED COMPONENTS (k)		
		2	3	4
Circuit Breakers - CB	2	3.90E-02	-	-
	3	2.80E-02	5.21E-02	-
	4	2.40E-02	2.38E-02	6.13E-02
Check Valves - CV	2	3.10E-02	-	-
	3	2.15E-02	1.89E-02	-
	4	1.60E-02	1.09E-02	1.21E-02
Diesel Generators - DG	2	1.00E-02*	-	-
	3	5.00E-03*	5.80E-03*	-
Motor Operated Valves - MV	2	1.10E-02	-	-
	3	8.00E-03	6.88E-03	-
	4	6.33E-03	3.42E-03	5.34E-03
Fans - FN	2	5.40E-02	-	-
	3	2.90E-02	4.93E-02	-
	4	1.93E-02	1.31E-02	3.08E-02
Low Head Pumps - PM [1]	2	8.10E-02	-	-
HVAC Chiller - RF	2	1.30E-02	-	-
	3	1.20E-02	3.12E-03	-
Relief Valves - AM, AS, AV	2	2.20E-01	-	-
	3	1.35E-01	1.73E-01	-
[2]	4	1.07E-01	5.76E-02	1.11E-01
Pumps - PD, PM [3]	2	1.00E-02*	-	-
[4]	3	6.50E-03	4.55E-03	-
All	2	1.50E-02	-	-
	3	9.50E-03	1.10E-02	-
[5]	4	7.00E-03	4.76E-03	9.14E-03

Notes: * Values modified as a result of applying a minimum β factor of 1.0E-02
[1] Applied to Core Spray Pumps
[2] Applied to Electromatic Main Steam Relief Valves
[3] Applied to SLC, CRD, and Fire Diesel Pumps
[4] Applied to DG Cooling Water and Feedwater Pumps
[5] Applied to RHR, RHR Service Water, Condensate, and non-safety related Service Water Pumps, in addition to other components

MODIFICATION 3 HVAC and ATWS Success Criteria

The NRC's Staff Evaluation Report (SER) raised concerns regarding the treatment of specific success criteria. Specifically, page 4 of the SER states:

> *The licensee utilized realistic plant-specific analyses based on material access authorization program [sic] calculations for establishing plant-specific success criteria for each initiator group. Based on the licensee's responses to the RAI, the staff concludes that the Quad Cities IPE success criteria are generally reasonable. The staff, however, believes that additional technical basis is required to support the specific success criteria: room cooling is not required for the electrical components and for the control room and the operators will be 100 percent successful to inhibit automatic depressurization in response to an anticipated transient without scram (action assumed to take place, but not modeled).*

Electrical System Success Dependence on HVAC

No changes were adopted in the modeling of HVAC for system success criteria in the Modified IPE. A discussion of the treatment of HVAC is given in Enclosure 2, "Response to Other Issues Raised by the SER." This discussion provides a basis for concluding that HVAC is not required for electrical equipment areas and the Control Room.

Operator Action to Inhibit ADS

The operator action to inhibit ADS was explicitly modeled in the original IPE for the Small and Medium LOCA events. The assumption in the original IPE that operator action would be 100% successful in inhibiting automatic depressurization for the other events was reviewed in response to the concern in the SER. The assumption was found to be conservative for all other initiating events except ATWS. Therefore, the ATWS PRT was revised for the Modified

Quad Cities IPE to include a node for this operator action, and appropriate Human Error Probability (HEP) estimates were developed.

As discussed later in the writeup for Modification 6, the main impact of including operator action to inhibit ADS for ATWS was to increase the CDF contribution of the ATWS initiator.

MODIFICATION 4 Revised Human Error Probabilities and Treatment of Pre-Initiator Events

The NRC's Staff Evaluation Report (SER) raised concerns regarding the treatment of human reliability analysis (HRA). Specifically, pages 6 through 8 of the SER state that:

> *The licensee's HRA focused on actions needed to mitigate an accident (i.e., post-initiator human events). The submittal and the licensee's responses to the staff's RAI did not provide sufficient evidence that human events were adequately and appropriately treated. The staff has several concerns regarding the licensee's HRA.*
>
> a. *The licensee's diagnosis model does not appear to be in agreement with the HRA method/technique utilized (i.e., NUREG/CR-1278). "Diagnosis" in NUREG/CR-1278 includes the actions to "perceive, discriminate, interpret, diagnose" an event and the operators "first-level of decision making." While using symptom-based emergency operating procedures removes the need to identify the type of accident, such as a loss-of-coolant accident, their use does not remove the need for other aspects of diagnosis. It appears, however, that only "detection" was modeled and no basis was provided as to why other diagnostic tasks were excluded. Diagnosis is an important contributor to human error. In the EPRI-sponsored Operator Error Experiment program, 70 percent of the errors and near misses by the operating crew observed in simulator experiments were categorized as errors "in information processing and decision making."*
>
> b. *The licensee's method for addressing the influence of the accident progression on human performance and the dependencies between human actions does not appear to be in agreement with the HRA method/technique utilized (i.e., NUREG/CR-1278). The licensee used combinations of (the same) three performance shaping factors (PSF) for all human actions for all accident conditions. It was not explained why these three PSFs are adequate to account for the specifics of human performance under all accident conditions, nor why a particular PSF combination applies to*

a particular human action in a particular accident sequence. Also, explicit dependencies between separate human actions on the event trees were not provided and there is insufficient detail to understand whether these dependencies were appropriately modeled. Therefore, the staff could not evaluate whether the licensee appropriately addressed the influences of different accident conditions on human performance.

c. *The licensee's consideration of time does not appear to be in agreement with the HRA method/technique utilized. The licensee appeared to have only considered a "slack time" which the licensee defined as "the amount of time available to the operator over and above that necessary to diagnose and perform the action." The staff has two major concerns:*

- *It is not apparent what the basis was for the calculation of the "slack time"; whether it was based on "real time measures" or on analysts' assumptions.*

- *The calculation of "slack time" does not appear to consider the time needed to perform an action versus the time available to perform the action.*

d. *The licensee's treatment of plant-specific performance shaping factors does not appear to be adequately justified. General conclusions appear to have been assumed regarding such items as training, communications, supervision, and procedures that resulted in reducing the human error probabilities. It is not apparent that plant-specific experience or history (e.g., detailed control room reviews, NRC or INPO training audits, NRC SALP reports or other reviews of plant operating history) was sufficiently considered and, therefore, the human error probabilities appear to be artificially derived.*

e. *The licensee did not provide a discussion in the submittal of human events associated with normal operations that may leave a system in an undetected disabled condition (pre-initiator human events). However, material provided by the licensee in response to the staff's RAI indicates that six pre-initiator human events were quantified and included in the IPE model. Their quantification was based on the HRA method described in NUREG/CR-1278. Generic human error probabilities were obtained from appropriate*

NUREG/CR-1278 tables, but were not modified to reflect plant-specific factors and experience. In addition, none of these events are associated with calibration activities. The staff concludes that the IPE's treatment of pre-initiator human events is limited in scope and level of detail. In particular, the staff believes that calibration errors were dismissed from consideration without a rigorous plant-specific assessment. NUREG-1335 (Section 2.1.5) requested licensees to examine "human failures in maintenance and operations" which includes equipment calibration. Many IPEs/PRAs have identified restoration and calibration events as important contributing events to core damage.

The concerns (a through e) were reviewed and resulted in changes for the Modified Quad Cities IPE. These changes are discussed below.

Introduction

A commitment was made to the NRC to revise the Dresden and Quad Cities PRA models. As part of these revisions, human error probabilities were recalculated using a newer technique, the EPRI CBDTM (Cause Based Decision Tree Methodology) from EPRI TR-100259. The method used to determine the HEPs for the Quad Cities IPE modification is very similar to the method used for the Zion IPE modification. Summaries of the Zion Modified IPE and the Zion HRA notebook were submitted for NRC review. For the Quad Cities IPE modification, complete evaluations were performed for only those operator actions that were deemed to be significant (see the Modified Human Reliability Analysis section below for discussion of the significant actions).

Actions taken to address concerns a through d are summarized below. These actions resulted in changes to the Human Reliability Analysis results and thus changes in the overall CDF. A review and analysis of pre-initiators was completed to address concern e. This additional analysis, also resulted in changes for the Modified Quad Cities IPE.

Concern a: Diagnosis Model

The HRA portion of the modified PRA was revised to address concerns related to both detection and diagnosis. The detection and diagnosis analysis methodology used is the EPRI CBDTM (Cause Based Decision Tree Methodology) described in EPRI TR-100259. This method was used to determine the cognitive (Pc) portion of the operator action. The EPRI CBDTM considers such factors as data availability, attention failure, miscommunication and misreading of data, misleading information, missing or misreading procedure steps, misinterpretation of instructions or decision logic, and deliberate violations.

Concern b: Influence of Accident Progression of Human Performance

During the revision of the HRA, each operator action was evaluated on a sequence by sequence basis to determine the appropriate HEP (Human Error Probability) considering plant conditions, dependency on previously failed operator actions, time available to perform the action, stress levels and opportunities for recovery.

Concern c: Consideration of Time

The use of "slack time" recovery was not used in the revised HRA. Only those recovery opportunities that have been specifically identified in the sequence by sequence evaluation of each operator action have been credited. The time consideration during these sequence by sequence evaluations first determined the feasibility of the action, and secondly assessed the feasibility of recovery given an initial error. When the time is limited, the time pressure was also taken into account by increasing the stress factor (addressed within NUREG/CR-1278) in the evaluation of the basic HEP.

The times available for the action were determined from MAAP (Modular Accident Analysis Program) runs which established the times for the cues and the time for action completion. The time required for the detection, analysis, diagnosis, decision and action was determined by discussions with operators and operator trainers, observation of simulator runs, and judgment by the HRA analyst.

Concern d: Treatment of Plant-Specific Shaping Factors

A review of the plant operating history and discussions with the plant training and operating personnel did not identify any deficiencies in control room design, procedure format or use, or operator training/experience. It was assumed based upon the detailed control room reviews, NRC and INPO audits of training, and various other initiatives, that the control room design, training, experience, etc. were nominal. Therefore, the nominal (mean) values for the various EPRI decision trees and NUREG tables were used in the revised HRA analysis.

Modified Human Reliability Analysis

For the modified HRA, only those actions which had a Risk Achievement Worth of greater than 2.5 and those actions which were added as a result of changes to the Plant Response Trees and Fault Trees, received a complete evaluation utilizing the EPRI CBDTM technique. The remaining HEPs (with RAWs less than 2.5) were reviewed for reasonableness, including consideration of dependencies and stress factors. The modified Human Reliability Analysis modeled the current (December 1995) Quad Cities procedures.

The EPRI CBDTM technique provided values for two operator actions, OHX (Operator action to align cooling to RHR) and OSPC (Operator action to initiate suppression pool cooling), which appeared overly conservative. The OHX contains two pump starts and two valve manipulations. The action OSPC contains a single valve manipulation. These actions are practiced simultaneously during simulator exercises and multiple times during the year to keep the suppression pool cooled. Also, these actions are directed to be performed early in the events (5-10 minutes), but the need does not arise until much later (5-24 hours) into the event. There are multiple indications and alarms on suppression temperature and containment pressure so there are multiple opportunities to recover the actions. The CBDTM technique does not adequately treat actions which can be delayed for hours and have multiple recovery opportunities. After reviewing similar actions from other IPEs and discussions with the operators and instructors, a value of 1.0E-5 (for both OHX & OSPC) was assumed for all initiating events except for ATWS, SBO and Loss of DC. For these three initiators a value of 1.0E-4 (for both OHX & OSPC) was assumed to properly reflect the added stress and the possibly shortened response times.

All of the HEPs after the re-evaluation have values above 1.0E-4, except for OHX and OSPC as discussed above. As can be seen in the summary of the HEP evaluations (Tables 4-1 and 4-2 below), some of the actions changed significantly, some increasing and some decreasing. Table 4-1 lists the Quad Cities Fault Tree HRA Results and Table 4-2 lists the Quad Cities Plant Response Tree HRA Results. Listed for each operator action are the node case name, the description, the Risk Achievement Worth, the original HEP value, the revised HEP value and the percent difference in the HEP. (Note: after each set of operator action cases there is a summary line that gives the combined Risk Achievement Worth for all the cases.)

No new vulnerabilities were identified during the Human Reliability Analysis.

Table 4-1
QUAD CITIES FAULT TREE HRA RESULTS

NAME	DESCRIPTION	RAW*	Orig. HEP	Rev. HEP	%CHG
1ACBS13-1-13-H--	Manually Transfer Power to Bus 13	1.450	5.3E-3	1.2E-3	-77.2%
1ACBS14-1-14-H--	Manually Transfer Power to Bus 14	4.800	5.3E-3	1.2E-3	-77.2%
1ACBS14-124-1H--	Energize Bus 14-1 from Cross Tie Bus	1.450	1.0E-2	1.0E-2	
1ACBS18-19---H--	Energize Bus 18 from 19		1.0E-2	1.0E-2	
1ACBS19------H--	Energize Bus 19 from 18		1.0E-2	1.0E-2	
1ACCB1516----H--	Energize Bus 15 from 16		5.3E-3	5.3E-3	
1ACCB1517----H--	Energize Bus 15 from 17		5.3E-3	5.3E-3	
1ATAD22A-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.080	4.5E-3	4.5E-3	
1ATAD22B-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.090	4.5E-3	4.5E-3	
1ATAD22C-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.080	4.5E-3	4.5E-3	
1ATAD22D-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.090	4.5E-3	4.5E-3	
1CRPMOPERATORH--	Start CRD Pump B		4.0E-3	4.0E-3	
1CSSWMANSTRT-H--	Init Core Spray Following Failure of Auto Initiation		7.5E-3	7.5E-3	
1HISY--------H--	Manually Controlling HPCI in Accordance with QCOP 2300-3	6.540	2.4E-3	1.3E-3	-43.8%
1HISYMANSTARTH--	Init HPCI Following Failure of Auto Initiation		1.2E-3	1.2E-3	
1INSYH-5KGPM-H--	Miss QCOP 2300-6 5000gpm Caution (High Stress)	1.000	6.0E-3	6.0E-3	
1INSYH-ALTINJH--	Switch To An Alternate Inj Source (High Stress)	1.030	4.3E-2	4.3E-2	
1INSYH-COLLDNH--	Decide To Cool Down (High Stress)	1.000	3.7E-3	3.7E-3	
1INSYH-LEVEL-H--	Acknowledge +/-.5" Torus Lvl Alarm (High Stress)	1.000	1.3E-3	1.3E-3	
1INSYH-MINFLWH--	Miss QCOP 2300-6 Caution Minimize Min Flow (High Stress)	1.000	6.0E-3	6.0E-3	
1INSYH-PRESS-H--	Place HPCI In Pressure Control (High Stress)	1.010	1.8E-1	1.8E-1	
1INSYH-TORUS-H--	Switch HPCI Suction To Torus (High Stress)	1.020	4.0E-2	4.0E-2	
1INSYH-X-FER-H--	Defeat HPCI Auto Suction X-Fer (High Stress)	1.000	3.1E-2	3.1E-2	
1INSYO-5KGPM-H--	Miss QCOP 2300-6 5000gpm Caution (Opt Stress)		1.2E-3	1.2E-3	
1INSYO-ALTINJH--	Switch To An Alternate Inj Source (Opt Stress)		7.1E-3	7.1E-3	
1INSYO-COLLDNH--	Decide To Cool Down (Opt Stress)		3.7E-3	3.7E-3	
1INSYO-LEVEL-H--	Acknowledge +/-.5" Torus Lvl Alarm (Opt Stress)		2.7E-4	2.7E-4	
1INSYO-MINFLWH--	Miss QCOP 2300-6 Caution Minimize Min Flow (Opt Stress)		1.2E-3	1.2E-3	
1INSYO-PRESS-H--	Place HPCI In Pressure Control (Opt Stress)		3.6E-2	3.6E-2	
1INSYO-TORUS-H--	Switch HPCI Suction To Torus (Opt Stress)		7.9E-3	7.9E-3	
1INSYO-X-FER-H--	Defeat HPCI Auto Suction X-Fer (Opt Stress)		7.9E-3	7.9E-3	
1RISYMANSTARTH--	Init RCIC Following Failure of Auto Initiation		1.2E-3	1.2E-3	

Table 4-1
QUAD CITIES FAULT TREE HRA RESULTS

NAME	DESCRIPTION	RAW*	Orig. HEP	Rev. HEP	%CHG
1TBPM1-3801B-H--	Start TBCCW Pmp after the Running Pmp has Failed (Fault/Loss of Pwr)		1.4E-3	1.4E-3	
BFPPD12-4101AH--	Fail to Restore Diesel Fire Pmp 1/2-4101A following Test or Maint		5.5E-2	5.5E-2	
BFPPD12-4101BH--	Fail to Restore Diesel Fire Pmp 1/2-4101B following Test or Maint		5.5E-2	5.5E-2	
BSWPM1/2-3901H--	Start the Standby Service Water Pump	1.060	5.3E-3	5.3E-3	
BSWPM2/5-3901-H--	Start Two Service Water Pumps Following a Dual Unit LOOP	1.020	2.7E-2	2.7E-2	

* The RAW value was not determined for those operator actions whose contribution was below the quantification cutoff. These operator actions were deemed to be insignificant contributors and were not evaluated further.

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OAD1-CS01	Init ADS (MLOCA) - [Opt-stress, Zero-depend, No-recov]	24.991	4.9E-3	1.3E-2	165.3%
OAD1-CS02	Init ADS (TRAN, SBO, LOOP, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.742	5.4E-4	1.3E-3	140.8%
OAD1-CS04	Init ADS (TRAN, SBO, ATWS) - [Opt-stress, Low-depend, Yes-recov]	1.000	6.0E-3	5.1E-2	753.9%
OAD1-CS09	Init ADS (TRAN, MLOCA, SLOCA, LOOP, ATWS) - [Mod-stress, Zero-depend, No-recov]	5.996	9.8E-3	1.6E-2	63.3%
OAD1-CS10	Init ADS (SLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.000	1.1E-3	5.2E-2	4583.7%
OAD1-CS11	Init ADS (TRAN, MLOCA, LOOP, ATWS) - [Mod-stress, Zero-depend, No-recov]	2.602	9.8E-3	6.5E-2	565.3%
OAD1-CS12	Init ADS (SLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.000	6.5E-3	5.2E-2	692.6%
OAD1-CS17	Init ADS (TRAN, MLOCA, SLOCA, SBO, LOOP, ATWS) - [High-stress, Zero-depend, No-recov]	1.083	2.5E-2	1.6E-2	-36.0%
OAD1-CS18	Init ADS (SLOCA, LOOP) - [High-stress, Zero-depend, Yes-recov]	4.738	2.7E-3	1.6E-3	-40.7%
OAD1-CS19	Init ADS (TRAN, SLOCA, SBO, LOOP, ATWS) - [High-stress, Low-depend, No-recov]	1.018	7.4E-2	6.5E-2	-11.9%
OAD1-CS20	Init ADS (LOOP) - [High-stress, Zero-depend, Yes-recov]	1.265	2.7E-3	5.2E-2	1808.2%
OAD1-CS21	Init ADS (SLOCA) - [High-stress, Low-depend, No-recov]	1.008	7.4E-2	1.6E-1	111.6%
OAD1-SUM	Init ADS	36.443			
OAD2-CS01	Depress per the HCL Curve - [Op-stress, Zero-depend, No-recov]	1.000	3.2E-2	1.3E-2	-59.4%
OAD2-CS02	Depress per the HCL Curve (TRAN, SLOCA, ISLOCA, SBO, LOOP, ATWS) - [Op-stress, Zero-depend, Yes-recov]	1.383	3.5E-3	1.3E-3	-62.9%
OAD2-CS04	Depress per the HCL Curve (SLOCA) - [Opt-stress, Low-depend, Yes-recov]	1.000	8.8E-3	5.1E-2	482.2%
OAD2-CS08	Depress per the HCL Curve (SBO) - [Opt-stress, High-depend, Yes-recov]		1.8E-2	5.0E-1	2681.4%
OAD2-CS09	Depress per the HCL Curve (ATWS) - [Mod-stress, Zero-depend, No-recov]	1.005	6.4E-2	1.3E-2	-79.7%
OAD2-CS10	Depress per the HCL Curve (TRAN, SLOCA, SBO, LOOP, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	7.137	7.0E-3	1.3E-3	-81.4%
OAD2-CS11	Depress per the HCL Curve (ATWS) - [Mod-stress, Low-depend, No-recov]	1.013	1.1E-1	6.2E-2	-43.3%
OAD2-CS12	Depress per the HCL Curve (LOOP) - [Mod-stress, Low-depend, Yes-recov]	1.000	1.2E-2	5.1E-2	327.0%
OAD2-CS17	Depress per the HCL Curve (ISLOCA) - [High-stress, Zero-depend, No-recov]	1.000	1.6E-1	1.3E-2	-91.9%
OAD2-CS18	Depress per the HCL Curve (SBO) - [High-stress, Zero-depend, Yes-recov]	1.000	1.8E-2	1.3E-3	-92.8%
OAD2-CS19	Depress per the HCL Curve (ISLOCA) - [High-stress, Low-depend, No-recov]	1.001	2.0E-1	6.2E-2	-68.8%
OAD2-SUM	Depress per the HCL Curve	7.539			
OAL-CS01	Lower Water Level - [Opt-stress, Zero-depend, No-recov]	1.000	4.7E-2	4.7E-2	

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OAL-CS09	Lower Water Level (ATWS) - [Mod-stress, Zero-depend, No-recov]	1.007	9.4E-2	9.4E-2	
OAL-CS11	Lower Water Level (ATWS) - [Mod-stress, Low-depend, No-recov]	1.014	1.4E-1	1.4E-1	
OAL-SUM	Lower Water Level	1.021			
OAT-CS01	Manually Init ARI - [Opt-stress, Zero-depend, No-recov]	1.000	1.2E-3	1.2E-3	
OAT-CS17	Manually Init ARI (ATWS) - [High-stress, Zero-depend, No-recov]	1.000	6.0E-3	6.0E-3	
OAT-CS21	Manually Init ARI (ATWS) - [High-stress, Mod-depend, No-recov]	1.000	1.5E-1	1.5E-1	
OAT-SUM	Manually Init ARI	1.000			
OCD-CS01	Cooldown With the Main Condenser - [Opt-stress, Zero-depend, No-recov]	1.000	5.0E-3	5.0E-3	
OCD-CS02	Cooldown With the Main Condenser (SLOCA) - [Opt-stress, Zero-depend, Yes-recov]	1.013	5.5E-4	5.5E-4	
OCD-CS04	Cooldown With the Main Condenser (SLOCA) - [Opt-stress, Low-depend, Yes-recov]	1.032	6.0E-3	6.0E-3	
OCD-SUM	Cooldown With the Main Condenser	1.045			
OCNTS-CS01	Init Containment Sprays (MLOCA, SLOCA) - [Opt-stress, Zero-depend, No-recov]	1.232	1.2E-2	1.2E-2	
OCNTS-CS03	Init Containment Sprays (MLOCA, SLOCA) - [Opt-stress, Low-depend, No-recov]	1.000	6.1E-2	6.1E-2	
OCNTS-CS09	Init Containment Sprays (LLOCA, SLOCA) - [Mod-stress, Zero-depend, No-recov]	1.006	2.5E-2	2.5E-2	
OCNTS-CS10	Init Containment Sprays (TRAN, LLOCA, SBO, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.003	2.7E-3	2.7E-3	
OCNTS-CS11	Init Containment Sprays (SLOCA) - [Mod-stress, Low-depend, No-recov]	1.011	7.4E-2	7.4E-2	
OCNTS-CS13	Init Containment Sprays (LLOCA) - [Mod-stress, Mod-depend, No-recov]	1.000	1.6E-1	1.6E-1	
OCNTS-CS17	Init Containment Sprays - [High-stress, Zero-depend, No-recov]	1.000	6.2E-2	6.2E-2	
OCNTS-CS18	Init Containment Sprays (TRAN, SBO, LOOP, ATWS, IORV) - [High-stress, Zero-depend, Yes-recov]	1.000	6.8E-3	6.8E-3	
OCNTS-CS20	Init Containment Sprays (TRAN, SBO, ATWS) - [High-stress, Low-depend, Yes-recov]	1.000	1.2E-2	1.2E-2	
OCNTS-CS21	Init Containment Sprays - [High-stress, Mod-depend, No-recov]	1.000	2.0E-1	2.0E-1	
OCNTS-CS22	Init Containment Sprays (LLOCA) - [High-stress, Mod-depend, Yes-recov]	1.000	2.2E-2	2.2E-2	
OCNTS-SUM	Init Containment Sprays	1.252			
OCRD-CS01	Restore CRD Injection (SBO) - [Opt-stress, Zero-depend, No-recov]	1.000	6.2E-3	6.2E-3	
OCRD-CS02	Restore CRD Injection (SBO, LOOP) - [Opt-stress, Zero-depend, Yes-recov]	1.000	6.4E-4	6.4E-4	

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OCRD-CS04	Restore CRD Injection - [Opt-stress, Low-depend, Yes-recov]	1.000	6.1E-3	6.1E-3	
OCRD-CS10	Restore CRD Injection (SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	1.000	1.3E-3	1.3E-3	
OCRD-CS12	Restore CRD Injection - [Mod-stress, Low-depend, Yes-recov]	1.000	6.8E-3	6.8E-3	
OCRD-CS17	Restore CRD Injection (SBO) - [High-stress, Zero-depend, No-recov]	1.000	3.1E-2	3.1E-2	
OCRD-CS18	Restore CRD Injection (LOOP) - [High-stress, Zero-depend, Yes-recov]	1.000	3.4E-3	3.4E-3	
OCRD-SUM	Restore CRD Injection	1.000			
OCST-CS01	Align Low Pressure Pump Suction to the CCST (IORV) - [Opt-stress, Zero-depend, No-recov]	1.000	8.4E-2	1.4E-2	-83.1%
OCST-CS02	Align Low Pressure Pump Suction to the CCST (IORV) - [Opt-stress, Zero-depend, Yes-recov]	1.002	9.2E-3	1.4E-3	-84.6%
OCST-CS09	Align Low-Press Pump Suct to the CCST - [Mod-stress, Zero-depend, No-recov]	New	New	1.5E-2	
OCST-CS10	Align Low-Press Pump Suct to the CCST (TRAN, IORV, LOOP, SBO, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.415	1.9E-2	1.5E-3	-91.9%
OCST-CS12	Align Low Pressure Pump Suction to the CCST (SBO, LOOP) - [Mod-stress, Low-depend, Yes-recov]	6.174	2.3E-2	5.1E-2	123.8%
OCST-CS18	Align Low Pressure Pump Suction to the CCST (TRAN, IORV, ATWS) - [High-stress, Zero-depend, Yes-recov]	1.018	4.6E-2	1.9E-3	-95.9%
OCST-CS20	Align Low-Press Pump Suct to the CCST (TRAN, SBO, LOOP, ATWS) - [High-stress, Low-depend, Yes-recov]	1.017	4.9E-2	5.2E-2	5.7%
OCST-CS25	Align Low Press Pmp Suct to the CST	New	New	1.0E+0	
OCST-SUM	Align Low Pressure Pump Suction to the CCST	6.627			
OFW1-CS00	Restart Main Feedwater Pumps or Recover Hotwell Level with Manual Makeup Valves (LOIA) [Opt-stress, Zero-depend, Yes-recover]	New	New	8.4E-3	
OFW1-CS01	Restart Main Feedwater Pumps (TRAN, SLOCA, IORV) - [Opt-stress, Zero-depend, No-recov]	4.988	5.0E-3	1.4E-2	172.4%
OFW1-CS02	Restart Main Feedwater Pumps (TRAN, SLOCA, IORV) - [Opt-stress, Zero-depend, Yes-recov]	New	New	1.4E-3	
OFW1-SUM	Restart Main Feedwater Pumps	4.988			
OFW2-CS01	Manually Scram and cntrl Feedwater - [Opt-stress, Zero-depend, No-recov]	1.000	8.8E-3	8.8E-3	
OFW2-CS09	Manually Scram and cntrl Feedwater (ISLOCA) - [Mod-stress, Zero-depend, No-recov]	1.019	1.7E-2	1.7E-2	
OFW2-SUM	Manually Scram and cntrl Feedwater	1.019			
OFW3-CS01	Restore Feedwater Injection - [Opt-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OFW3-CS17	Restore Feedwater Injection (SBO, LOOP) - [High-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OFW3-SUM	Restore Feedwater Injection	1.000			
OHX-CS01	Align Cooling to RHR (ATWS) - [Opt-stress, Zero-depend, No-recov]	1.000	8.2E-3	1.0E-4	-98.8%
OHX-CS02	Align Cooling to RHR (TRAN, MLOCA, SLOCA, IORV, LLOCA) - [Opt-stress, Zero-depend, Yes-recov]	5.174	9.0E-4	1.0E-5	-98.9%
OHX-CS06	Align Cooling to RHR (IORV) - [Opt-stress, Mod-depend, Yes-recov]	1.001	1.6E-2	1.6E-2	
OHX-CS09	Align Cooling to RHR (ATWS) - [Mod-stress, Zero-depend, No-recov]	2.386	1.7E-2	1.7E-2	
OHX-CS10	Align Cooling to RHR (MLOCA, SLOCA, IORV, SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	54.048	1.9E-3	1.9E-3	
OHX-CS14	Align Cooling to RHR (SLOCA, IORV) - [Mod-stress, Mod-depend, Yes-recov]	1.000	1.7E-2	1.7E-2	
OHX-CS17	Align Cooling to RHR (SBO) - [High-stress, Zero-depend, No-recov]	59.647	4.5E-2	4.5E-2	
OHX-CS18	Align Cooling to RHR (SLOCA, LLOCA) - [High-stress, Zero-depend, Yes-recov]	1.007	4.9E-3	4.9E-3	
OHX-CS22	Align Cooling to RHR (SLOCA) - [High-stress, Mod-depend, Yes-recov]	1.001	2.0E-2	2.0E-2	
OHX-SUM	Align Cooling to RHR	118.264			
OIADS-CS01	Inhibit ADS (MLOCA) - [Opt-stress, Zero-depend, No-recov]	3.770	1.7E-4	1.3E-2	7547.1%
OIADS-CS02	Inhibit ADS (MLOCA) - [Opt-stress, Zero-depend, Yes-recov]	New	New	3.3E-3	
OIADS-CS09	Inhibit ADS (SLOCA) - [Mod-stress, Zero-depend, No-recov]	1.000	3.8E-4	1.3E-2	3321.1%
OIADS-CS10	Inhibit ADS (SLOCA) - [Mod-stress, Zero-depend, Yes-recov]	New	New	3.3E-3	
OIADS-CS26	Inhibit ADS (ATWS) - [Mod-stress, Zero-depend, No-recov]	New	New	3.0E-3	
OIADS-SUM	Inhibit ADS	3.770			
OIB-CS01	Isolate the Break - [Opt-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS09	Isolate the Break (ISLOCA) - [Mod-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS10	Isolate the Break (ISLOCA) - [Mod-stress, Zero-depend, Yes-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS13	Isolate the Break (ISLOCA) - [Mod-stress, Mod-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS14	Isolate the Break (ISLOCA) - [Mod-stress, Mod-depend, Yes-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS17	Isolate the Break (ISLOCA) - [High-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OIB-SUM	Isolate the Break	1.000			
ORP-CS01	Trip Recirculation Pumps - [Opt-stress, Zero-depend, No-recov]	1.000	1.0E-2	1.0E-2	
ORP-CS17	Trip Recirculation Pumps (ATWS) - [High-stress, Zero-depend, No-recov]	1.007	5.1E-2	5.1E-2	
ORP-SUM	Trip Recirculation Pumps	1.007			
OSBCS-CS01	Init Standby Coolant Supply - [Opt-stress, Zero-depend, No-recov]	1.000	1.4E-2	1.4E-2	

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OSBCS-CS02	Init Standby Coolant Supply (ISLOCA) - [Opt-stress, Zero-depend, Yes-recov]	1.000	1.5E-3	1.5E-3	
OSBCS-CS09	Init Standby Coolant Supply (MLOCA) - [Mod-stress, Zero-depend, No-recov]	1.014	2.8E-2	2.8E-2	
OSBCS-CS10	Init Standby Coolant Supply (LLOCA, MLOCA) - [Mod-stress, Zero-depend, Yes-recov]	1.143	3.1E-3	3.1E-3	
OSBCS-CS12	Init Standby Coolant Supply (LLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.200	8.4E-3	8.4E-3	
OSBCS-CS17	Init Standby Coolant Supply (LLOCA, MLOCA) - [High-stress, Zero-depend, No-recov]	1.002	7.2E-2	7.2E-2	
OSBCS-CS19	Init Standby Coolant Supply (LLOCA, MLOCA, ISLOCA) - [High-stress, Low-depend, No-recov]	1.001	1.2E-1	1.2E-1	
OSBCS-SUM	Init Standby Coolant Supply	1.360			
OSDC-CS01	Init Shutdown Cooling - [Opt-stress, Zero-depend, No-recov]	1.000	8.4E-2	8.4E-2	
OSDC-CS02	Init Shutdown Cooling (TRANS, SBO, LOOP, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.002	9.2E-3	9.2E-3	
OSDC-SUM	Init Shutdown Cooling	1.002			
OSL1-CS01	Init one SLC pump (ATWS) - [Opt-stress, Zero-depend, No-recov]	1.501	8.7E-3	8.7E-3	
OSL1-CS17	Init one SLC pump (ATWS) - [High-stress, Zero-depend, No-recov]	1.739	4.5E-2	4.5E-2	
OSL1-SUM	*Init one SLC pump*	2.240			
OSL2-CS01	Init two SLC pump - [Opt-stress, Zero-depend, No-recov]	1.000	8.5E-3	8.5E-3	
OSL2-CS16	Init two SLC pump (ATWS) - [Mod-stress, High-depend, Yes-recov]	1.645	5.6E-2	5.6E-2	
OSL2-SUM	Init two SLC pump	1.645			
OSMP1-CS01	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Opt-stress, Zero-depend, No-recov]	1.001	1.1E-2	1.4E-2	29.1%
OSMP1-CS02	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.009	1.2E-3	1.4E-3	18.3%
OSMP1-CS03	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Opt-stress, Low-depend, No-recov]	1.000	6.0E-2	6.3E-2	5.8%
OSMP1-CS04	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, LOOP, ATWS) - [Opt-stress, Low-depend, Yes-recov]	1.000	6.6E-3	5.1E-2	678.0%
OSMP1-CS09	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Mod-stress, Zero-depend, No-recov]	1.001	2.3E-2	1.5E-2	-33.0%
OSMP1-CS10	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	6.292	2.5E-3	1.5E-3	-38.4%

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OSMP1-CS11	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Mod-stress, Low-depend, No-recov]	1.000	7.2E-2	6.5E-2	-10.2%
OSMP1-CS12	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [Mod-stress, Low-depend, Yes-recov]	1.837	7.9E-3	5.1E-2	551.4%
OSMP1-CS17	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (SBO with No HPCI or RCIC success) - [High-stress, Zero-depend, No-recov]	New	New	1.9E-2	
OSMP1-CS18	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [High-stress, Zero-depend, Yes-recov]	1.176	7.3E-3	1.9E-3	-74.0%
OSMP1-CS19	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (SBO) - [High-stress, Low-depend, No-recov]	1.000	1.1E-1	6.8E-2	-38.1%
OSMP1-CS20	Init SSMP with Suct Aligned to the CCST - No HPCI Inj Sig (TRAN, SBO, LOOP, ATWS) - [High-stress, Low-depend, Yes-recov]	1.062	1.2E-2	5.2E-2	331.7%
OSMP1-SUM	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal	7.378			
OSMP2-CS01	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal - [Opt-stress, Zero-depend, No-recov]	1.000	9.0E-3	9.0E-3	
OSMP2-CS10	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (TRAN, SBO, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.000	2.0E-3	2.0E-3	
OSMP2-CS12	Init SSMP with Suct Aligned to the FP - No HPCI Inj Sig (TRAN, SBO, ATWS) - [Mod-stress, Low-depend, Yes-recov]	1.000	7.4E-3	7.4E-3	
OSMP2-CS17	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (SBO) - [High-stress, Zero-depend, No-recov]	1.000	6.3E-2	6.3E-2	
OSMP2-CS18	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (LOOP) - [High-stress, Zero-depend, Yes-recov]	1.023	6.9E-3	6.9E-3	
OSMP2-CS20	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (LOOP) - [High-stress, Low-depend, Yes-recov]	1.005	1.2E-2	1.2E-2	
OSMP2-SUM	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal	1.028			
OSMP3-CS01	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal - [Opt-stress, Zero-depend, No-recov]	1.000	7.8E-3	7.8E-3	
OSMP3-CS03	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (ATWS) - [Opt-stress, Low-depend, No-recov]	1.299	5.6E-2	5.6E-2	
OSMP3-CS09	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (TRAN, SLOCA, LOOP, ATWS) - [Mod-stress, Zero-depend, No-recov]	2.083	1.4E-2	1.4E-2	
OSMP3-CS10	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (SLOCA) -	1.000	1.5E-3	1.5E-3	

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
	[Mod-stress, Zero-depend, Yes-recov]				
OSMP3-CS11	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA, ATWS) - [Mod-stress, Low-depend, No-recov]	1.004	6.3E-2	6.3E-2	
OSMP3-CS12	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.000	7.0E-3	7.0E-3	
OSMP3-CS13	Init SSMP with Suct Aligned to the CCST or with HPCI Injt Sig (TRAN, ATWS) - [Mod-stress, Mod-depend, No-recov]	1.000	1.5E-1	1.5E-1	
OSMP3-CS17	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (TRAN, SLOCA, SBO, LOOP, ATWS) - [Hi-stress, Zero-depend, No-recov]	1.061	4.8E-3	4.8E-3	
OSMP3-CS18	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (SLOCA) - [High-stress, Zero-depend, Yes-recov]	1.000	5.3E-3	5.3E-3	
OSMP3-CS19	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA, LOOP, ATWS) - [High-stress, Low-depend, No-recov]	1.015	9.6E-2	9.6E-2	
OSMP3-CS20	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA) - [High-stress, Low-depend, Yes-recov]	1.000	1.1E-2	1.1E-2	
OSMP3-SUM	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal	2.462			
OSPC-CS01	Init Supp Pool Cooling (ATWS) - [Opt-stress, Zero-depend, No-recov]	1.000	4.7E-4	1.0E-4	-78.7%
OSPC-CS02	Init Supp Pool Cooling (TRAN, MLOCA, SLOCA, IORV, LLOCA) - [Opt-stress, Zero-depend, Yes-recov]	1.008	5.2E-5	1.0E-5	-80.8%
OSPC-CS04	Init Supp Pool Cooling (MLOCA) - [Opt-stress, Low-depend, Yes-recov]	1.000	5.5E-3	5.5E-3	
OSPC-CS09	Init Supp Pool Cooling (ATWS) - [Mod-stress, Zero-depend, No-recov]	2.155	1.3E-3	1.3E-3	
OSPC-CS10	Init Supp Pool Cooling (SLOCA, IORV, SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	4.817	1.4E-4	1.4E-4	
OSPC-CS17	Init Supp Pool Cooling (SBO) - [High-stress, Zero-depend, No-recov]	1.000	6.3E-3	6.3E-3	
OSPC-CS18	Init Supp Pool Cooling (SLOCA) - [High-stress, Zero-depend, Yes-recov]	1.000	6.9E-4	6.9E-4	
OSPC-SUM	Init Supp Pool Cooling	5.979			
OSS-CS01	Rest Supp Systems (SBO) - [Opt-stress, Zero-depend, No-recov]	1.000	4.0E-2	4.0E-2	
OSS-CS02	Rest Supp Systems (SBO, LOOP) - [Opt-stress, Zero-depend, Yes-recov]	1.000	4.4E-3	4.4E-3	
OSS-CS04	Rest Supp Systems (SBO, LOOP) - [Opt-stress, Low-depend, Yes-recov]	1.000	9.7E-3	9.7E-3	
OSS-CS09	Rest Supp Systems (SBO) - [Mod-stress, Zero-depend, No-recov]	1.000	8.0E-2	8.0E-2	
OSS-CS10	Rest Supp Systems (SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	1.000	8.8E-3	8.8E-3	
OSS-CS11	Rest Supp Systems (SBO) - [Mod-stress, Low-depend, No-recov]	1.000	1.3E-1	1.3E-1	

Table 4-2
QUAD CITIES PLANT RESPONSE TREE HRA RESULTS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OSS-CS12	Rest Supp Systems (SBO, LOOP) - [Mod-stress, Low-depend, Yes-recov]	1.000	1.4E-2	1.4E-2	
OSS-CS17	Rest Supp Systems (SBO) - [High-stress, Zero-depend, No-recov]	1.000	2.0E-1	2.0E-1	
OSS-CS18	Rest Supp Systems (SBO, LOOP) - [High-stress, Zero-depend, Yes-recov]	1.000	2.2E-2	2.2E-2	
OSS-CS19	Rest Supp Systems (SBO) - [High-stress, Low-depend, No-recov]	1.000	2.4E-1	2.4E-1	
OSS-CS20	Rest Supp Systems (SBO, LOOP) - [High-stress, Low-depend, Yes-recov]	1.000	2.6E-2	2.6E-2	
OSS-SUM	Rest Supp Systems	1.000			
OVNT-CS01	Init Cont Vent - [Opt-stress, Zero-depend, No-recov]	1.000	9.2E-2	9.2E-2	
OVNT-CS02	Init Cont Vent (TRAN, SBO, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.000	1.0E-2	1.0E-2	
OVNT-CS10	Init Cont Vent (TRAN, LLOCA, SBO, LOOP, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.009	2.0E-2	2.0E-2	
OVNT-CS18	Init Cont Vent (TRAN, LLOCA, MLOCA, SLOCA, IORV, SBO, LOOP, ATWS) - [High-stress, Zero-depend, Yes-recov]	1.238	5.1E-2	5.1E-2	
OVNT-SUM	Init Cont Vent	1.247			

(Note: after each set of operator action cases there is a summary line that gives the combined Risk Achievement Worth for all the cases.)

Concern e: Treatment of Equipment Calibration Errors

The SER is correct in stating that the original IPE included six pre-initiator basic events for which human error probability (HEP) estimates were used. The conclusion that "calibration errors were dismissed from consideration," however, is incorrect. ComEd regrets this misunderstanding; the apparent cause of this misunderstanding is discussed later in this section.

A review of the SER concern with calibration errors found that:

- The original IPE included numerous basic events for single instruments and common-cause failure of multiple instruments;
- Generic instrument failure rates listed in the original IPE submittal report (not HEPs) had been used in the fault tree quantification and (based on basic event descriptions used in fault trees) the generic failure rates had been considered to apply to out-of-calibration and other failures; and
- The mission times used for most instruments were found to have been non-conservative.

More detail on these findings during the review of the SER concern is given below. Consequently, failure probability changes for pertinent instrument basic events were adopted for the Modified Quad Cities IPE; these changes reflect results of a plant-specific review of calibration events over an approximate five-year period (1991 - 1995), i.e., actual performance of plant personnel for thousands of recent calibrations was considered rather than estimating HEPs using HRA methods.

The changes in instrument failure probabilities resulted in significant changes in the quantification of some fault trees. Nevertheless, the impact of these changes on total CDF was small. As discussed in more detail in the writeup for Modification 6, "Discussion of the Modified Quad Cities IPE Results," the main impact of changes made in response to the instrument miscalibration concern was an increase in the CDF contribution of the General Transient initiator.

Background Information on SER Concern

As indicated in the SER, information on six pre-initiator human events was provided in the ComEd response to the RAI (specifically, RAI question 18). The RAI question cited information in the original IPE Submittal Report "which indicates that some pre-initiator human error analysis was performed," and requested additional information.

For this reason, the information request in RAI question 18 (and in the related question 41 in the Dresden RAI) had been interpreted as dealing with details on the use of HRA in the ComEd IPEs. Consequently, the ComEd response had dealt only with those pre-initiator basic events included in fault trees for which HEPs were used. Most of those events involved failing to restore a system following testing.

> NOTE: In discussing the six pre-initiator events for which HEPs were used, the SER stated: "none of these events are associated with calibration activities." In fact, four of the six events were failure to restore ATWS master trip units (MTUs) following calibration testing. The HEPs applied to failure to restore the MTUs, however, not miscalibration.

Instrument basic events in fault trees for which HEPs were not used had been considered to be outside the scope of the RAI question due to the question's apparent concern with "human error analysis." ComEd regrets this misunderstanding.

Fault trees involving the Common Actuation, High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (RHR) and RHR Service Water (RHRSW), Core Spray, and Anticipated Transient Without Scram (ATWS) systems include 70 basic events for single instrument failures and over 40 basic events for common cause instrument failure. Instruments modeled can be categorized as follows:

- Pressure (including flow switches)
- Level
- Temperature
- Time Delay

For example, the HPCI fault tree (developed for the original IPE) includes the following instruments that could potentially give a false system isolation signal and thus cause a system failure:

- Reactor Pressure
- HPCI Steam Line Flow
- HPCI Area Temperature

As an example, failure (i.e., a spurious trip at low flows) of any of the four HPCI High Steam Line Flow flow switches would cause system failure. A basic event description for one of these switches shown on the fault tree is "DPS 1-2391-04 OUT OF CALIBRATION."

Similarly, four basic events for HPCI area temperature switches are shown on the fault tree. An example of a basic event description shown on the fault tree is

"TS 1-2370A OUT OF CALIBRATION." Because of the trip logic employed, specific combinations of two temperature switches are required to give a system isolation; therefore, the fault tree modeling of the temperature switches includes both AND and OR gates. The fault tree also includes four common cause basic events for temperature switches, (i.e., common cause failure of each combination of two temperature switches that could give a system isolation). An example of a common cause basic event description shown on the fault tree is "COMMON CAUSE FAILURE OF TS 1-2370A & C."

These HPCI high flow and temperature instruments are examples of instruments where an out-of-calibration instrument, whether due to miscalibration or instrument drift, could go undetected until the next calibration surveillance and thus represent a pre-initiator event.

The failure probabilities used for these and similar basic events for instruments were based on generic failure rates. For single instruments, values used were given in Table 4.4.1-6, "Quad Cities IPE Generic Failure Data," of the original IPE Submittal Report. The common cause failure rates for instruments were given in Table 4.4.1-5, "Common Cause Failure Data for Quad Cities IPE."

In summary, although the ComEd RAI response did not address pre-initiator events for which generic failure rates rather than HEPs were used, many such instruments were, in fact, included in the fault trees developed for the original IPE. Nevertheless, in response to the concern in the SER on this issue, a plant-specific assessment of the suitability of using generic failure rates was performed and is discussed below.

Plant-Specific Assessment of Use of Generic Failure Rates for Instruments

The fault tree quantification for the original IPE was reviewed with respect to treatment of instruments where miscalibration was judged to represent a pre-initiator concern.

> In theory, the failure probability for components has a complex time dependence. The approach used in the original IPE for major items such as ECCS pumps and the Emergency Diesel Generators was to collect site-specific failure to start (probability per demand) and failure to run (probability per hour) data. This approach is reflected in the information given in Table 4.4.1-3, "Quad Cities-Specific Component Failure Rates," of the original IPE Submittal Report.
>
> For many highly reliable items such as instruments, however, the approach used in the original IPE was to adopt generic failure rate data (probability per hour) and to combine these data with a mission time.

The original IPE quantification was found to have used system mission times for instruments to determine failure probabilities based on generic failure rates. Use of system mission times (24 hours or less) is inappropriate for "standby" instruments where miscalibration could not be detected until the next calibration surveillance. For such instruments, therefore, mission times for the Modified Quad Cities IPE were increased to one-half the instrument surveillance interval.

In response to the SER concern with pre-initiator treatment, a plant-specific review of Quad Cities Licensee Event Reports (LERs), Deviation Reports (DVRs), and Problem Investigation Reports (PIRs) was performed using a computer database that included the approximate period of 1991 - 1995.

The review identified no failure to restore events that warrant addition to the Modified Quad Cities IPE as a pre-initiator.

For the instrument types discussed above (which were included in fault trees for the original IPE and for which miscalibration is a pre-initiator concern), the review covering the approximate period of 1991 - 1995 also identified no events that could have caused a miscalibration failure with respect to the IPE model.

Events involving instruments did occur during that period, but the review concluded that those events were not pertinent to the pre-initiator concern. Examples included:

Improper Oiler Setup for an EDG Cooling Water Pump
This event resulted in damage to the pump bearings. Extensive corrective actions included marking the proper oil level on the bearings for important pumps with red paint. Also, Maintenance and Operations personnel were trained on proper oiler setup. A repeat occurrence of this type of failure-to-restore event is considered unlikely due to these actions.

Miscalibration of Reactor Building Ventilation Radiation Monitor Setpoints
The cause was a computer program used in the calibrations that was written by an IM supervisor. Because of the corrective actions taken, the fact that this program was specific to these radiation monitors, and that this event involved radiation monitors that are not included in the IPE model, this event does not constitute a miscalibration event for the Modified IPE model.

Instabilities in RCIC Flow Due To Flow Controller Proportional Band Set Too Low

This resulted in RCIC flow instabilities discovered during the initial system testing following a refueling outage. This was the first opportunity for testing the proportional band setting after the settings were changed, so the system was never declared operable.

Standby Gas Treatment System Time Delay Relay

A time delay relay for the standby gas treatment system was set to 27 rather than within 25 seconds as required by the calibration procedure. The miscalibration occurred because the work instructions were written after a new procedure was implemented, but before the procedure books in System Engineering were updated. Corrective actions will prevent this type of miscalibration from occurring in the future.

HPCI Flow Transmitter Set to an Incorrect Span

This event occurred because a procedure was not updated when the transmitter was changed out to a different model. The result was that the HPCI pump would have produced 2% more flow than what was indicated. Although this event is a miscalibration, the root cause was a failure in the design control process for not updating the procedures following implementation of a design change. Because the error was small and in the conservative direction, HPCI was considered operable the entire time.

Further plant-specific analysis was carried out by dividing the instruments into three categories:

- Temperature
- Time Delay
- Pressure (including level and flow instruments)

To estimate plant-specific miscalibration rates pertinent to the Modified Quad Cities IPE, 0.5 failures each were assumed for the Pressure, Temperature, and Time Delay categories (as a Bayesian estimate). These failure numbers divided by the estimated number of surveillances during the approximately 5 year period gives the following site-specific estimates for miscalibration probability due to human error:

Instrument Category	**Failures**	**Failure Probability (due to human error)**
Time Delay	0.5	5.95E-3/demand
Temperature	0.5	3.47E-3/demand
Pressure (including level and flow)	0.5	2.77E-4/demand

Note that: (1) additional instruments (in balance of plant systems, for example) are not included in the estimated number of surveillances (based on instruments reviewed for the original IPE); and (2) the database search for failures included events involving systems not in the PRA model, including balance of plant systems. Therefore, these site-specific failure probability estimates are judged to be conservative.

The increased instrument mission times adopted for the Modified Quad Cities IPE multiplied by the appropriate generic failure rates were found to give instrument failure probabilities that are comparable to or greater than the demand failure probability estimates given above for human error events.

Therefore, the plant-specific assessment for time delay, temperature, and pressure (including level and flow) instruments concluded that the use of generic time-dependent failure rates and instrument-specific mission times in the Modified Quad Cities IPE is appropriate for modeling all types of failures of these instruments, including failures due to human error such as miscalibration.

As discussed below in the writeup for Modification 6, "Discussion of the Modified Quad Cities IPE Results," the main impact of these changes appears to have been an increase in the CDF contribution of the General Transient initiator. Although these changes had a significant impact on the quantification of some fault trees, the overall impact on other initiators and on total CDF appears to be small.

MODIFICATION 5: Plant-Specific and Generic Data Impact on CDF

The NRC's Staff Evaluation Report (SER) on the Quad Cities IPE raised a concern regarding the inappropriate use of generic data. The SER stated, in part:

> *The data used for estimating initiating event frequencies, component unavailabilities, and failure probabilities are comparable with generic data...however, the plant-specific failure-to-open-at-demand rate for the electromatic relief valves (ERV) was estimated during the IPE to be two orders of magnitude higher than the generic (NUREG/CR 4450) value used in the IPE...The staff...believes that more supporting evidence is needed to confirm that the plant-specific and generic data used in the IPE reflect the actual plant experience at the time of the IPE and that the CDF is not significantly affected from the use of inapplicable data.*

Another NRC concern with the Quad Cities IPE plant specific data was first raised in 1993 by the Diagnostic Evaluation Team at Quad Cities. The DET questioned the impact on CDF of more recent unavailability and failure rates for key plant systems such as HPCI, RCIC, Safe Shutdown Makeup Pump (SSMP), RHR, and Emergency Diesel Generators (EDGs). In a February 28, 1996, request, the NRC formally asked Quad Cities to estimate the impact on CDF of recent (1993 through 1995) unavailability and reliability data for those key systems. This data was collected and put into the model. The result was a factor of 2.8 increase in CDF, most of this increase being attributable to a single occurrence of unavailability involving the SSMP room cooler in 1993. This information was provided to the NRC in a March 15, 1996, letter from the station to NRR.

Reliability data for the Modified IPE for key systems was based on the last three years of failure data because this information was already available from the data collection effort earlier this year. The fail-to-start and fail-to-run probabilities are shown in Table 5-1. Reliability values from the original IPE, the March 15th submittal, and generic values from NUREG/CR-4550 are shown for comparison. For the most part, the reliability values used in the Modified IPE are the same as for the March 15th submittal. In four cases where no failures were experienced in the 1993 through 1995 time frame, a Bayesian method was used to estimate the failure probability. This method blended the actual number of demands or run hours with values from NUREG/CR-4550. The EDG and RHR pumps experienced no failures to run and the RHR pumps experienced no failures to start during the 1993 through 1995 time period. For the EDG fail-to-run probability, 0.5 failures were assumed and divided by the total run time of the three EDGs. The assumed 0.5 failure method was also used for the RHR pump fail-to-start and fail-to-run probabilities. The assumption of 0.5 failures resulted in failure rates that are comparable to the respective NUREG/CR-4550 values. In contrast, the Bayesian update method gave failure rates that were considerably less than the generic values. The assumption of 0.5 failures was considered to be the more conservative approach for these three failure probabilities and was the method used for the Modified IPE.

In the 1994 RAI on the Quad Cities IPE, the NRC questioned the use of a generic fail-to-open probability for Electromatic Relief Valves (ERVs) rather than the plant-specific value that was two orders of magnitude higher. In their response to the RAI, ComEd stated that the generic ERV failure rate was used in error, but that the error had little impact on the resulting total CDF. To correct this error from the original IPE, the generic ERV fail-to-open probability of 3.0E-04 was replaced with the plant specific failure rate of 3.57E-02 for the Modified IPE. This plant specific value is derived from data from the time period of the original IPE (January 1985 through December 1991).

Unavailability data from the most recent two years (1994 and 1995) was used in the Modified IPE for components in the same key systems addressed in the March 15th submittal. This data is summarized for comparison in Table 5-2, along with the corresponding maintenance unavailabilities from the original IPE and from the March 15th submittal. Using the most recent two years worth of data is in keeping with the methodology being used for the ComEd plant PRA model updates.

A final data-related change that was made for the Modified IPE was the use of common failure and unavailability values for the three EDGs rather than use of a separate value for each diesel. This change has the advantages of providing a larger data source (all three rather than a single diesel) for the failure data and also removes any unit-to-unit asymmetries from the PRA model. The three EDGs are identical and are maintained in a similar way, therefore the use of common failure and unavailability values is justified.

The revised reliability and unavailability values used for the Modified IPE were based on information collected for the March 15th submittal to the NRC. This failure and unavailability data was collected in a manner similar to, but not identical to the methods used for the original IPE. For the first periodic update to the PRA model for Quad Cities, another data collection and analysis effort will take place for key systems from the original and Modified IPEs, and for any systems or components that are perceived to be "bad actors". This effort will encompass the most recent unavailability data, and failures over a time period long enough to contain statistically significant data.

TABLE 5-1

RELIABILITY DATA

From Table 4.4.1-3 in Original IPE Submittal

COMPONENT TYPE GROUPING AND FAILURE MODE	ORIGINAL IPE	3/15/96 SUBMITTAL	NUREG/CR 4550	MODIFIED IPE	SOURCE
U1 EDG FAILURE TO START	1.60E-02	9.26E-03	-	-	
U2 EDG FAILURE TO START	1.38E-02	2.47E-02	-	-	
1/2 EDG FAILURE TO START	9.94E-03	1.30E-02	-	-	
EDG FAILURE TO START	-	-	3.00E-02	1.42E-02	93-95 Failures
U1 EDG FAILURE TO RUN	4.27E-03	3.83E-03	-	-	
U2 EDG FAILURE TO RUN	1.83E-02	2.81E-03	-	-	
1/2 EDG FAILURE TO RUN	3.19E-03	2.76E-03	-	-	
EDG FAILURE TO RUN	-	-	2.00E-03	1.02E-03	(1)
DGCWP FAILURE TO START	4.29E-03	1.33E-02	4.00E-04	1.33E-02	93-95 Failures
DGCWP FAILURE TO RUN	1.10E-03	1.81E-03	3.00E-05	1.81E-03	93-95 Failures
EDG OUTPUT BREAKER FAILS TO FUNCTION	5.49E-03	5.24E-03	3.00E-03	5.24E-03	93-95 Failures
HPCI TURBINE FAILS TO START	1.38E-02	3.77E-02	1.00E-02	3.77E-02	93-95 Failures
HPCI TURBINE FAILS TO RUN	2.20E-04	2.20E-04	5.00E-03	2.77E-03	(2)
RHR PUMP FAILS TO START	4.05E-04	4.43E-04	4.00E-04	4.43E-04	(1)
RHR PUMP FAILS TO RUN	7.24E-04	5.77E-05	3.00E-05	5.77E-05	(1)
RHRSW PUMP FAILS TO START	5.18E-04	5.75E-04	4.00E-04	5.75E-04	93-95 Failures
RHRSW PUMP FAILS TO RUN	2.70E-05	6.95E-05	3.00E-05	6.95E-05	93-95 Failures
RCIC TURBINE FAILS TO START	1.74E-02	1.08E-02	1.00E-02	1.08E-02	93-95 Failures
RCIC TURBINE FAILS TO RUN	2.20E-04	2.20E-04	5.00E-03	2.84E-03	(2)
SSMP FAILS TO START	6.33E-03	4.03E-03	4.00E-04	3.69E-04	(2)
SSMP FAILS TO RUN	1.06E-04	1.06E-04	3.00E-05	2.99E-05	(2)
ERV FAILS TO OPEN	3.00E-04	(3)	3.00E-04	3.57E-02	85-91 Failures

Notes:

(1) Based on an assumed 0.5 failure divided by the sum of the 93-95 run times.

(2) No failures were experienced during the 93-95 time period. Values shown are a Bayesian estimate using 93-95 demands or run times and NUREG/CR-4550 generic data.

(3) ERV failure rates were not adjusted for the March 15, 1996, submittal to the NRC.

TABLE 5-2

MAINTENANCE UNAVAILABILITY DATA

From Table 4.4.1-4 in Original IPE Submittal

COMPONENT TYPE	ORIGINAL IPE (1/85 - 12/91)	3/15/96 SUBMITTAL (1/93 - 12/95)	MODIFIED IPE	SOURCE
U1 EDG	8.69E-03	8.95E-03	-	-
U2 EDG	1.28E-02	1.38E-02	-	-
1/2 EDG	1.38E-02	5.33E-02	-	-
ALL EDGs	-	-	1.90E-02	94-95 SSPI
DGCWP	5.45E-03	1.07E-03	5.03E-03	94-95 OOS
EDG OUTPUT BREAKER	2.26E-03	8.83E-04	8.83E-04	93-95 OOS
HPCI TURBINE	1.45E-02	3.33E-02	1.80E-02	94-95 SSPI
RHR PUMP	6.51E-03	3.44E-03	3.37E-03	94-95 OOS
RHRSW PUMP	7.77E-03	2.05E-02	1.76E-03	94-95 OOS
RCIC TURBINE	9.40E-03	1.12E-02	1.27E-02	94-95 SSPI
SSMP	9.37E-03	1.11E-01	4.72E-03	94-95 OOS/NTS

Notes:

SSPI - INPO Safety System Performance Indicator
OOS - Out-of-service information from Maintenance Rule Database
NTS - ComEd Nuclear Tracking System

MODIFICATION 6: Discussion of the Modified Quad Cities IPE Results

ComEd combined the modifications described in the previous sections in an integrated Modified Quad Cities IPE model.

This model includes:

- Addition of several support system based initiators;
- Revision of Common Cause Factors (CCFs); and
- Revision of Human Reliability Analysis (HRA).

Additionally, the original IPE model was changed to provide as realistic a representation as possible of the current Quad Cities severe accident risk profile. These changes included:

- Revision of ATWS success criterion;
- Revisions, including mission time changes, for standby instruments to reflect a plant-specific assessment of pre-initiator concerns involving instrument calibration;
- Use of recent unavailability and failure data for five risk significant systems - HPCI, RCIC, SSMP, Diesel Generators, and RHR; and

- Use of a plant-specific rather than generic value for Electromatic (main steam) Relief Valve failure to open.

The Modified Quad Cities IPE results show a core damage frequency (CDF) of 2.2E-06/yr. These results represent the integration of the changes listed above.

Table 6-1 summarizes the significant changes that resulted in these CDF values for Quad Cities. Similarly, Table 6-2 provides the CDF contribution by initiating event and Table 6-3 shows the changes, by initiator, resulting from the modifications made to the model subsequent to the October 1994 RAI response. Table 6-4 provides a listing of the top 100 sequences. The following discussion provides some insight into the contribution of each of these changes, including competing effects, on the final, integrated model.

Modifications Quantified for the Quad Cities RAI Response (Results submitted to the NRC in October 1994)

<u>Special Initiators due to Loss of AC Buses</u>

For the response to the RAI on the original Quad Cities IPE submittal, two new special initiators were quantified due to a reanalysis of contributions due to internal flooding and water spray. The special initiators added for this quantification were Loss of Bus 13 (LB13) and Loss of MCC 18-2 (L182).

Subsequent Modifications and Other Significant Changes

Addition of Several Support System Based Initiators

New initiators added to the model are: Loss of Service Water (LOSW), Loss of Instrument Air (LOIA), Loss of 4 kV Bus 11 (LB11), Loss of 4 kV Bus 12 (LB12), Loss of 4 kV Bus 14 (LB14), Loss of 480 V Bus 18 (LB18), and Loss of 125 VDC Bus 1B-1 (L1B1). The three special initiators added and quantified for the RAI response were also included in the Modified IPE model.

The Loss of 125 VDC Bus initiator contributes approximately 3.5% to the total CDF. The Loss of Service Water initiator contributes approximately 3% of the total CDF. The contribution of the other new special initiators is very small and is discussed in the writeup for Modification 1. The sequences represented by these initiators are variations of General Transient (GTR) initiator sequences identified in the original IPE.

Revised Component Data Values

During this analysis changes were made to data values used in the quantification of the Quad Cities model. Changes were made to some Common Cause Factors (CCFs) and to some component reliability and unavailability data.

A lower bound of 0.01 was imposed on the site-specific beta factors used in the calculation of CCFs. This increased many of the CCFs used in the Quad Cities fault trees. Although this increase had an impact on some fault trees, this change was combined with changes in the reliability and unavailability data for components identified in Modification 5 and the overall impact on the results was small.

Revised Success Criterion for ATWS Initiator

The new ATWS success criterion is that operator action to inhibit ADS must succeed. Addition of this criterion via a new node (OIADS) in the plant response tree (PRT) resulted in an increase in the CDF contribution of the ATWS initiator.

Revised Human Reliability Analysis

As discussed in the writeup for Modification 4, "Revised Human Error Probabilities and Treatment of Pre-Initiator Events," HEP values were reviewed and revised. Significant HEP revisions included the following:

- Values for operator failure to initiate ADS (node OAD) were increased;
- Values for operator failure to swap the low pressure ECCS pump suction to the Condensate Storage Tank (node OCST) were increased;
- Values for operator failure to restart a feedwater pump (node OFW1) were increased;
- Values for operator failure to align cooling to RHR (node OHX) were reduced;
- Values for operator failure to inhibit automatic depressurization (node OIADS) were increased;

- Values for operator failure to initiate the safe shutdown makeup pump from the CCST (node OSMP1) were increased; and

- Values for operator failure to initiate suppression pool cooling (node OSPC) were decreased.

Revised Modeling of Standby Instruments, Including Mission Times

As a result of a review of instruments that could potentially be miscalibrated and represent a pre-initiator type error, the modeling of standby instruments was revised. The significant instruments were found to have been included correctly in the fault trees for the original IPE. In calculation of the failure probabilities for these instruments, however, non-conservative mission times had been used. New mission times of one-half the surveillance interval were chosen for the pertinent instruments.

A review of site-specific experience for a five-year period concluded that the existing standby instrument basic events (including common cause basic events in some systems) were appropriate. The review concluded that the generic failure rates used in the original IPE, together with the corrected mission times, adequately treated pre-initiator events related to calibration issues.

Results of the Modifications

The overall results of the changes are shown in Tables 6-2 and 6-3. As can be seen from the comparison with the original IPE (and RAI response) CDF, the modifications have resulted in a net increase in CDF by about 80%.

The safe shutdown makeup pump (SSMP) plays a very significant role in reducing the overall CDF at Quad Cities. This pump fulfills a mission similar to RCIC but is motor-driven and has diverse power sources. A sensitivity calculation was run to demonstrate the value of this system. The results show that the CDF would be approximately 5 times higher without the SSMP.

TABLE 6-1

CORE DAMAGE FREQUENCY (CDF) VALUES FROM SUCCESSIVE QUAD CITIES PRA MODEL CHANGES

MODEL AND NRC SUBMITTAL DATE	CDF	SIGNIFICANT CHANGES
Original IPE Model (December 1993)	1.2 E-06/yr	(Base Model)
RAI Submittal (October 1994)	1.2 E-06/yr	• Consideration of water spray as an initiator causing Losses of Bus 13 and MCC 18-2.
Modified IPE Model (August 1996)	2.2 E-06/yr	• Additional ATWS success criterion of operator inhibiting ADS. • Human Reliability Analysis revisions. • Common Cause Factor revisions. • Additional support system based initiators and revision of several initiating event frequencies. • Revised modeling of standby instruments, including mission times. • Revised unavailability and reliability data for key systems

TABLE 6-2

QUAD CITIES MODIFIED IPE INITIATING EVENTS

NAME	DESCRIPTION	CDF	PERCENT
DLOOP	LOSS OF OFFSITE POWER IN BOTH UNITS	1.23E-06	56.67%
LOOP	LOSS OF OFFSITE POWER IN ONE UNIT	2.51E-07	11.58%
GTR	GENERAL TRANSIENT	2.34E-07	10.80%
ATWS	ATWS INITIATOR	1.66E-07	7.68%
MLOCA	MEDIUM LOCA	1.12E-07	5.16%
L1B1	LOSS OF 125 VDC BUS 1B-1	7.44E-08	3.43%
LOSW	LOSS OF SERVICE WATER (INCLUDING LOIA CONTRIBUTION)	6.49E-08	2.99%
LLOCA	LARGE LOCA	2.10E-08	0.97%
LB13	LOSS OF BUS 13	5.19E-09	0.24%
LOIA	LOSS OF INSTRUMENT AIR (EXCLUDING CONTRIBUTION TO LOSW)	4.03E-09	0.19%
SLOCA	SMALL LOCA	2.56E-09	0.12%
IORV	INADVERTENT OPEN RV + OTHER INITIATORS x RV FAILURE TO CLOSE	1.23E-09	0.06%
LB14	LOSS OF BUS 14	8.15E-10	0.04%
LB18	LOSS OF BUS 18	6.79E-10	0.03%
L182	LOSS OF MCC 18-2	4.08E-10	0.02%
LB11	LOSS OF BUS 11	3.30E-10	0.02%
LB12	LOSS OF BUS 12	3.06E-10	0.01%
ISLOCA	INTERFACING SYSTEM LOCA	2.64E-10	0.01%
	TOTAL	2.2E-06	100%

TABLE 6-3

QUAD CITIES MODIFIED IPE CHANGES BY INITIATOR

INITIATOR	MODIFIED IPE CDF	ORIGINAL IPE CDF	CHANGE IN CDF	MAIN REASONS APPARENT FOR CHANGE
DLOOP	1.23E-06	6.74E-07	+82%	Higher failure probabilities for diesels, HPCI and RCIC
LOOP	2.51E-07	1.92E-07	+31%	Increased instrument failure probability (increased mission time) and increased human error probabilities.
GTR	2.34E-07	4.69E-08	+399%	Increased instrument failure probability (increased mission time) and increased human error probabilities.
ATWS	1.66E-07	7.61E-08	+118%	Addition of top node OIADS, failure of operator to inhibit ADS.
MLOCA	1.12E-07	1.72E-07	-35%	Reevaluation of human error probabilities, primarily reduction in operator action to depressurize human error probability.
L1B1	7.44E-08	N/A	N/A	New special initiator.
LOSW	6.49E-08	N/A	N/A	New special initiator.
LLOCA	2.10E-08	2.48E-08	-15%	No significant changes.
LB13	5.19E-09	5.37E-09 *	-3%	No significant changes.
LOIA	4.03E-09	N/A	N/A	New special initiator.
SLOCA	2.55E-09	1.14E-08	-78%	Changes in human error probabilities, particularly failure to depressurize and failure to restart feedwater.
IORV	1.23E-09	8.71E-10	+41%	Large percentage change in frequency due to the low initial failure frequency close to truncation limit of 1E-12 - no significant changes that impacted this event.
LB14	8.15E-10	N/A	N/A	New special initiator.
LB18	6.79E-10	N/A	N/A	New special initiator.
L182	4.09E-10	4.08E-10 *	0%	No significant changes.
LB11	3.30E-10	N/A	N/A	New special initiator.
LB12	3.07E-10	N/A	N/A	New special initiator.
ISLOCA	2.64E-10	6.30E-10	-58%	Large percentage change in frequency due to the low initial failure frequency close to truncation limit of 1E-12 - no significant changes that impacted this event.
Total	**2.2E-06**	**1.2E-06/yr**	**+83%**	**Primarily HRA changes and failure data changes**

* Not included in original IPE submittal, but quantified for response to NRC RAI.

TABLE 6-4

DOMINANT ACCIDENT SEQUENCES FOR BASE IPE MODEL QUANTIFICATION

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
1	8.62E-07	39.8	BLAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
2	1.56E-07	7.2	BEAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				ROP2	1.00E+00	FAILURE TO REC OSP; SBO, SHORT TIME AVAILABLE
3	1.46E-07	6.72	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
4	1.03E-07	4.74	MEFGS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
5	8.75E-08	4.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
6	7.04E-08	3.25	BLAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				ROP2	5.09E-02	FAILURE TO REC OSP
7	6.84E-08	3.15	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
8	5.79E-08	2.67	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
9	5.35E-08	2.47	TEABS	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				CRD	1.00E+00	EVENT FAILS
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
10	5.22E-08	2.41	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
11	4.08E-08	1.88	TEFEB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
12	2.85E-08	1.31	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
13	2.72E-08	1.25	TEFEB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
14	2.43E-08	1.12	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
15	2.04E-08	0.94	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
16	1.89E-08	0.87	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
17	1.56E-08	0.72	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
18	1.43E-08	0.66	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
19	1.35E-08	0.62	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
20	1.31E-08	0.6	TEFEB	GTR	3.87E+00	GENERAL TRANSIENT IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
21	1.05E-08	0.48	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
22	8.73E-09	0.4	TEFEB	GTR	3.87E+00	GENERAL TRANSIENT IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
23	7.27E-09	0.34	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
24	7.15E-09	0.33	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
25	6.92E-09	0.32	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
26	6.46E-09	0.3	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
27	6.38E-09	0.29	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
28	6.09E-09	0.28	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
29	6.04E-09	0.28	ALCEB	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
30	5.80E-09	0.27	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
31	5.71E-09	0.26	AEGGA	LLOCA	3.00E-04	LLOCA IE
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
				LV	2.89E-03	LV FAILS; ALL SUPPORTS AVAILABLE
32	5.67E-09	0.26	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
33	5.28E-09	0.24	LLABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)
34	5.21E-09	0.24	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
35	5.04E-09	0.23	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
36	4.31E-09	0.2	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
37	3.86E-09	0.18	TLABS	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
38	3.29E-09	0.15	LLCOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
39	3.17E-09	0.15	TEFBS	LOIA	1.43E-02	LOSS OF INSTRUMENT AIR IE (EXCL LOSW CONT.)
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				OFW1	8.40E-03	OPTR FAILS TO RSTRT A FW PMP OR RCVR HW LVL W/ MNL MU VLV
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
40	2.91E-09	0.13	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				1IA	1.96E-02	IA FAILS (LOOP, DLOOP)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
41	2.64E-09	0.12	LLBOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
42	2.54E-09	0.12	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
43	2.07E-09	0.1	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
44	2.05E-09	0.09	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				SW	2.85E-02	FAILURE OF SW (DLOOP), 23 UNAVAIL
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
45	2.04E-09	0.09	ALCEB	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	1.00E+00	EVENT FAILS
				SBCS	1.00E+00	EVENT FAILS

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
46	1.85E-09	0.09	TEABS	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
47	1.83E-09	0.08	BLASB	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO OCCURS IN UNIT 1
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
48	1.82E-09	0.08	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				SLC	2.35E-04	SLC FAILS; 1/2 PUMPS; ALL SUPPORTS AVAILABLE
49	1.79E-09	0.08	LLCOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP B SUCCESS
50	1.73E-09	0.08	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
51	1.70E-09	0.08	TEFSB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
52	1.66E-09	0.08	LLCOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
53	1.62E-09	0.07	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				RPT1	1.99E-04	AUTO RPT FAILS; ALL SUPPORTS AVAILABLE
54	1.45E-09	0.07	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.00E+00	EVENT FAILS
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
55	1.43E-09	0.07	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
56	1.37E-09	0.06	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				1IA	1.96E-02	IA FAILS (LOOP, DLOOP)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
57	1.34E-09	0.06	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				AT1	1.28E-02	ATWS1 ACTUATION FAILS
				AT2	1.28E-02	ATWS2 ACTUATION FAILS (GIVEN AT1 SUCCESS)
58	1.31E-09	0.06	BLAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				1ES	1.51E-03	LOSS OF ESS BUS (901-49), 17 & 18 UNAVAIL
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
59	1.29E-09	0.06	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
60	1.24E-09	0.06	MEFBS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				OCNTS	1.20E-02	OPTR FAILS TO INITIATE CONT SPR (1)
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
61	1.24E-09	0.06	MEFGS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
				LV	2.89E-03	LV FAILS; ALL SUPPORTS AVAILABLE
62	1.21E-09	0.06	TEFGS	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
63	1.18E-09	0.05	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	8.70E-03	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (1)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				OAL	1.40E-01	OPTR FAILS TO CONTROL RV LEVEL AFTER ATWS (11)
64	1.14E-09	0.05	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	1.00E+00	EVENT FAILS
65	1.13E-09	0.05	TEFSB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)
66	1.12E-09	0.05	AEGGA	LLOCA	3.00E-04	LLOCA IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				LV	1.00E+00	EVENT FAILS
67	1.03E-09	0.05	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
68	9.87E-10	0.05	TLBSB	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
69	9.86E-10	0.05	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
70	9.74E-10	0.04	LEAHD	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
71	9.71E-10	0.04	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)
72	9.47E-10	0.04	ALCEB	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				DG1	1.37E-01	LOP FROM DG1 TO BUS 14-1 (24 HRS)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
73	9.21E-10	0.04	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)
74	9.09E-10	0.04	LLCOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP B SUCCESS

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
75	8.59E-10	0.04	BLASB	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO IN UNIT 1, NO SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.99E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 1T2 AVAILABLE
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
76	8.26E-10	0.04	BEAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; 1R1, 1T2 AVAILABLE
				RCIC	1.00E+00	EVENT FAILS
				ROP2	1.00E+00	FAILURE TO REC OSP; SBO, SHORT TIME AVAILABLE
77	8.00E-10	0.04	LEACD	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	9.60E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (19)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				ROP1	1.00E+00	EVENT FAILS
78	7.43E-10	0.03	LEAHD	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
79	7.02E-10	0.03	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
80	6.98E-10	0.03	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	8.70E-03	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (1)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
81	6.83E-10	0.03	LLCOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP A SUCCESS

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
82	6.70E-10	0.03	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)
				DGB	1.12E-01	LOSS OF DG1/2 AFTER DG2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
83	6.69E-10	0.03	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
84	6.65E-10	0.03	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				AT1	1.28E-02	ATWS1 ACTUATION FAILS
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
85	6.62E-10	0.03	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
86	6.53E-10	0.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
87	6.49E-10	0.03	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
88	6.40E-10	0.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
89	6.38E-10	0.03	AEGGA	LLOCA	3.00E-04	LLOCA IE
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				CS	5.24E-02	CS FAILS; 13-1, 18, 1M1 AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				LV	5.01E-01	LV FAILS; 18, 19, 1M1 AVAILABLE
90	6.11E-10	0.03	IEBOG	IORV	1.06E-01	IORV + OTHER IEs x RVC
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
91	6.01E-10	0.03	TIGSB	GTR	3.87E+00	GENERAL TRANSIENT IE
				131	1.84E-04	LOSS OF BUS 13-1, 13 AVAIL
				141	1.26E-02	LOSS OF BUS 14-1 AFTER 13-1, 14 AVAIL
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				CST	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
92	6.00E-10	0.03	MLCSB	MLOCA	8.00E-04	MLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)
93	5.87E-10	0.03	MEFGS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				ADS	9.14E-06	ADS FAILS; ALL SUPPORTS AVAILABLE
94	5.85E-10	0.03	TEFHS	GTR	3.87E+00	GENERAL TRANSIENT IE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
95	5.79E-10	0.03	LLBOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	1.00E+00	EVENT FAILS

TABLE 6-4 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
96	5.78E-10	0.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				1IA	4.17E-05	IA FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
97	5.75E-10	0.03	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	5.75E-03	CCST SUCTION VALVES FAIL, CS SUCCESS
98	5.64E-10	0.03	LEBGA	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				19	1.13E-02	LOSS OF BUS 19, 14-1 UNAVAIL
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 13-1, 18, 1M1 AVAILABLE
				LV	1.00E+00	EVENT FAILS
99	5.52E-10	0.03	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
100	5.48E-10	0.03	BLATF	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO OCCURS IN UNIT 1
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)

Notes:
1. "Seq." refers to accident sequence ranking in the top 100 sequences.
2. "Freq." is the frequency per year that this sequence is expected to occur.
3. "Percent" is the percent of total core damage represented by this single sequence.
4. "Damage State" is the plant damage state to which this sequence belongs. The fifth character presents the release associated with this type of sequence and is manually assigned at the end of the analysis in presentations of dominant sequences.
5. "Node" is the list of PRT and support system event tree top events which have failed in this sequence.
6. "Value" is frequency (for initiators) or probability (for failures) associated with each node.
7. "Description" defines the "Node" label.

ENCLOSURE 2

RESPONSE TO OTHER SER ISSUES

CONTENTS

ISSUE 1	**Review of System Success Criteria**
ISSUE 2	**Peer Review**

ISSUE 1: Review of System Success Criteria

The NRC's Staff Evaluation Report raised some concerns related system success criteria. Specifically, it contains the following statement:

> *The licensee used realistic plant-specific analyses based on material access authorization program [sic] calculations for establishing plant-specific system success criteria for each initiator group. Based on the licensee's responses to the RAI, the staff concludes that the Quad Cities IPE success criteria are generally reasonable. The staff, however, believes that additional technical basis is required to support the specific success criteria: room cooling is not required for the electrical components and for the control room and the operators will be 100 percent successful to inhibit automatic depressurization in response to anticipated transients without scram (action assumed to take place, but not modeled).*

Success criteria used in the original IPE for room cooling of electrical components and the control room and for the success of the operator action to inihibit ADS during an anticipated transient without scram (ATWS) event were reviewed. The operator action was added to the ATWS event trees as described in the writeup for Modification 3 in Enclosure 1. No other changes were made for the Modified IPE. Details of the review follow.

Electrical Component Room Cooling

At Quad Cities, the large AC switchgear and most DC switchgear is located in open areas of the turbine or reactor building. These areas are normally supplied with general area ventilation but do not have specific area coolers. Operating experience has shown that loss of ventilation in the open areas of the turbine building during warm weather is easily mitigated by opening outer doors of the building, especially the large trackway doors. Extended loss of ventilation to the reactor building could result in a controlled unit shutdown due to electrical equipment concerns. However, the evaluation performed for a station blackout event of the electrical equipment in these open areas concluded that temperatures would not rise above 120°F and further evaluation was not required.

The survivability review carried out during the original IPE included consideration of local heating during an interfacing system LOCA (ISLOCA). The calculated temperature for the second floor of the reactor building, where most of the electrical switchgear in the reactor building is located, was found to briefly exceed 210°F. However, the survivability review did not identify a resulting failure to function of this switchgear during the ISLOCA event.

Battery Rooms

The original IPE did not include loss of room heating or ventilation as a failure mode for station batteries. The battery rooms have their own self-contained HVAC units that are checked daily on operator rounds. Temperature in the battery rooms is verified once per shift to be between 70°F and 90°F. Initial temperature for an event is therefore expected to be in this temperature range.

Loss of HVAC for a battery room would result in a slow temperature transient and that could impact the operability of the battery. Design documents for the batteries specify that battery room temperatures should be kept between 65°F and 120°F. Temperatures below 65°F can impact the load-carrying capacity of the batteries and temperatures above 120°F can be a long-term battery lifetime concern. Temporary heating units have been used in the past during extreme weather conditions to maintain battery room temperatures above 65°F. Portable air moving equipment is also available if high temperatures become a concern.

Evaluations for a station blackout event concluded that battery room temperatures would not rise above 120°F and further evaluation was not required. Also, the station blackout procedure QCOA 6100-04 specifically addresses monitoring battery room conditions to ensure they are kept within acceptable bounds.

Because of the procedural controls, loss of room heating or ventilation is judged to have an insignificant impact on the probability of battery failure.

Battery Charger Rooms

Some of the important DC electrical equipment (battery chargers, main battery buses, and turbine building distribution panels) is located in compartments below the battery rooms. These areas are supplied with ventilation air from the east turbine building supply fans. Design documents for the chargers and switchgear specify that equipment in the battery charger room be kept between 65°F and 120°F. Loss of HVAC to the battery charger rooms could potentially lead to overheating of equipment, but would require loss of the supply fans. Loss of the fans is annunciated in the Control Room. Furthermore, high temperatures in these DC electrical equipment compartments would be identified during once-per-shift operator rounds or, in extreme cases, by equipment alarms. Overheating would be addressed by opening the compartment doors to the turbine building and positioning temporary fans, as necessary.

Evaluations performed for a station blackout event concluded that temperatures in the battery charger rooms would not rise above 120°F. Also, the station blackout procedure QCOA 6100-04 specifically addresses monitoring operability of DC systems, so temperature problems within the charger rooms would be noticed and dealt with expeditiously.

Because of the procedural controls, loss of room heating or ventilation is judged to have an insignificant impact on the probability of DC equipment failure.

Control Room and Auxiliary Electrical Equipment Room

The Main Control Room and the Auxiliary Electrical Equipment Room are normally cooled by the "A" train of HVAC. If the "A" train fails, a redundant "B" HVAC train will automatically start and take on the cooling loads in these two areas. Cooling water for the HVAC chillers is normally supplied by service water. However, if service water is lost, the "B" train HVAC chiller can be lined up to either Unit 1 or Unit 2 RHR service water. Except for station blackout events, therefore, loss of Main Control Room ventilation is unlikely.

Complete loss of cooling to these areas would be addressed by opening panel doors and room doors to the service building and by using temporary fans and generators, if required. These actions are proceduralized in QOA 5750-15, "Complete Loss of Control Room HVAC." Equipment for this procedure is pre-staged as a part of the Station's response to station blackout requirements. The station blackout evaluation concluded that these actions were sufficient to keep equipment temperatures below 120°F.

Because of the procedural controls, loss of Main Control Room ventilation, even for an SBO event, is judged to have an insignificant impact on modeled equipment failure rates or on modeled human error probabilities (HEPs).

Safe Shutdown Makeup Pump Room

The Safe Shutdown Makeup Pump (SSMP) room contains 4KV Bus 31, Transformer 30, 480 V Bus 31, and various instruments and controls for operation of the SSMP. The room has an independent cooling system that uses service water or fire protection water as a heat sink for the chiller. Failure of the SSMP cooling system could threaten pump operability in the event of extended pump operation with the room doors closed.

Table 4.2.2-8 and the notes for the table in the original IPE submittal stated that the SSMP has a delayed dependence on the room cooler. A revised calculation performed since the cutoff date for the original IPE has shown that the room coolers are required for normal operation of the SSMP if the room doors are kept closed. This has prompted the Station to now declare the SSMP inoperable if the room cooler is unavailable. No detailed calculations exist to support operation of SSMP with the room doors open, as would be possible for a plant transient (but not for an Appendix R fire, the original reason for installing the SSMP). To be consistent with current Station policy on the SSMP room cooler and the assumption in the original IPE on normal configuration of room doors (i.e., closed) for RHRSW vaults and diesel generator rooms, the success criteria for the SSMP will be modified in a future PRA model update to include the requirement for successful operation of the room cooler for SSMP success. Addition of basic events for room cooler failure in SSMP fault trees will be considered for this future PRA update. For the Modified IPE, SSMP unavailability includes unavailability of the room cooler.

Operator Action to Inhibit Automatic Depressurization During Anticipated Transient Without Scram (ATWS) Events

In the original IPE, the operator action to inhibit ADS was explicitly modeled in the small and medium LOCA event trees. The assumption in the original IPE that operator action would be 100% successful in inhibiting automatic depressurization during other events was reviewed. This assumption was found to be conservative for all other initiating events except ATWS. The ATWS PRT has been revised in the Modified IPE to include a node for this operator action as described in the writeup for Modification 3 in Enclosure 1.

ISSUE 2: Peer Review

ComEd committed to obtain a peer review of the Modified Quad Cities IPE. During the Quad Cities IPE modification process, ComEd has employed two "peers" to review the changes being made to the Quad Cities IPE model.

Mr. Robert C. Bertucio

Mr. Robert C. Bertucio of NUS was employed to perform a peer review of: 1) modifications of success criteria; and 2) the final Modified Quad Cities IPE quantification and key results.

Mr. Bertucio, whose detailed resume is available upon request, has more than twenty years experience in the field of probabilistic safety analysis and risk assessment. His experience includes involvement in numerous utility, industry, and government programs associated with Individual Plant Examinations. His previous experience in PRA developmental programs such as IREP, IDCOR, and NUREG-1150 provide him with understanding and insight of the current capabilities of PRA technology and expected risk characteristics of reactor facilities. His utility IPE involvement includes significant work on the Point Beach, Surry, North Anna, Palo Verde, Brunswick, and Indian Point 3 IPEs. He previously performed a peer review of the Modified Zion and Modified Dresden IPEs.

During his review of the Modified Quad Cities IPE, Mr. Bertucio asked a number of questions related to initiating events, initiating event frequencies, success criteria, and the dominant core damage sequences. These questions were answered to his satisfaction and required minor changes to the Modified Quad Cities IPE. More detailed documentation of his review and acceptance of the final modifications is available upon request.

Dr. Gareth W. Parry

An industry recognized HRA expert, Dr. Gareth W. Parry, formerly of NUS, had previously been retained to train ComEd analysts in an alternative HRA methodology to support the Modified Zion and Modified Dresden IPEs. Dr. Parry was retained to oversee the ComEd application of that methodology to Quad Cities, and to perform a detailed peer review of its use and of the results.

The alternative HRA used the EPRI Cause Based Decision Tree methodology to estimate the cognitive failure probability (P_C) of each modeled human interaction. This methodology was authored by Dr. Parry and published after much of the initial ComEd IPE effort had been completed. The execution failure probabilities (P_e) were estimated using an evaluation process recommended by Dr. Parry based on THERP data.

Dr. Parry reviewed the modeling and calculations of the operator actions as well as the plant response trees for identification of operator action mode dependencies and calculation of conditional failure probabilities.

Dr. Parry's major comments, recommendations, and corrections (not only those changes that resulted in human error probability increases, but also many changes that resulted in decreases) were incorporated into the revised HRA as they were identified.

ENCLOSURE 3

REVISED PAGES

FOR

BASE QUAD CITIES IPE SUBMITTAL REPORT

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QUAD CITIES NUCLEAR POWER STATION
UNITS 1 AND 2

INDIVIDUAL PLANT EXAMINATION
SUBMITTAL REPORT

VOLUME 1

REVISION 1

AUGUST 1996

Submitted By

COMMONWEALTH EDISON COMPANY

726316SU.2CV/121593

MAIN SUMMARY REPORT

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EXECUTIVE SUMMARY OF THE MODIFIED QUAD CITIES INDIVIDUAL PLANT EXAMINATION

OVERVIEW

The Quad Cities Individual Plant Examination (IPE) conducted by Commonwealth Edison and the Individual Plant Evaluation Partnership (IPEP) demonstrates that no severe accident issue requiring remedial action exists. The IPE results are well within the safety goals established by the Nuclear Regulatory Commission (NRC). The IPE concludes that Quad Cities Station functions well within accepted safety limits due to safety margins incorporated in the original design and to the effectiveness of the emergency procedures.

The following paragraphs present more detailed information on the features and results of the IPE.

FEATURES OF THE QUAD CITIES IPE

The Quad Cities IPE is a Probabilistic Risk Assessment (PRA) study which has been integrated with an Accident Management Program. During each step of the study, there was a systematic search for insights to identify plant characteristics that are good "as-is," as well as to identify potential enhancements for improving plant safety relative to severe accidents and, in the future, for developing a Severe Accident Management Program. The study employed realistic, best-estimate analyses and realistic treatment of operator actions.

The Quad Cities IPE incorporated a number of significant innovative features such as the following:

- The development of plant response trees (PRTs). These improvements on traditional event trees, which trace a sequence of events and subsequent actions, permitted an evaluation of the total plant response to a severe accident. Because this methodology considers the total plant response, the interface between the core damage analysis and the containment analysis is fully integrated.

- The full integration of the Quad Cities General Abnormal (QGA) procedures and the other plant operating procedures. The accident progression reflected a realistic operator response and its impact on the accident consequences. Alternative recovery strategies, already included in the QGAs, were considered as well as the total capabilities of the plant, rather than just the capabilities of the dedicated safety systems.

- The development of realistic success criteria for systems and operators based on many transient calculations which utilized a CECo-specific version of the MAAP computer code. These computer analyses defined the minimum system functional requirements and the time windows for successful operator action.

- The introduction of a Success with Accident Management (SAM) endstate to track and collect sequences that would progress to core damage well after the traditional 24 hour evaluation time of PRAs. In traditional PRAs, these sequences would be grouped with all of the other "success" sequences which are in a safe, stable state before 24 hours. However, Commonwealth Edison did not want to lose information to be gained from these sequences in regard to the development of an Accident Management program.

The original Quad Cities IPE represents the plant as of the cut-off date of July 1991. The Modified IPE incorporates updated failure data for five key plant systems for the three year period of 1993 through 1995. Maintenance unavailability data for these same five systems was collected for the most recent two year interval (1994 and 1995). Further, the revised Human Reliability Analysis for important operator actions from the original IPE models were based on operating procedures in place as of December 1995. Additional changes in plant design or operation since July 1991 which may affect the risk profile will be evaluated as part of the periodic review and update of the Quad Cities PRA - the "living PRA" process.

QUAD CITIES MODIFIED IPE RESULTS

Two basic measures of severe accident risks were employed for these studies:

- The frequency of damage to the reactor core in any given year (or core damage frequency, referred to as CDF). CDF is expressed as "chances" of core damage per year of reactor operation.

- The large, early release frequency (referred to as LERF) as defined in the NEI-sponsored PSA Applications Guide (EPRI TR-105396) published in August 1995. ***Large*** means that the event results in "the rapid, unscrubbed release of airborne aerosol fission products to the environment." ***Early*** means that the event occurs "before the effective implementation of the off-site emergency response and protective actions."

The IPE study produced the following statistics which provide a better appreciation of the high level of safety provided by the station design and operating practices:

- The CDF is 2.17×10^{-6} per year or once in 461,000 years of operation.

- The LERF is 3.74×10^{-8} per year or once in 26,700,000 years of operation.

The following summarizes important characteristics of the Quad Cities Modified IPE analysis:

- Of the total CDF, over 91% is due to five initiating events.
  - The Dual Unit Loss of Offsite Power initiator contributes 57%.
  - The Single Unit Loss of Offsite Power initiator contributes 12%.
  - The General Transient initiator contributes 11%.
  - The Anticipated Transient without Scram (ATWS) initiator contributes 8%.
  - The Medium Loss of Coolant Accident (MLOCA) initiator contributes 5%.

- Fourteen accident sequences have individual contributions to the total CDF exceeding 1%.
  - A single sequence initiated by a Dual Unit Loss of Offsite Power and in which the diesel generators fail to function contributes 40% of the CDF.
  - All other sequences have individual contributions of less than 8%.

- Containment failure or venting within 24 hours is associated with 67% of the total CDF.

EVALUATION OF RESULTS

The Commonwealth Edison IPE/AM program has identified 164 Quad Cities-related IPE and accident management insights. IPE insights deal with plant procedures, hardware, training, information, and test/maintenance. Accident management insights address issues involving Accident Management strategies, organization, training, computational tools, and information systems. Most insights are fairly minor in significance.

A review of the modified IPE results against NUMARC Severe Accident Issue Closure Guidelines (NUMARC 91-04, January, 1992), reveals that with the exception of the Class IB sequences, all the accident sequence groups fall below the Severe Accident Closure Guidelines frequency cutoffs.

The Class IB sequences (Station Blackout) exceed the percentage cutoff value specified by the guidelines and just exceed the lowest frequency limit requiring action. Recently, additional diesel generators have been added at the station; one new diesel generator has been added at each unit. These diesel generators will significantly reduce the likelihood of station blackout and will be included in an update to the Quad Cities PRA model.

CONCLUSIONS

The IPE took several major steps toward injecting more realism into the evaluation for severe accidents at Quad Cities Station. These included the following:

- Integration of Level 1 and Level 2 analyses using plant response trees
- Use of best-estimate success criteria
- Implementation of the control room operator QGAs into the accident evaluation

The realistic modeling employed shows that the design of Quad Cities Station is very good. The QGAs are effective in responding to severe accidents, and they contribute to Quad Cities Station's low core damage frequency (2.17×10^{-6} per year). Quad Cities Station was found to have no serious weaknesses or vulnerabilities.

The Commonwealth Edison engineering staff has been intimately involved in the IPE process and has acted as both originator of IPE analyses and reviewer of all IPE analyses. As a result of the Integrated IPE/AM Program, the CECo PRA staff has developed a unique understanding of the behavior of the plant under accident conditions and of the total plant capabilities to respond to accidents.

The principal purpose of the Quad Cities IPE was to develop an understanding of the response of the plant to severe accidents. It accomplished this purpose. A second purpose of the Quad Cities IPE was to serve as the basis for an Accident Management program. The insights developed during performance of the Quad Cities IPE will form the basis for future development and implementation of the Quad Cities Accident Management program. The final results of the study support the idea that the best improvement for plant safety is a good Accident Management program.

1.0 SUMMARY OF THE QUAD CITIES IPE

This section provides a summary of the Quad Cities Individual Plant Examination (IPE); all of the information presented in this section can be found in greater detail in subsequent sections of this document.

1.1 Philosophy and Conformance with GL 88-20

The Quad Cities IPE has been performed to identify and resolve severe accident issues germane to Quad Cities Station. To assure that this purpose was accomplished, CECo performed a full-scope Level II Probabilistic Risk Assessment (PRA), into which Accident Management (AM) considerations were fully integrated.

Commonwealth Edison Company (CECo) conducted the Quad Cities Level II PRA to be in full compliance with the requirements of NRC Generic Letter 88-20 and its Supplement 1. CECo's approach to the IPE has been to perform realistic evaluations of Quad Cities Station's capability with emphasis on the prevention of severe accidents and on the need to effectively respond to accident sequence progression in the event of a severe accident. CECo's evaluations were carried out in a manner that supported senior management decision-making processes, relative to potential enhancement of plant design and/or operation, aimed at reduction of risk from severe accidents.

Integrated throughout the IPE was the development of insights and information that either suggested plant improvements, or which evolved into the framework of an accident management program for Quad Cities Station. In performing the IPE, standard PRA systems analysis practices such as those outlined in the PRA Procedures Guide (NUREG/CR-2300) were used. The Quad Cities IPE employs the large event tree/support state method. An innovative approach to integrating the traditional systems analysis and containment analysis portions of the PRA was used that involves the development of combined, fully integrated, event trees referred to as Plant Response Trees (PRTs). The methods employed were presented to the NRC during a series of technical exchange meetings which took place during 1991 and during the Zion IPE Stage II review in September 1993.

The focus of the investigation was on the realistic assessment of the plant response to potential accident sequences, so that insights feeding CECo's accident management program represented CECo's best understanding of the plant response. The Quad Cities IPE specifically models the Quad Cities emergency operating procedures, which are based on the generic Boiling Water Reactor Owners Group (BWROG) symptom-based guidance. The success criteria used to determine whether or not plant systems achieve their intended safety function was realistically determined for each important type of accident sequence rather than relying on the Quad Cities FSAR (Final Safety Analysis Report) success criteria.

These success criteria considered both equipment capability and timing of the accident progression. Well-known, detailed approaches for common cause failure and human error that supported the conduct of realistic studies were adopted for the Quad Cities IPE. Special attention was also given to the treatment of dual-unit site issues.

1.2 Project Organization

Commonwealth Edison Company engaged the Individual Plant Evaluation Partnership (IPEP) to support the analysis efforts on the Quad Cities IPE and the IPE's for CECo's other nuclear generating stations. The IPEP companies are Westinghouse, Fauske and Associates, Inc. and TENERA. CECo created an organization for the performance of these projects which effectively utilizes its personnel resources and provides CECo with complete control and involvement in the analysis of each plant. The CECo personnel assigned to conduct the IPE program collectively have extensive experience in plant operations and systems engineering, as well as PRA experience. Many of the methods used in the Quad Cities IPE were originated by CECo. IPEP personnel performed the basic modeling and analysis, while CECo personnel performed success criteria analysis using a CECo-specific version of the Modular Accident Analysis Program (MAAP) code and conducted detailed reviews of the models, assumptions, and results.

Interactions between CECo personnel and the IPEP analysts were conducted on a continual basis and intensively at each intermediate step to resolve CECo comments and incorporate plant-specific knowledge. Figure 1.2-1 shows the overall organizational structure for the CECo IPE program. Insights developed during the performance of the PRA were evaluated by a "Tiger Team" of experienced IPEP and CECo personnel. Key insights and key results from each stage of the study were also reviewed by an IPEP Senior Management Support Team (SMST). The SMST consisted of a senior manager from each IPEP company who was not involved in the day-to-day conduct of the IPE. In addition, CECo senior management actively reviewed all results and insights as well as the IPE program team's recommendation to decide which of the insights and/or recommendations to pursue. As noted in the initial CECo response to the Nuclear Regulatory Commission (NRC) on Generic Letter 88-20, no separate "independent review" of the base Quad Cities IPE was performed. It is CECo's view that the quality of the study is assured by the employment of knowledgeable, experienced analysts both at IPEP and at CECo, as well as the many levels of review within the CECo program. The changes made in the generation of the modified Quad Cities IPE received an independent, peer review.

1.3 Methodology

This section summarizes the overall PRA methodology used for the Quad Cities IPE/AM Program.

1.3.1 Overall Model

The IPE was conducted using standard analysis practices, such as those outlined in NUREG/CR-2300, "PRA Procedures Guide - A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants" and NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide." However, innovative techniques were developed for several areas of the analysis. The traditional systems analysis and containment analysis portions

FIGURE 1.2-1

CECO IPE/AM PROGRAM ORGANIZATIONAL STRUCTURE

CECo IPE/AM Program Manager
Bruce Momsen

IPEP SMST	IPEP Tiger Team	CECo Quad Cities PRA Group	CECo Support	Quad Cities Site Interface
Nick Liparulo	Bruce Monty	Robert Harding*	George Klopp*	Mike MacLennan*
Bob Henry	Bob Hammersley	Leland Raney*	James Hawley	
John Raulston	Tim Andreychek	Milad Kalache	Kong Wang	
Mike Hitchler	Jim Carter	Paul Knoespel*	Douglas Swartz	
		Phil Cretens		
		Raymond Christensen*		

IPEP Program Manager
Bob Henry

Containment Assessment\
Accident Management Support
Bill Berger
Frank Highland

IPEP Quad Cities
Project Manager

Ed Krantz

Mark Mansell, Data Collection & Analysis Task Leader
Bob Buell, PRT Development Task Leader
Jack Trainer*, Accident Management Coordinator
James Hawley, Containment and Source Term Task Leader
Ed Krantz, Accident Sequence Quantification Task
Marty Hinton, System Analysis Task Leader

* Also Members of the Tiger Team.

of the PRA were fully integrated by plant response trees that depict the combinations of interactions that can impact the plant behavior from the initiating event to an endstate characterized by retention of fission products within the containment boundary or release to the environment. A CECo-specific version of the MAAP computer code was utilized to characterize success criteria, timing and containment response.

The models developed in the IPE represent, with minor exception, the as-built, as-operated Quad Cities Station as of a cut-off date in July 1991. The Modified Quad Cities IPE includes updated failure data for five key plant systems (HPCI, RCIC, Emergency Diesel Generators, Safe Shutdown Makeup Pump, and RHR) for the three year period of 1993 through 1995. Maintenance unavailability data for these same five systems was collected for the most recent two year interval (1994 and 1995). These time spans were judged to provide the best indication of recent performance for these key systems.

Additional data for the Modified IPE was also gathered to search for pre-initiator type failures involving calibration of pertinent instruments. Records of events were gathered for a five year period through the end of 1995.

For the Modified IPE, the significant operator actions described in the Quad Cities General Abnormal Emergency Operating Procedures (QGAs) in place as of December 1995 were re-evaluated utilizing a newer alternate HRA technique. Care has been taken to ensure that only formal procedures, which the operators are trained to use, have been credited.

The key tasks in the overall IPE model are described below:

- Plant familiarization was accomplished by the analysts through a review of the Quad Cities Updated Final Safety Analysis Report (UFSAR), design drawings, design descriptions, training materials, normal and emergency procedures, technical specifications, test procedures, location and layout drawings, and plant walkdowns.

- Plant specific information was collected from a variety of logs, reports, and operator interviews for the period from January 1, 1985 to December 31, 1991 to examine plant specific component failure, testing and maintenance data, as well as initiating events which have led to reactor trips. Generic data from IEEE-500, NUREG-2815, Revision 1, and other sources were used to supplement the plant specific information. For common cause failure, the Multiple Greek Letter (MGL) method was used to generate failure probabilities.

- The accident initiators were identified from the collection and analysis of plant trip data. This was supplemented by the use of other industry sources, such as NUREG/CR-3862, where Quad Cities plant specific data was insufficient due to low or non-existent frequency of occurrences. Some of the loss of coolant accident initiating event information was derived from WASH-1400. Loss of offsite power and plant centered losses were derived from generic data in NUREG-1032, NSAC-147 and NSAC-166 which are applicable to the dual unit Quad Cities Station.

- Internal flooding was treated as a special initiator. Separate analyses were performed to determine whether there are areas in Quad Cities Station that are susceptible to flooding or spray from pipe breaks and whether there is sensitive equipment in those areas that could cause plant shutdown or result in a failed safety system.

- A detailed analysis of the various front-line safety systems and supporting systems was conducted for each of the identified initiators and for the interactions between the two Quad Cities units.

- Plant Response Trees (PRTs) and support system event trees were used to develop the Quad Cities accident sequence model. A plant response tree was developed for each initiator; a support system event tree model was developed for each major class of initiating events. The support system event tree model was developed as a dual unit model, where appropriate, because the two Quad Cities units share important support systems. A CECo-specific version of the MAAP computer code was used to develop realistic accident sequence models, including success criteria and operator actions, so that the accident sequences represent the best estimate plant response.

- The Quad Cities systems represented in the PRTs were modeled with fault trees. The development of the fault trees was done starting from the success criteria for the system specified in the PRTs. The relationship between the two units was carefully examined and, where appropriate, modeled. The systems modeled include safety systems, support systems, containment systems and miscellaneous systems, as dictated by the PRTs.

- Extensive phenomenological evaluations were made to study accident progression and the possible containment failure mechanisms. These evaluations serve as the primary means by which phenomenological issues were addressed. A combination of these evaluations and CECo-specific MAAP analyses were used to assess the importance of the phenomenological issues and the significance of uncertainty. For some issues, CECo BWR-3/Mark I specific experiments were developed and performed to support the phenomenological evaluations.

- Source terms were developed by analyzing the dominant accident sequences that led to containment failure, using a CECo-specific version of the MAAP code. Source terms were binned into release categories based on type, timing, and magnitude of release.

1.3.2 Initiating Events

The Quad Cities-specific initiating events considered in the IPE are as follows:

- Large Loss of Coolant Accident (LLOCA)
- Medium Loss of Coolant Accident (MLOCA)
- Small Loss of Coolant Accident (SLOCA)

- Interfacing Systems LOCA (ISLOCA)
- Inadvertent Opening of a (Main Steam) Relief Valve (IORV)
- Anticipated Transients
- Single Unit Loss of Offsite Power (LOSP)
- Dual Unit LOSP (DLOSP)
- Loss of 4KV Bus 11 (LB11)
- Loss of 4KV Bus 12 (LB12)
- Loss of 4KV Bus 13 (LB13)
- Loss of 4KV Bus 14 (LB14)
- Loss of 125 VDC Reserve Bus 1B-1
- Loss of Instrument Air (LOIA)
- Loss of Service Water (LOSW)
- Loss of MSIV Room HVAC Due To Loss of 480 V Bus 18 (LB18)
- Loss of MSIV Room HVAC Due To Loss of 480 V MCC 18-2 (L182)

In addition, two events were treated as consequential failures in the accident sequence analysis and thus, no frequencies were calculated:

- Loss of all AC Power (Station Blackout (SBO))
- Anticipated Transient Without Scram (ATWS)

The LOCA frequencies were taken from WASH-1400 for this analysis. The interfacing system LOCA frequency was determined by a Quad Cities specific calculation considering all likely flow paths. The frequency of an inadvertent opening of a relief valve was determined from plant-specific data on such events.

Transient events were identified through BWR operating experience. The steps taken to create a database of transient initiating events and make them specifically applicable to Quad Cities Station include the following:

- The trip history was reviewed to identify events that have occurred at Quad Cities Station.
- Data from NUREG/CR-3862 was used to supplement historical Quad Cities anticipated transient data.
- The results of plant systems analyses were utilized to identify potential initiating events.

The general transient frequency is the sum of anticipated transient frequencies for Quad Cities Station. The NUREG/CR-3862 anticipated transient categories relevant to Quad Cities Station are grouped as one initiating event, with the exception of LOSP which was considered as a special initiator.

The loss of the heating, ventilation, and air conditioning (HVAC) system for the outboard MSIV Room, loss of service water, loss of instrument air, and loss of 4 KV buses that could lead to a reactor trip were included as special initiating events in the Modified IPE based upon Quad Cities Station specific analyses. The initiator frequencies for these events are based on system fault trees and on the potential for loss of the electrical buses due to water spray.

The frequencies for single unit LOSP and dual unit LOSP were calculated separately. The methodology and site specific values developed in NUREG-1032 for grid related losses, weather related losses, and extreme weather related losses were used to calculate the LOSP frequencies. The values for Plant Centered Loss (PCL) were calculated from generic data presented in NSAC-147 and NSAC-166 for LOSP at dual unit sites. The generic PCL frequencies for dual unit and single unit at a dual unit site were used in the Quad Cities specific analysis.

1.3.3 Systems Analysis

To develop an understanding of the contribution of system performance to accident sequences and to quantify the Plant Response Trees, a comprehensive analysis of all key plant systems (from a risk perspective) was performed. This included a plant familiarization activity, a search for dependencies between plant systems, and detailed fault tree analysis for each key system.

To ensure the IPE accurately represents how the plant's systems contribute to the overall risk profile, a thorough understanding of key frontline and support systems is essential. Prior to the development of the fault trees, a comprehensive evaluation was performed for each system, which included collection, evaluation, and documentation of information. Included in this documentation are the important dependencies, instrumentation and control requirements, and the results of a review of equipment maintenance and surveillance practices. A plant walkdown was used to verify that the plant configuration modeled in the IPE is consistent with the manner in which the individual systems are installed and operated. The results of an operating experience review are also documented, to be sure that plant specific operating experience is reflected in the model development and in the quantification of system and component performance parameters.

Because Quad Cities Station is a dual unit site, a careful examination of the documentation for both unit's systems was performed. Any key differences were identified and documented. Shared systems or shared components were identified, including the type of sharing (total or partial) and any preferential alignments. Any unit-to-unit cross-ties, along with the normal alignment and emergency alignment capabilities, were identified. Plant procedures, operator training manuals, and plant administrative policies were reviewed concerning such shared and cross-tied systems to be sure accurate modeling was performed in the IPE and that the full plant capabilities were understood from an accident management perspective.

Any configuration or uses of a system which could be important in accident management were identified and documented. Examples of this type of information include the identification of other systems that could fulfill the same function, instrumentation that might be beneficial in restoring certain systems, equipment access pathways and location, etc.

The systems modeled in the Quad Cities IPE are:

- Reactor Core Isolation Cooling System
- Safe Shutdown Makeup Pump System
- Feedwater and Condensate System
- Control Rod Hydraulic System
- High Pressure Coolant Injection System
- Automatic Depressurization System
- Residual Heat Removal System including its service water, suppression pool cooling and containment sprays
- Core Spray System
- Turbine Building Closed Cooling Water System
- Common Actuation including components actuating HPCI, RHR and CS
- Containment Vents including both the Torus and Drywell vent paths
- Service Water System
- Electric Power System including AC, DC and diesel generators

Fault trees were used to model the performance of plant systems in the Quad Cities IPE. These fault tree models depict the various combinations of hardware faults, human errors, test and maintenance unavailabilities, and other events that can lead to a failure to perform a given safety function. The definition of success for each fault tree is determined by the success criteria established for each PRT heading involving system performance.

Fault trees were developed for both frontline, containment, miscellaneous and support systems. Their analysis is conditional on both the initiating event (and its effects), and the availability of support systems that impact system operation.

1.3.4 Support System Modeling

The "support state methodology" was used to model the key support systems and their impact on the safety systems that are required to respond to the modeled initiating events. The concept of a support state model allowed the major support systems to be modeled outside of the accident sequence plant response trees. Quad Cities Station contains two units which share major support systems. Shared systems were modeled to ensure that the influence on both units was captured.

A support system is defined as a system that is depended upon for the successful operation of frontline systems, safety systems, miscellaneous systems or other support systems. The support systems were identified by reviewing the Quad Cities Station Updated Final Safety Analysis Report (UFSAR), system descriptions, and piping and instrumentation diagrams.

The second step in the development of the support state model was a review of the system dependency matrices. The dependency matrices were developed to identify the interrelationships among the various systems modeled in the IPE and focus the investigation on key dependencies between initiating events, support systems, and frontline systems for major system components. The dependencies considered in the development of these matrices considered partial dependence as well as complete dependence.

The third step in developing the support state model was to identify the key support systems. The key support systems are those systems which interact with the other frontline and support systems. The criteria for identifying key support systems include whether the system supports multiple frontline systems and whether the frontline systems would not function without the support system (further discussion of the selection of key support systems is provided in Section 4). Based on the review of system descriptions and the system dependency matrices, the following key support systems were selected for modeling in the support system event trees:

- Electrical Power - DC
- Electrical Power - AC
- Common Actuation System - CAS
- Service Water - SW
- Turbine Building Closed Cooling Water - TBCCW
- Instrument Air - IA

The AC and DC electric power systems provide the motive or control power for a majority of the safety-related pumps and valves. The CAS provides the actuation signals for the safety systems on an ECCS signal. The SW system provides the ultimate heat sink for the cooling of major heat loads, the TBCCW system provides the cooling for the feed and condensate system pumps and other turbine building auxiliaries and the IA system provides motive force for various components in the plant.

Following the identification of the key support systems, the fourth step in the support state modeling process was to identify the possible operating states for each key support system, and from these develop the support system event trees. This step was completed by identifying the possible operational states for each key support system individually. Having identified the various operational states associated with each support system, the states were combined to form a support system event tree (SSET) model. Several support system event trees were developed based on the differing impact of the initiator on the response of the plant.

Vector impact analysis to reduce the number of support states that had to be evaluated was not performed for Quad Cities because the computer code that was used for event tree quantification is capable of quantifying all support model and plant response tree sequences. Each plant response tree was then quantified for each support model sequence.

1.3.5 Plant Response Trees

Plant Response Trees (PRTs) are used to logically model the accident progression of each initiating event through successful mitigation or core damage and containment disposition. The plant response trees were used to define the possible outcomes of each initiating event as determined by the availability of plant systems and the success of essential operator actions. These outcomes, or 'endstates', were then used as part of the IPE process to assess the design and operation of Quad Cities Station.

The plant response tree approach developed for the Quad Cities IPE differs somewhat from the traditional PRA event tree approach (NUREG/CR-2300). The traditional approach consists of two nearly independent analyses, the Level 1 or "front end" analysis and the Level 2 or "back end" analysis. The Level 1 analysis considers the plant systems and models the event progression from initiating event through core damage, while the Level 2 analysis considers the containment safety systems and models the event progression from core damage states through final containment disposition. As a result, separate event trees are developed for each of these analyses. For the Quad Cities IPE, the PRT concept was developed to model the plant response from initiating event through the entire accident progression including the containment response. This logic involves a complete integration of the traditional Level 1 and Level 2 PRA analyses thereby permitting synergistic modeling of the plant.

Additionally, the traditional PRA approach considers only high level operator actions (e.g., initiation of suppression pool cooling) while the PRTs incorporate a direct causal relationship between accident progression and symptom-based operator actions from the Quad Cities General Abnormal Procedures. Also, traditional PRA methods incorporate very conservative definitions of system success which results in a higher likelihood of system failure and, ultimately, in unnecessarily pessimistic overall results. The PRTs incorporate realistic analyses to define success of a system or operator action; thus resulting in a true "best estimate" understanding of risk. In this respect, potential weaknesses are not masked by conservatisms in the analysis.

A final important facet of this integrated PRT approach is the ability to excerpt meaningful accident management insights from an evaluation of the various PRT accident sequences. The coupling of the plant systems, operator actions and containment systems allows a more direct examination of the factors which influence risk. As a result, insights regarding these 'risks' can be developed which aid in the management of a severe accident, in the unlikely event that one occurs.

The plant response tree consists of an initiating event, nodes, accident sequence paths and an endstate for each path. An initiating event was defined as an event which causes plant trip and places some demand on plant safety systems. The nodes are the decision points on the tree and are shown across the top of the tree. These nodes represent success or failure of a plant system or operator action and are ordered to consider the time phasing and hierarchy of cause and effect. The paths, or sequences, are simply the representations of credible combinations of successes and failures of the plant systems and/or operator actions. Ultimately, the product of the PRT is the frequency of these paths. The endstates define the unique set of plant system conditions following the initiating event.

The development of a PRT consists of a number of major, distinct steps. These steps are discussed below:

STEP 1

Define Critical Safety Functions

Preventive actions are required to maintain the plant in a safe, stable condition following an initiating event. These actions can be defined in terms of critical safety functions. Critical safety functions which prevent core damage are defined first; additional safety functions are defined as needed (i.e., post-core damage) to prevent containment failure and minimize fission product releases.

The critical safety functions required to prevent core damage are as follows:

- Reactivity Control
- Reactor Pressure Control
- Reactor Coolant System Inventory Control
- Decay Heat Removal, which consists of:
    - Coolant Inventory Makeup, and
    - Coolant Heat Removal

The critical safety function to prevent containment failure and to minimize fission product releases, if core damage results, is:

- Containment Integrity, which consists of:
    - Containment Heat Removal functions
    - Containment Isolation
    - Radioactivity Scrubbing

STEP 2

Develop Core Damage Prevention Models

These models identify the requisite combinations of systems and operator actions required to bring the plant to a safe, stable condition and prevent core damage. The resulting accident sequences accurately represent the combination of plant systems and operator actions needed to prevent core damage. Only operator actions defined in the Quad Cities emergency procedures are modeled.

STEP 3

Integrate Containment Systems

Containment systems which satisfy the containment critical safety function are included in the PRT in order to determine the containment disposition as well as to consider possible dependencies with other `front end' systems. The integration of the plant systems, operator actions AND containment systems allows treatment of the plant synergistically, as a complete "system".

STEP 4

Endstate Definition

Each initiating event is tracked through its own PRT, evaluating the success or failure of each plant system, operator action and containment system. Each accident path, or sequence, eventually results in a unique 'endstate' depending upon the initiating event and the combinations of success/failure of the nodes addressed. These PRT outcomes, or 'endstates', are then categorized by assigning an identifier. For those paths which end in a long-term safe stable state, the endstate is designated SCS, meaning success. Those paths which result in a successful core state for 24 hours, but which require additional actions or functions to maintain this state in the long term, are designated SAM (Success with Accident Management). Finally, those sequences ending in core damage are designated by 5-character identifiers to characterize fission product releases.

STEP 5

Definition of Accident Sequence and Success Criteria

Determining the sequence success states is one of the most important tasks in developing the PRT structure. The objective is to determine the combinations of plant systems, operator actions and containment systems that are realistically expected to activate chronologically to prevent core damage and/or maintain containment integrity.

To determine PRT nodal success criteria, detailed information regarding plant functions, plant systems, plant operation, emergency operating procedures, abnormal operating procedures, engineered safeguards features, technical specifications, etc. is necessary. Best estimate thermal hydraulic analyses, using the CECo-specific MAAP computer code, were used to determine success criteria for the aforementioned critical safety functions. These analyses also establish the time available to accomplish the operator actions to prevent core damage and/or containment failure.

STEP 6

Accident Management

The final step in the development of a PRT is the definition of potential accident management enhancements which could mitigate the accident.

As part of the Quad Cities IPE Program, an additional PRT endstate designation (SAM, for Success with Accident Management) was defined in order to highlight those accident sequences which require accident management activities to achieve an ultimate safe, stable state in the period after the initial 24 hours. Traditionally, in PRAs, if core damage had not occurred during the first 24 hours, the endstate was considered a success. In the Quad Cities IPE, endstates in which core damage could occur after 24 hours unless something is done, are categorized separately and assigned the designator "SAM." Consistent with traditional PRA philosophy, the PRT accident sequences designated as SAM are not core damage sequences. However, accident management activities are required to ensure that the plant attains a long term safe, stable state.

1.3.6 Containment Analysis

Quad Cities employs a BWR-3 Mark I containment design. The primary containment consists of a drywell, a pressure suppression pool chamber (torus), and interconnecting vent pipes (downcomer pipes). The primary containment surrounds the reactor pressure vessel (RPV) and the recirculation cooling system and provides the first barrier to offsite radioactivity releases. Any leakage from the primary containment system will go directly to the secondary containment system (Reactor Building). The wetwell or drywell may be vented through either the Standby Gas Treatment (SBGT) system or directly to the station chimney through the 8-inch "hardened" vent.

The design free volume of the drywell (including the pedestal region) is 158,236 ft^3, with a gas space height of 102 feet. The drywell is a steel pressure vessel with a 66′ diameter spherical lower portion and a 37′ diameter cylindrical upper portion. This vessel is enclosed in reinforced concrete for shielding purposes with a two-inch gap (average) between the steel shell and concrete to allow for thermal expansion of the steel shell. The internal design pressures of this structure are 56 psig and -2 psig at 281°F. The ambient drywell atmosphere temperature ranges from 135°F to 150°F.

There are eight circular vent pipes which form a connection between the drywell and suppression pool (wetwell) to control drywell pressurization under accident conditions. The pipes are enclosed in sleeves and are provided with expansion joints (bellows) to accommodate differential motion between the drywell and the wetwell. These pipes, in turn, are connected to a toroidal vent header contained in the airspace of the wetwell. Projecting downward from the header assembly are 96 downcomer pipes which terminate roughly 4 feet below the surface of the suppression pool water line.

Other than the suppression pool, several other systems exist to control primary containment pressure. The Quad Cities design implements the following systems to aid the suppression pool in containment heat removal:

1. The Residual Heat Removal (RHR) pumps can be aligned to discharge to either the drywell or wetwell spray headers. These pumps can alternately be used in conjunction with the RHR heat exchangers to provide suppression pool cooling.

2. Operators are instructed to initiate all available drywell coolers if the drywell gas temperature reaches 180°F.

3. Drywell or wetwell venting is performed as a means of primary containment pressure control.

A Quad Cities containment fragility curve was produced (see Section 4.3). Several observations and conclusions are immediately evident:

1. Low pressure failures are dominated by the drywell head closure, which follows directly from the high degree of uncertainty associated with this critical location. Not until above 95 psig, where containment failure

probability is around 15%, does any other component contribute significantly to the total failure probability.

2. The mean failure pressure of the Quad Cities containment is shown to be approximately 105 psig, with the drywell head closure and bellows assemblies contributing approximately 20 and 30%, respectively.

3. If containment fails at relatively high pressures, the likely location will be the vent line bellows. These components have the lowest mean failure pressure and all eight of them were factored into the analysis.

4. The failure probabilities increase rapidly between 100 and 110 psig and reach steady values above 115 psig. The final asymptotic failure probabilities for each of the critical locations are approximately as follows:

   - Vent line bellows 75%
   - Drywell head closure 22%
   - Wetwell shell 2.3%
   - Drywell shell 0.5%

The containment studies reviewed in conjunction with constructing the Quad Cities fragility curve generally considered containment pressure loadings applied at relatively low temperatures (i.e., near the design limit 281-350°F). In the temperature ranges beyond 281°F, the high pressure performance of the containment is expected to degrade due to reductions in material strength and seal properties. For example, silicone seals begin to fail at a projected temperature of 500°F. At this temperature, the drywell head closure would be the most likely failure location. Thus a figure was constructed to show the temperature effects on the ultimate pressure capacity of the containment.

Source term analyses are performed following accident sequence quantification and designation of PRT endstates. The purpose of the source term analysis is to quantify the radionuclide release characteristics for core damage accident sequences. The source term analysis includes the specification of containment failure timing and fission product release magnitude. Source term analysis was performed with the CECo-specific version of MAAP 3.0B referred to as CECo-MAAP Rev. 7.03B.

Since assumptions regarding key severe accident phenomena may dictate the analysis outcome, due consideration of phenomenological uncertainties is a cornerstone of the CECo IPE approach to the containment and source term analysis. The CECo IPE methodology addresses the phenomenological issues in two ways, 1) plant-specific phenomenological evaluations, and 2) CECo-specific MAAP sensitivity studies. This approach provides a bounding assessment of source term release timing and magnitude.

Phenomenological Evaluations

Quad Cities-specific phenomenological evaluation summaries are a principal means of addressing the impact of phenomenological uncertainties on plant response. These summaries address a wide range of phenomenological issues and provide an in-depth review of plant-specific features which influence the uncertainty, or act to mitigate, the consequences of such phenomena. The phenomenological evaluation summaries investigate both the likelihood of occurrence and the probable consequences of key severe accident phenomena.

Sensitivity Studies

The purpose of the sensitivity studies is to determine which remaining phenomenological uncertainties have a significant impact on the likelihood or timing of containment failure and the magnitude of the source term release. In performing Quad Cities deterministic calculations, a limited number of model parameters are investigated with respect to the influences of modeling uncertainties on the radionuclide source terms. In particular, uncertainties in the various physical processes were considered as documented in the IDCOR/NRC issue resolution process. The various phenomena and the uncertainties are described in several NRC and EPRI documents (e.g., NUREG-1335, EPRI TR-100167) and in the IPE Generic Letter 88-20 (including supplements).

1.4 Supporting Analysis

The following sections describe several analyses that support the quantification of the fault trees and the plant response trees. These supporting analyses include the generation of plant specific and generic component data, the generation of human error probabilities, the generation of plant specific common cause failure probabilities, the identification of any internal flooding initiating events, and the analysis of equipment survivability under the expected accident conditions.

1.4.1 Data Analysis

The purpose of the data analysis task was to collect data and obtain realistic estimates of the failure rates and unavailabilities of basic components of the IPE. Random failure rates (including failure probabilities per demand), unavailabilities due to maintenance, and common cause failure rates were the basic quantities that were evaluated extensively in the data analysis task.

At the onset of the data collection task, important key components were identified as "likely to dominate" or "have an important impact on" core damage frequencies, based on knowledge of previous PRAs. The list of key components for the Quad Cities IPE defined the scope of the intensive phase of the plant-specific data collection effort. The key component approach permitted resources to be focused on the most important failures and unavailabilities. Failure and unavailability data for non-key components was obtained from generic data sources.

The failure and component unavailability data collected for the original Quad Cities IPE spanned the period of January 1, 1985 through December 31, 1991. The 7 year period (i.e., 1985 through 1991) gives failure rate and unavailability results that come the closest to the current true state of unreliability of the key components. Plant-specific data was collected from the operating records of both units and was combined to form one data base. No significant differences between the components of Unit 1 and Unit 2 were identified; therefore, no basis was found for pursuing the hypothesis that the unreliability of Unit 1 components could be different from the unreliability of Unit 2 components.

For the Modified IPE, new failure data was collected and analyzed for five key plant systems (HPCI, RCIC, Emergency Diesel Generators, Safe Shutdown Makeup Pump, and RHR) for the three year period of 1993 through 1995. Maintenance unavailability data for these same five systems was collected for the most recent two year interval (1994 and 1995). These time spans were judged to provide the best indication of recent performance for these key systems. Selection of these time spans is also consistent with the methodology being used for the ComEd plant PRA model updates.

Additional data for the Modified IPE was also gathered to search for pre-initiator type failures involving calibration of pertinent instruments. Records of events as preserved in the DVR (Deviation Report) and PIF (Problem Identification Form) databases were gathered for a five year period through the end of 1995. No "pre-initiator" type failures were identified. The review for the Modified IPE concluded that use of generic failure rates and mission times based on calibration interval was appropriate for standby instruments and adequately accounts for calibration failures due to human error.

Failure rates were calculated as point-estimate values. An hourly failure rate is defined as the number of failures that occur during a particular period of component operation divided by the operating hours of the component. This type of point estimate was used to calculate the failure rates of pumps and diesel generators failing to run. The demand failure rate is the number of failures during a particular period of time divided by the number of component demands that occurred during the same period. This type of point estimate was used to calculate the failure rates of components failing to start, and motor-operated valves failing to open or close.

The boundaries of each component were also considered in the screening of failures and maintenance events. For example, circuit breakers and handswitches were included within the boundaries of pumps, and failures of the subcomponents were counted as failures of the pump.

NUREG/CR-2815 was the primary source of generic failure rate data. NUREG/CR-2815 was the first source consulted and was used except in cases where it did not provide data for the particular failure mode needed or where some other source was determined to provide more relevant data. NUREG/CR-4550 was the primary source of generic maintenance unavailability data. Generic component failure data from NUREG/CR-4550 was also used for producing a Bayesian update estimate for several pieces of equipment that had not experienced a failure during the three year data collection period for the

Modified IPE. Generic data were obtained from other industry sources for use in this task, including IEEE Std. 500-1984 and WASH-1400.

Testing was found to affect the unavailability of only a few systems analyzed (High Pressure Coolant Injection and Reactor Core Isolation Cooling systems). These unavailabilities were calculated from the test frequencies and their average durations based on Quad Cities-specific experience and documentation.

1.4.2 Success Criteria

For the Quad Cities IPE/AM project, a large number of plant specific analyses were performed to define the 'success criteria' for the Quad Cities model. These analyses were performed using computer codes and hand calculations.

To develop the success criteria, the following definitions of success related to core cooling (prevention of core damage) and containment integrity (prevention of containment failure) were used.

Core Cooling Success

Core cooling is defined as being successful if the hottest fuel temperature never exceeds 4040°F. This temperature corresponds to the melting temperature of the U-Zr-O eutectic formed during core degradation.

Containment Integrity Success

If the containment pressure exceeds the allowable pressure at the given drywell shell temperature, containment failure is assumed, and release of fission products from the containment, beyond that associated with normal leakage, is initiated.

Using the broad definitions of successful core cooling and containment integrity, the success criteria for systems, components and operator actions were developed. These success criteria can be grouped into support systems, PRT systems, operator actions and equipment survivability.

Support System Event Tree Model Development

Analyses performed previously by CECo were used to show that the failure of the reactor building HVAC system does not generally lead to the failure of equipment modeled in the PRTs and therefore, does not need to be included in the Support System Event Tree. Some components do require auxiliary cooling and such cooling is considered within the fault trees.

Systems Analysis and Plant Response Tree Development

Extensive analyses were performed to support the development of the PRTs. These analyses determined which systems and combinations of systems are required to prevent core damage and containment failure, and the specific success criteria for the different systems in each sequence on the PRTs. The analyses also were used to determine the mission times to be used in the systems analyses. Extensive analyses were also performed to determine the definition of the Loss-of-Coolant break ranges.

Human Reliability Analysis

The CECo-specific version of the MAAP code was used to develop realistic times available to complete operator actions modeled in the PRTs, based on the Quad Cities emergency procedures. This timing information was then used in the HRA analyses.

Equipment Survivability Analyses

Analyses were performed to predict the reactor building response following an interfacing system LOCA outside containment, specifically the rupture of low pressure RHR piping. The CECo-specific version of the MAAP computer code was used to predict plant responses for the equipment survivability evaluations.

1.4.3 Human Reliability Analysis

The original Human Reliability Analysis (HRA) for the Quad Cities IPE utilized the Technique for Human Error Rate Prediction (THERP). The revised HRA implemented a newer alternative technique, the EPRI CBDTM (Cause Based Decision Tree Methodology) from EPRI TR-100259. The methodology estimates the cognitive (Pc) part of the HEPs by evaluating a series of decision trees that evaluate failure mechanisms such as: data availability, attention failure, miscommunication and misreading of data, misleading information, missing or misreading procedure steps, misinterpretation of instructions or decision logic, and deliberate violations.

Estimation of the execution (Pe) portion of the HEPs was based on THERP data. Various tables in Chapter 20 of the THERP manual were utilized, as appropriate, in determining the HEPs. The approach was essentially to review each procedure to identify the critical sub-tasks (i.e., those essential to completion of the task(s)) and to determine whether any potential recoveries (such as verification of flow or valve position, alternative steps accomplishing the same action, revisitation of the step due to a procedure 'loop', etc.) were present in the procedure and would be read in accordance with the procedure format.

The operator actions within each of the Plant Response Trees were analyzed on a sequence-by-sequence basis to identify the conditions of stress, dependency, and availability of recovery opportunities. This sequence by sequence evaluation generated multiple cases for each of the operator actions.

For the Modified IPE, significant operator actions were re-evaluated. Only those actions which had a Risk Achievement Worth of greater than 2.5 and those actions which were added as a result of changes to the Plant Response Trees and Fault Trees received a complete evaluation utilizing the EPRI CBDTM technique. The remaining HEPs were reviewed for reasonableness of values and for the selection of the appropriate value for each branch of the PRTs.

All of the HEPs after the re-evaluation have values above 1.0E-4, except for OHX (Operator action to align cooling to RHR) and OSPC (Operator action to initiate suppression pool cooling), as discussed in Section 4.4.2. Some of the actions changed significantly, some increasing and some decreasing.

1.4.4 Common Cause Analysis

"Common cause" describes multiple failures of functionally identical components due to a single, shared cause. Common cause analysis (CCA) evaluates the effects of these dependencies that may affect the ability of a system to prevent or mitigate a severe accident.

The Quad Cities CCA modeled common cause failures at the basic event level, employing the Multiple Greek Letter (MGL) method as defined in NUREG/CR-4780, "Procedures for Treating Common Cause Failure in Safety and Reliability Studies."

The evaluation of Quad Cities failure data indicated that there had been no common cause events at the Quad Cities site applicable to current maintenance and operating practices. As a result, to more realistically model current experience at Quad Cities, a plant-specific evaluation of common cause failure events was performed. Quad Cities-specific common cause parameters were developed for components that had data available, including the following:

- Circuit breakers
- Check valves
- Diesel generators
- Motor-operated valves
- Fans
- Low-head pumps
- HVAC chillers
- Relief valves
- Pumps

A generic common cause failure database was developed from EPRI NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events", supplemented with events from the September 1990 EPRI draft report, "A Database of Common Cause Events for Risk and Reliability Evaluations".

A four-member expert judgment panel reviewed data from the generic common cause failure database for applicability to CECo plants. The expert panel came to a consensus opinion on each generic common cause event's applicability to Quad Cities, based upon current Quad Cities system configuration, and maintenance and operating practices. Events involving known common cause mechanisms addressed by specific programs in place at Quad Cities were discarded from the database as were common cause events that occurred due to specific system configurations not present at Quad Cities. Events involving common cause mechanisms that have been addressed in general by maintenance or operating practices at Quad Cities were assigned a lesser probability of occurrence based on judgment of the panel. Furthermore, as part of the Modified IPE, a "floor" of 0.01 was imposed on beta values.

An average common cause component group was quantified from a composite of all the common cause failures for all components in the database. Use of the parameters calculated for this average common cause group was extended to components that have no history of common cause failure, but were judged by the analyst to have some potential for common cause failure. The common cause contribution for the following components was calculated using the average MGL values:

- Air compressors
- Batteries and DC power supplies
- Relays, including contacts and coils
- Switches, including temperature, level, and pressure switches
- Dampers
- Explosive, manual and solenoid valves
- Heat exchangers
- Manual pushbutton switches
- Strainers and filters
- Temperature and pressure transmitters and controllers
- Signal comparators
- Transformers
- Buses
- Electrical filters
- Inverters

In general, the components included in this list were judged to be less complex than the components in the database and thought to have less potential for common cause failure mechanisms. Therefore, assignment of the average common cause parameters is judged to be realistic.

1.4.5 Internal Flooding Analysis

The internal flooding analysis was performed to identify potential sources of flooding and spraying from pipe breaks internal to Quad Cities Station, and the event sequences associated with these sources that could potentially lead to core damage. Pipe, tank, and valve ruptures, etc., could lead to flooding and/or spraying of plant equipment, resulting in failures that could trip the reactor and impair the operation of equipment needed to safely shutdown the plant. The impact of the potential flooding/spraying was assessed to assure that all potential core damage sequences of high probability would be identified.

Much of the information needed for the analysis was taken from the Safe Shutdown Report (SSR) prepared in response to the requirements of 10CFR Part 50, Appendix R. The fire zones developed for the SSR were found to be acceptable for use as flooding zones. The list of equipment necessary for safe shutdown developed for the SSR was used for the internal flooding analysis as well.

Additional information was collected during plant walkdowns. This included investigation of the potential flooding and spraying sources, the equipment that would be affected by these sources, the potential for flooding propagation between areas, and flood mitigation features in the various areas. The walkdowns encompassed those areas judged to be of possible significance in terms of core damage potential in a flooding zone screening process.

Flooding events such as pipe, valve, and tank breaks or ruptures are sufficiently infrequent to be unimportant as trip initiators alone. Only if the same flooding event also degrades safe shutdown capability will the potential for core damage become significant.

The information gathered was used to analyze the flooding zones with the potential to result in equipment failures that could lead to core damage. Many zones were found to have drainage adequate to mitigate the effects of any flooding that could affect the zone. The potential for flood propagation to other zones and the potential for water spray to result in equipment failure was investigated. Shielding and distance from potential spray sources was also considered in the evaluation, as well as the qualification of equipment for operation in adverse environments. All of the flooding zones except for the Unit 1 and 2 Turbine Building Condensate Pump Rooms were eliminated from consideration during the qualitative analysis. The frequency for flooding occurring in the Condensate Pump rooms is approximately 1.3E-02 per year. This event would be similar to a loss of feedwater transient which is already considered in the evaluation of transient events. This contribution to the transient initiator is probabilistically insignificant in comparison with the transient initiator frequency.

Although electrical switchgear was not identified as being threatened by flooding, water spray from nearby pressurized piping is possible should a pipe leak occur. An electrical bus or motor control center was conservatively assumed to fail if sprayed by a pipe leak. Potential water spray of electrical switchgear was included in estimating initiating event frequencies for the pertinent special (support system based) initiators.

1.4.6 Equipment Survivability

As part of the Quad Cities IPE, equipment important for prevention of core damage and/or containment failure was evaluated for survivability during the range of accident conditions postulated in the IPE. To accomplish this task, the Quad Cities equipment survivability study was divided into three phases:

Phase I: Support State and Fault Tree Assumptions
Phase II: IPE Conditions
Phase III: Accident Management/Core Damage Conditions.

For Phase I of the study, the assumptions regarding support equipment in the support state and fault tree models were reviewed. Analyses were then completed, as necessary, to verify the assumptions. All support state and fault tree assumptions were confirmed by this analysis.

Phase II of the study involved a review of all Plant Response Trees (PRTs) for a determination of the components (including instrumentation) important in achieving 'successful' endstates. The limiting conditions, with respect to the PRTs, were then identified for each piece of equipment and a survivability evaluation was completed. The results of the phase II investigation show that all components that are modeled in the PRTs would be available for the appropriate accident sequences.

Phase III of the study will consider the equipment identified for accident management purposes. This will include the equipment needed for post-24 hour accident management to maintain the plant in a safe, stable state (i.e., a SAM endstate); the equipment needed for containment accident management following a core damage event; and any other equipment which is identified for the overall CECo accident management program. The Phase III effort is beyond the scope of the IPE and will be included in the implementation portion of the CECo accident management program for Quad Cities Station.

1.4.7 Source Term Analysis

Any sequence of events that causes core damage may result in a release of radioactivity to the environment in excess of design-basis limits. Such radioactivity releases are possible whether or not the containment building remains intact, because no structure is perfectly leak-tight. The amount of radioactivity that may be released from the containment building if core damage occurs is sequence-dependent and strongly influenced by the size and complexity of the flow paths out of the building. The amounts of radioactivity released from containment, reported as various isotopes, constitute the so-called source term for an accident sequence.

The purpose of a source term analysis is to quantitatively estimate the masses of the various fission products that are released from the containment structure for the PRT end-states (i.e., accident sequence types) that result in core damage. Performing actual source term calculations for each sequence would be impracticable, however, given the large number of sequences defined by the PRTs. Thus, the scope of the source term analysis was limited to a consideration of the 100 highest-frequency sequences that result in core damage.

Each of the top 100 core damage sequences was traced through the PRTs to determine the disposition (i.e., success, failure, not asked) of every node in a sequence's complete PRT path. These nodal dispositions were then translated into a list of systems and equipment that would operate during each sequence. In this way, similarity of event progression was verified for each sequence within a PDS. The system and equipment operability lists for the PDSs were also compared. This comparison identified a number of PDSs with very similar combinations of functional failures, accident progressions, and fission product release paths after core damage. These similarities allowed a further reduction in the number of deterministic fission product release calculations needed to characterize the source term behavior of the top 100 sequences.

A total of fourteen PDSs were thus identified for source term analysis from the twenty-four PDSs containing the top 100 sequences. Detailed fission product release calculations then were performed using the specific sequence progression of the highest frequency sequence in each of these fourteen PDSs. The range of containment performance obtained from these detailed analyses includes:

- One sequence in which the combination of core spray injection into the failed RPV and torus cooling prevent torus venting and containment structural failure;
- One sequence in which the combination of core spray and CRD injection into the failed RPV and torus venting prevent containment structural failure;
- One sequence in which the combination of drywell spray and CRD injection into the failed RPV in combination with torus cooling and torus venting prevent containment structural failure;
- Two sequences in which CRD injection into the failed RPV and torus venting prevent containment structural failure;
- Three sequences in which torus venting cannot prevent drywell structural failure between 25 and 48 hours after the sequence start;
- Two sequences in which the torus is the probabilistically-determined failure site and torus failure occurs after 24 hours;
- Two ATWS sequences in which drywell and torus structural failures occur within the first hour of the sequences;
- One SBO sequence in which drywell structural failure is predicted to occur between 12 and 24 hours after the sequence start; and
- One SBO sequence in which drywell structural failure is predicted to occur within the first 12 hours of the sequence.

1.5 IPE Results

This section provides a discussion and explanation of the Quad Cities IPE accident sequence results. First, traditional results based on the mission time of 24 hours are reported: overall core damage frequency, with a subsequent breakdown of core damage frequency by initiating event, plant damage state, containment status, equipment/operator failures, and sequences. Then, an innovative aspect of the Quad Cities IPE/AM program is reported: the frequency of occurrence of so-called Accident Management (AM) sequence endstates.

In the Quad Cities IPE/AM Program, two types of AM sequences have been defined. "Success with Accident Management" or "SAM" endstates have been defined for sequences with no core damage within 24 hours, but requiring accident management actions after 24 hours to assure continued long-term core cooling. "Containment success with Accident Management" or "CAM" endstates have been defined for sequences with core damage and no containment failure within 24 hours, but requiring accident management actions after 24 hours to assure continued long-term containment integrity. The method used for classifying sequences in the Quad Cities IPE/AM project is shown in Table 1.5-1.

1.5.1 Summary of Results

The core damage frequency for Quad Cities Station is 2.17E-06/yr. Of this total, the frequency of core damage and containment success is 7.1E-07/yr. The remainder of the core damage frequency would result in fission product releases due to venting the containment or containment failure (or both), that exceed normal containment leakage.

The initiating event frequencies and the respective contributions to the total core damage frequency (CDF) are shown in Table 1.5.1-1. As can be seen from this table, about 98.3% of the CDF comes from the top seven initiating events. The top contributor to core damage frequency is Dual Unit Loss of Offsite Power at56.67% followed by Single Unit Loss of Offsite Power at 11.58%.

The core damage frequency by plant damage state, for those PDSs contributing more than 0.5%, is shown in Table 1.5.1-2; these PDSs account for more than 95% of the total CDF. As can be seen from this table, failures of the high-pressure makeup function tend to dominate the overall core damage frequency. In fact, nearly 87% of the CDF involves loss of the high-pressure makeup function. This is primarily attributed to the long-term consequences of station blackout and hardware failures of high pressure injection following a LOOP.

TABLE 1.5-1
SEQUENCE CLASSIFICATION METHOD

SEQUENCE CLASSIFIED AS:	SEQUENCE TIME		APPROXIMATE SOURCE TERM MAGNITUDE	
	0 to 24 HR	> 24 HR	AT 24 HR	AT 48 HR
SUCCESS	CD NO Vent NO CF N/A	CD NO Vent NO CF N/A	Noble Gas 0 Volatile 0	Noble Gas 0 Volatile 0
SUCCESS WITH ACCIDENT MANAGEMENT (SAM)	CD NO Vent NO CF N/A	CD YES Vent NO CF N/A	Noble Gas 0 Volatile 0	Noble Gas * Volatile *
CORE DAMAGE, CONTAINMENT VENTED AND INTACT	CD YES Vent YES CF NO	CD YES Vent YES CF NO	Noble Gas ~100% Volatile ~0.01%	Noble Gas ~100% Volatile ~0.1%
CORE DAMAGE, CONTAINMENT VENTED AND INTACT	CD YES Vent YES CF NO	CD YES Vent YES CF YES	Noble Gas ~25% Volatile ~0.01%	Noble Gas ~100% Volatile ~10%
CORE DAMAGE, CONTAINMENT NOT VENTED AND INTACT (POSSIBLE CAM)	CD YES Vent NO CF NO	CD YES Vent NO CF YES	Noble Gas ~1% Volatile ~0.01%	Noble Gas ~100% Volatile ~10%
CORE DAMAGE, CONTAINMENT VENTED AND FAILED	CD YES Vent YES CF YES	CD YES Vent YES CF YES	Noble Gas ~100% Volatile ~10%	Noble Gas ~100% Volatile ~10%
CORE DAMAGE, CONTAINMENT NOT VENTED AND FAILED	CD YES Vent NO CF YES	CD YES Vent NO CF YES	Noble Gas ~100% Volatile ~10%	Noble Gas ~100% Volatile ~10%

CD = Core Damage
Vent = Wetwell vent operated in accordance with the procedures within 24 hours
CF = Containment Failure
N/A = Not Applicable
* = Not Estimated
CAM = Containment success with Accident Management
Note: A sequence is designated as "core damage" if core damage is predicted to occur within 24 hours, in accordance with the traditional approach. Similarly, a sequence is designated as "containment intact" if containment failure is not predicted to occur within 24 hours, in accordance with the traditional approach.

TABLE 1.5.1-1
CORE DAMAGE FREQUENCY BY INITIATING EVENT

INITIATING EVENT	INITIATING EVENT FREQUENCY (/YR)	CORE DAMAGE FREQUENCY (/YR)	PERCENT CONTRIBUTION
Dual Unit LOSP[1]	1.61E-02	1.229E-06	56.67
Single Unit LOSP	3.20E-02	2.510E-07	11.58
General Transient	3.87	2.341E-07	10.8
ATWS[2]	1.16E-04	1.665E-07	7.68
Medium LOCA[3]	8.00E-04	1.118E-07	5.16
L1B1[4]	1.01E-03	7.442E-08	3.43
LOSW[5]	9.10E-03	6.486E-08	2.99
Large LOCA	3.00E-04	2.103E-08	0.97
LB13[6]	4.65E-03	5.190E-09	0.24
LOIA[7]	1.43E-02	4.029E-09	0.19
Small LOCA	3.00E-03	2.556E-09	0.12
IORV[8]	1.06E-01	1.234E-09	0.06
LB14[9]	5.88E-04	8.150E-10	0.04
LB18[10]	4.18E-04	6.786E-10	0.03
L182[11]	1.34E-03	4.085E-10	0.02
LB11[12]	2.75E-04	3.302E-10	0.02
LB12[13]	2.05E-04	3.066E-10	0.01
ISLOCA[14]	1.20E-07	2.631E-10	0.01
TOTAL		2.169E-06	100

1. LOSP = Loss of Offsite Power
2. ATWS = Anticipated Transient Without Scram
3. LOCA = Loss of Coolant Accident
4. L1B1 = Loss of 125VDC Bus 1B-1
5. LOSW = Loss of Service Water
6. LB13 = Loss of Bus 13
7. LOIA = Loss of Instrument Air
8. IORV = Inadvertent Open Relief Valve
9. LB14 = Loss of Bus 14
10. LB18 = Loss of Bus 18
11. L182 = Loss of MCC 18-2
12. LB11 = Loss of Bus 11
13. LB12 = Loss of Bus 12
14. ISLOCA = Interfacing System LOCA

TABLE 1.5.1-2
CORE DAMAGE FREQUENCY BY PLANT DAMAGE STATE

STATE	DESCRIPTION	FREQUENCY	PERCENT CONTRIBUTION
BLAY	Station Blackout with late core damage (6-24 hours), operator fails to recover offsite power and failure to supply high-pressure coolant makeup	5.52E-07	46.1%
MEFG	Medium LOCA with early core damage (0-2 hours) and failure to depressurize or maintain sufficient high-pressure coolant makeup	1.63E-07	13.6%
LLCO	Loss of Offsite Power (single or dual unit) with late core damage (6-24 hours) and torus cooling fails	1.24E-07	10.3%
TEEQ	ATWS with early core damage (0-2 hours) and failure to trip recirc pumps or failure to inject SLC	3.52E-08	2.9%
LEAB	Loss of Offsite Power (single or dual unit) and early core damage (0-2 hours) with failure to supply high-pressure coolant makeup	3.30E-08	2.8%
LLBO	Loss of Offsite Power (single or dual unit) and late core damage (6-24 hours) with loss of all low-pressure reactor coolant inventory makeup	3.28E-08	2.7%
LLAC	Loss of Offsite Power (single or dual unit) and late core damage (6-24 hours) with failure to supply high-pressure coolant makeup	2.84E-08	2.4%
LLCS	Loss of Offsite Power (single or dual unit) and late core damage (6-24 hours) with torus cooling failure	2.65E-08	2.2%
TEFE	Transient event with early core damage (0-2 hours) and failure to depressurize or maintain sufficient high-pressure coolant makeup	2.43E-08	2.0%
TIGE	ATWS event with core damage at 2-6 hours and loss of all low-pressure reactor coolant inventory makeup	2.05E-08	1.7%
BEAY	Station Blackout with early core damage (0-2 hours), operator fails to recover offsite power and failure to supply high-pressure coolant makeup	1.96E-08	1.6%
TEFB	Transient event with early core damage (0-2 hours) and failure to depressurize or maintain sufficient high-pressure coolant makeup	1.86E-08	1.6%
ALCE	Large LOCA with late core damage (6-24 hours) and torus cooling failure	1.48E-08	1.2%
LLAB	Loss of Offsite Power (single or dual unit) with late core damage (6-24 hours) with failure to supply high-pressure coolant makeup	1.27E-08	1.1%
TEER	ATWS with early core damage (0-2 hours) and failure to trip recirc pumps or failure to inject SLC	1.05E-08	0.9%
LLCT	Loss of Offsite Power (single or dual unit) with late core damage (6-24 hours) and torus cooling failure	7.90E-09	0.6%
AEGG	Large LOCA with early core damage (0-2 hours) and loss of all low-pressure reactor coolant inventory makeup	7.67E-09	0.6%

TABLE 1.5.1-2 (Continued)
CORE DAMAGE FREQUENCY BY PLANT DAMAGE STATE

STATE	DESCRIPTION	FREQUENCY	PERCENT CONTRIBUTION
SLFG	Small LOCA with late core damage (6-24 hours) and failure to maintain sufficient high-pressure coolant makeup	7.63E-09	0.6%
	TOTAL	1.14E-06	95.0% of total CDF

The plant damage states shown in Table 1.5.1-2 also provide an indication of the capability of the plant to contain radioactive fission products within the plant boundaries. Of the total CDF, 32.3% consists of sequences in which the containment remains intact structurally and is not vented. These "containment success" sequences are represented in Table 1.5.1-2 by plant damage states having "B", and "G" (except "GG") in the fourth position. The sequences in which the containment is intentionally vented at some time during the event but does not fail structurally are represented in this table by the plant damage states having "CE" in the third and fourth positions. These sequences constitute about 1% of the total CDF.

Containment structural failures include plant damage states in which the containment fails due to pressure loading in combination with high temperatures regardless of whether or not the containment had been previously vented. Plant damage states in Table 1.5.1-2 containing "O" in the fourth position or "FE" in the third and fourth positions represent the vented-and-failed sequences. These sequences contribute 6% to the total CDF. Plant damage states in Table 1.5.1-2 including "Y" in the fourth position represent sequences in which containment structure fails without venting; these sequences constitute 52% of the total CDF. Rapid (high pressure) containment failure sequences compose 7.5% of the CDF and are represented on this table by "Q" and "R" in the fourth position of their endstate designator.

Table 1.5.1-3 identifies the key contributors, both hardware failures and operator errors for each of the top 14 core damage sequences. These 14 sequences contribute approximately 82% to the CDF. (The top 100 sequences are described in detail in Section 4.5.3.)

The core damage frequency of 2.17E-06 per year for Quad Cities is dominated (39.77% of total CDF) by sequence #1, a dual-unit loss of offsite power and failure to restore AC power, which leads to SBO and late (6-24 hours) core damage . Dual unit loss of offsite power sequences contribute 56.67% of the CDF. Sequences initiated by a single unit loss of offsite power at 11.58% of the CDF, general transient sequences contribute 10.8%, anticipated transient without scram sequences contribute 7.68%, medium LOCA sequences contribute 5.16%, loss of 125VDC bus 1B-1 sequences contribute 3.43%, and loss of service water sequences contribute another 2.99%. These seven initiators contribute 98.3% of the CDF. The top 14 accident sequences are composed of these initiators. The top 14 sequences have individual contributions greater than 1% of the CDF; from sequences numbered #15 and higher the contributions are less than 1% and the distribution of sequence frequencies becomes nearly flat.

These results show a significant contribution to CDF from support systems, specifically DC and AC power. Also, there is a significant contribution to CDF from high pressure makeup failure. Failures to accomplish certain operator actions, such as failure to depressurize, failure to align the safe shutdown makeup pump to take suction from the CCST also contribute to the CDF.

TABLE 1.5.1-3
KEY CONTRIBUTORS TO DOMINANT ACCIDENT SEQUENCES

SEQUENCE	EVENT	NODE	DESCRIPTION OF KEY CONTRIBUTORS
1	DLOOP	DG1	DG1 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 14-1
		DG2	DG2 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 24-1
		DGB	DG1/2 fails to run for 6 hours due to common cause mechanisms after DG1 and DG2
		SBO?	Station blackout occurs in Units 1 and 2
		ROP2	Failure to recover offsite power within 6 hours to prevent core damage
2	DLOOP	DG1	DG1 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 14-1
		DG2	DG2 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 24-1
		DGB	DG1/2 fails to run for 6 hours due to common cause mechanisms after DG1 and DG2
		SBO?	Station blackout occurs in Units 1 and 2
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		ROP2	Failure to recover offsite power within 6 hours to prevent core damage
3	LOOP	1TB	By procedure, TBCCW not restarted after LOOP
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
4	MLOCA	HP1	HPCI turbine fails to run for 5 hours
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
5	GTR	OFW1	Operator fails to restart a feedwater pump
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP-3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
6	DLOOP	DG1	DG1 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 14-1
		DG2	DG2 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 24-1
		DGB	DG1/2 fails to run for 6 hours due to common cause mechanisms after DG1 and DG2
		SBO?	Station blackout occurs in Units 1 and 2
		HP1	HPCI turbine fails to run for 5 hours
		ROP2	Failure to recover offsite power to prevent core damage
7	DLOOP	1TB	By procedure, TBCCW not restarted after DLOOP
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
8	GTR	PCSA	Power Conversion System unavailable
		OFW1	Operator fails to restart a feedwater pump
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours

TABLE 1.5.1-3 (Continued)
KEY CONTRIBUTORS TO DOMINANT ACCIDENT SEQUENCES

SEQUENCE	EVENT	NODE	DESCRIPTION OF KEY CONTRIBUTORS
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
9	LOSW	SW	Service water system fails due to initiating event
		1IA	Instrument Air fails due to initiating event
		PCSA	Power Conversion System unavailable
		FW	Feedwater system fails due to initiating event
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		CRD	Control rod drive system injection fails
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
10	ATWS	RCFM	Fraction of RPS failures that are mechanical
		OIADS	Operator fails to inhibit Automatic Depressurization System
11	L1B1	1M1	125VDC main bus 1A fails for 24 hours due to initiating event
		1R1	125VDC reserve bus 1B-1 fails for 24 hours
		FW	Feedwater fails due to loss of control power
		HP1	HPCI fails due to loss of control power
		LPA	RHR A fails due to loss of control power
		LPB	RHR B fails due to loss of control power
		RCIC	RCIC fails due to loss of control power
		SSMP1	SSMP fails due to loss of control power
		ADS	ADS fails due to loss of control power
		CS	Core Spray fails due to loss of control power
12	GTR	FW	Feedwater system fails with all supports available
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
13	L1B1	1M1	125VDC main bus 1A fails for 24 hours due to initiating event
		1R1	125VDC reserve bus fails for 24 hours
		PCSA	Power Conversion System unavailable
		FW	Feedwater fails due to loss of control power
		HP1	HPCI fails due to loss of control power
		LPA	RHR A fails due to loss of control power
		LPB	RHR B fails due to loss of control power
		RCIC	RCIC fails due to loss of control power
		SSMP1	SSMP fails due to loss of control power
		ADS	ADS fails due to loss of control power
		CS	Core Spray fails due to loss of control power
14	ATWS	MC	Main Condenser unavailable (given FW success) after ATWS
		RCFM	RPS mechanical failure
		OIADS	Operator fails to inhibit Automatic Depressurization System

Most of the CDF occurs early (equal to or less than 2 hours). Containment rapid high-pressure failure contribution is small (18 of the top 100 sequences, all composed of ATWS sequences). Containment failure subsequent to venting occurs in 17 of the top 100 sequences. Containment failure without venting occurs in 13 of the top sequences and venting without containment failure occurs in 6 of the top 100 sequences. In 43 of the top 100 sequences, the containment is intact without venting.

1.5.2 AM Endstates

SAM Endstates - The SAM endstates occur with a cumulative frequency of 1.82E-06 per year. The 15 accident sequences with highest frequency of a SAM endstate (all those contributing more than 0.5% each) represent 60% of the total SAM frequency; these sequences are presented in Table 1.5.2-1.

Based on a review of the dominant sequences with a SAM endstate, a set of possible accident management strategies to bring the plant to a long term safe, stable state can be developed. Since a relatively long time is available for accident management activities for the SAM endstates, the possible accident management activities include both repair of unavailable equipment and the implementation of alternate methods of achieving a safe, stable state. The information developed from the review of the SAM sequences represents input to the Accident Management Program for Quad Cities Station, for sequences which otherwise would progress to core damage at a time beyond 24 hours after the initiating event.

CAM Endstates - The sequences having endstate designators with "B", "G" (except "GG"), or "H" as the fourth character are all sequences in which the containment remains intact throughout the initial 24 hours of the event. CAM sequences would be that subset of these sequences in which the containment conditions were trending toward failure.

Within the top 100 core damage sequences, there are 46 sequences having "B," "H," or "G" (except "GG") as their fourth character. These sequence types were investigated to determine whether the containment was pressurizing, heating up, or whether conditions were stable. Of the 46 containment success sequences, none have been found to trend towards failure in the second 24 hour period. No CAM sequences, therefore, were identified in the Quad Cities IPE analysis.

TABLE 1.5.2-1
KEY CONTRIBUTORS TO "SAM" ACCIDENT SEQUENCES

Number (1)	Frequency (2)	Percent (3)	Damage State	Event (4)	Value (5)	Description (6)
2	2.92E-07	7.31	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
4	1.48E-07	3.71	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
6	1.17E-07	2.92	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
8	9.99E-08	2.5	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
12	5.87E-08	1.47	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
14	5.74E-08	1.44	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				1IA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
17	5.06E-08	1.27	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
18	4.92E-08	1.23	SAM	LB13	4.65E-03	LOSS OF BUS 13 IE
				13	1.00E+00	EVENT FAILS
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				1IA	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
19	4.34E-08	1.09	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				14	1.40E-03	LOSS OF BUS 14 (345KV UNAVAIL)
				1IA	1.00E+00	EVENT FAILS

TABLE 1.5.2-1 (Continued)
KEY CONTRIBUTORS TO "SAM" ACCIDENT SEQUENCES

Number (1)	Frequency (2)	Percent (3)	Damage State	Event (4)	Value (5)	Description (6)
				LPA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
20	4.28E-08	1.07	SAM	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
22	3.09E-08	0.77	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
24	2.80E-08	0.7	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
26	2.56E-08	0.64	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
28	2.38E-08	0.6	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
29	2.14E-08	0.53	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				14	1.40E-03	LOSS OF BUS 14 (345KV UNAVAIL)
				LPA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS

Notes:

1. "Number" refers to accident sequence ranking in the top 100 sequences (if the SAM damage states were combined with core damage states in a new top 100 damage sequence listing).
2. "Frequency" is the frequency per year that this sequence is expected to occur.
3. "Percent" is the percent of total core damage represented by this single sequence.
4. "Event" is the list of PRT and support system event tree top events which have failed in this sequence.
5. "Value" is frequency (for initiators) or probability (for failures) associated with each event.
6. "Description" defines the "Event" label.

1.6 IPE Evaluations

1.6.1 IPE Insight Development

In the broadest sense, insights are those observations regarding the station configuration or practices which may affect the risk profile of the plant. Insights can suggest changes to enhance the capability of the plant and the plant operators to respond to an initiating event to either prevent core damage or to mitigate the consequences of core damage. Insights can also include those "good features" which have been identified during the IPE process. The IPE insights described in this section address the capability of the existing plant (July 1991) to respond to an initiating event. IPE insights are distinguished from Accident Management insights (see Section 1.7) which deal with enhancements to the capability of the plant emergency response organization to respond to a core damage accident situation, given that it has occurred.

In order to focus the IPE analysts on the identification of IPE insights, it was necessary to develop structured guidance. The development of the guidance began with the definition of the aspects of the plant which can impact the severe accident risk profile including: plant design features, testing and maintenance activities, the emergency procedures, training, and plant status information. These broad features were then correlated to the IPE work products to define the types of IPE insights which could be obtained from each task of the IPE analyses. This detailed correlation of possible plant features versus IPE work products was used to define a set of questions for each IPE task which would focus and stimulate the IPE analysts to identify insights as the tasks were being performed. Therefore, at each step of the risk assessment, analysts were systematically required to answer questions to stimulate the identification of insights. In addition to changes to Quad Cities Station to improve the accident risk profile, the IPE insights also include good features of Quad Cities Station which contribute to its present risk profile.

The IPE insights identified in the current study are, in many cases, significantly different from those identified in previous PRA studies. The primary difference is in completeness of the search for insights and the comprehensive coverage of all of the aspects of the IPE. The Quad Cities IPE insight development methodology prescribes the identification of insights by each analyst as the work is ongoing, instead of the process employed in previous PRA studies, which was backward looking from the IPE results.

Another aspect of the IPE insights identified during this study is the overall approach of using best estimate analyses for the accident progression and mapping the plant procedures to the accident progression to determine those operator actions which can impact the accident progression. This process has resulted in a comprehensive review of the plant procedures for their impact on the progression of accidents, including core damage accidents. As a result of this review, insights were developed relating to enhancements to the procedures, primarily to improve clarity and the likelihood that appropriate operator actions will be taken in response to plant parameters.

IPE Insight Evaluation

For each of the 81 IPE insights developed during the IPE, a process of evaluation was followed. The first step of the process was a distillation of the insights by a "Tiger Team", composed of individuals from CECo and the IPE Partnership. The first step of the distillation consisted of verifying the technical accuracy of each of the insights. The Tiger Team then grouped all of the insights related to the same subject together for further evaluation. Groupings were performed for the following subject areas:

- LOOP/SBO
- Drywell Flooding
- ISLOCA
- NRC Strategies
- Containment Performance

Some of the insights appeared in multiple subject areas (e.g., plant procedural changes and hardware modifications to address the plant response to a LOOP or SBO condition may also be one of the NRC Strategies.)

At that point the insights within a group were evaluated for their effect on the risk profile of the plant. The insights with the greatest impact on the risk profile were identified. Of particular interest are those insights which provide a major benefit to risk reduction and can be implemented with minor impact to plant hardware or procedures.

A further grouping of insights was performed to facilitate the disposition of plant enhancements by CECo management. This grouping consisted of the following types of enhancements: Plant Specific Procedure Enhancements (51%), Hardware Enhancements (26%), Training (6%), Information (15%), and Test & Maintenance (2%). The generic procedure enhancements that were identified during the Dresden IPE analysis are also applicable to the Quad Cities Station and have been forwarded to the BWR Owners Group (BWROG) and to the Quad Cities Station for their consideration.

All of the insight evaluation information was then given to the Senior Edison Management Review Team (SEMRT) for final evaluation and disposition. As part of their evaluation process, the SEMRT utilized the NUMARC Closure Guidelines (NUMARC 91-04).

1.6.2 Evaluation Against NUMARC Severe Accident Issue Closure Guidelines

The results of the Quad Cities Modified IPE have been evaluated against the NUMARC Severe Accident Closure Guidelines. The guidelines were used to assess the possible enhancements developed via insights related to severe accidents.

The first step in using the Severe Accident Closure Guidelines was to group the core damage sequences; the groupings used were those of Table B-1 of that document.

The grouping was carried out for all core damage sequences down to the quantification frequency cutoff of 1E-12 for a given sequence. The following groups contain some contribution to the total core damage frequency:

IA Accident sequences involving loss of coolant inventory makeup in which the reactor pressure remains high.

IB Accident sequences involving a loss of all AC power and loss of coolant inventory makeup (i.e., station blackout).

ID Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced.

II Accident sequences involving loss of containment heat removal leading to containment failure and subsequent loss of coolant inventory makeup.

IIIB Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized and inadequate coolant inventory makeup is available.

IIIC Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is at low pressure and inadequate coolant inventory makeup is available.

IV Accident sequences involving an ATWS leading to containment failure due to high pressure and subsequent loss of inventory makeup.

V Unisolated LOCA outside containment leading to loss of effective coolant inventory makeup.

The sequence numbers of the top 100 sequences included in each group are listed in Table 1.6.2-1 with the resulting mean group core damage frequency and percent contribution to the total core damage frequency. The group core damage frequency and contribution is based upon all sequences.

The core damage frequency and percent contribution to the total core damage frequency for each group were then evaluated against Tables 1 and 2 of the Severe Accident Closure Guidelines. Table 2 was used for the containment bypass sequences (group V only), and Table 1 was used for all other groups. With the exception of the Class 1B sequences, the comparison shows all the accident sequence groups fall below the Severe Accident Closure Guidelines frequency cutoffs.

The Class 1B sequences (Station Blackout) exceed the percentage cutoff value specified on Table 1 and just exceed the lowest frequency limit requiring action. Currently, additional diesel generators are being added at the station; one new generator is being added at each unit. These generators will significantly reduce the likelihood of station blackout and will be included in an update to the Quad Cities PRA model.

TABLE 1.6.2-1
NUMARC SEVERE ACCIDENT CLOSURE GUIDELINES
SEQUENCE GROUPING INFORMATION

Sequence Group	Sequence Numbers[1]	Total Group Core Damage Frequency	Percent Contribution to Total CDF
IA	3, 5, 7, 8, 9, 11, 12, 13, 16, 18, 19, 20, 21, 22, 24, 26, 27, 32, 33, 35, 37, 39, 40, 42, 44, 46, 51, 54, 55, 56, 62, 65, 67, 69, 70, 71, 73, 77, 78, 79, 82, 83, 85, 86, 87, 88, 94, 96	7.0E-07	32%
IB	1, 2, 6, 47, 58, 75, 76, 100	1.1E-06	51%
ID	34, 41, 64, 68, 90, 91, 95, 97, 98, 99	3.4E-08	2%
II	38, 49, 52, 74, 81, 92	2.4E-08	1%
IIIB	4, 60, 61, 93	1.1E-07	5%
IIIC	29, 31, 45, 66, 72, 89	2.3E-08	1%
IV	10, 14, 15, 17, 23, 25, 28, 30, 36, 43, 48, 50, 53, 57, 59, 63, 80, 84	1.6E-07	8%
V	(None in the top 100 sequences)	2.5E-10	<<1%

[1] Refers to the sequence position in the ranking of core damage sequences in descending magnitude of core damage frequency for the top 100 sequences only.

1.7 Accident Management

Commonwealth Edison has integrated the definition of an Accident Management (AM) Program with the performance of the IPE. The CECo AM elements are similar to those proposed by the NRC. The five elements of the CECo AM program are:

- Organization and Decision Making,
- Accident Management Guidance (Strategies),
- Calculational Tools,
- Training, and
- Plant Status Information.

Differences from the NRC approach include the expansion of the plant instrumentation area to include vital plant information needs for AM, the expansion of AM guidance to include the interface with the site emergency plan, and the consideration of predictive and decision-making tools within the calculational tool element, as necessary to meet BWR Owners Group AM Guideline implementation requirements.

CECo believes that the management of severe accidents with potential or actual core damage, where the situation is beyond the realm of the EOPs, should be the responsibility of the emergency response organization, outside the Control Room. The CECo AM program is being developed with this philosophy. The EPG action placement criteria being developed by the BWR Owners Group to optimize the location of the EPG action (Control Room vs. TSC) will be utilized in this determination.

The methodology used by CECo is a forward-looking process incorporated in each phase of the IPE work. The CECo approach encompasses the key aspects of the EPRI and NRC methodologies and employs a simultaneous "top-down" and "bottom-up" method. The top-down evaluation has logically defined the elements of an intuitive AM program framework, as described above, and identified where the various aspects of the IPE effort could support enhancement of these elements. The bottom-up approach examined the technical analysis at each of the major steps of the IPE for observations that could fall into one or more of the five AM framework elements.

The search for AM insights covers all aspects of the IPE analysis, not just the dominant accident sequences. Potential and possibly subtle strategies and insights are best identified and documented while related information is actively under evaluation by the IPE analysts. Improved understanding of the plant capability to respond to accidents and the operator response to accident symptoms is one of the most important benefits to be obtained from the Quad Cities IPE, and the decision to develop and evaluate AM insights at the onset of the IPE for Quad Cities Station has maximized this benefit.

A detailed matrix of the above AM program elements and IPE work products was used to define a set of questions for each IPE task which would focus and stimulate the IPE analysts to identify applicable AM insights as the IPE tasks were being performed. As was the case for IPE insights for plant enhancements to prevent core damage, each of the individual AM insights was evaluated by a "Tiger Team," composed of individuals from CECo and the IPE Partnership. The individual insights identified by the bottom-up approach were evaluated on their technical merit. Insights were combined, where appropriate, and a qualitative assessment was then performed.

For Quad Cities Station, 56 individual AM insights were identified by the IPE analysts and evaluated by the Quad Cities Tiger Team. The distribution of insights over the elements of the AM framework was concentrated in accident management strategies and information, as expected:

Organization	2%
AM Strategies	50%
AM Tools	18%
AM Training	2%
AM Information	28%

AM insights have also identified the benefit of providing alternate means of achieving containment sprays to control fission product release fractions, especially under station blackout conditions.

In addition, AM insights have been formulated based upon a series of experiments performed as part of the CECo IPE/AM program. These experiments verified that submerging the bottom portion of the reactor vessel can prevent vessel failure after relocation of the damaged core to the lower head, assuming that the RPV support skirt can be modified to allow the egress of steam. Severe accidents experiments were also conducted to investigate the efficacy of decay heat removal by a water spray on the outside surface of the lower plenum of the RPV. The use of a spray would require a modification but would avoid the need to modify the support skirt. The timely cooling of the RPV wall could be established immediately upon spray initiation. The ex-vessel spray system would be provided with an external source of water located outside the containment. In addition to providing the source of cooling water for ex-vessel spray and decay heat removal, such a flow path would provide a means of establishing a flooded containment condition that did not require injection through the RPV as is the case for the existing plant configuration. The successful maintenance of the reactor pressure vessel integrity during a severe accident could preclude the occurrence of several ex-vessel severe accident phenomena such as direct containment heating, ex-vessel steam explosions, molten core-concrete interactions, containment overpressurization due to debris cooling in the containment, and the ex-vessel release of fission products.

CECo has evaluated the benefits of implementing the above described external vessel cooling strategies utilizing the CECo Societal Risk Analysis Model. The conclusion reached is that the level of avoided risk does not justify implementation of such modifications at this time. As a result, CECo is not actively pursuing implementation of external vessel cooling for either pedestal flooding or external vessel spray strategies.

1.8 Conclusions

The Quad Cities IPE/AM project is believed to be one of the most comprehensive PRAs ever undertaken. It has provided a new level of understanding of the plant and its behavior under a variety of potential accidents.

The realistic modeling employed in the Quad Cities IPE shows that the Quad Cities Station design is very good. The Quad Cities emergency procedures are effective in responding to severe accidents, and they contribute to Quad Cities Station's low core damage frequency. Quad Cities Station was found to have no serious weaknesses or vulnerabilities.

The core damage frequency was determined to be 2.17E-06 per year. Quad Cities Station is somewhat sensitive to one particular initiating event, Loss of Offsite Power in Both Units. Of the total core damage frequency, over 91% is spread over five initiating events; the Loss of Offsite Power in Both Units contributes 57% toward this total. The next four types of events are Single Unit Loss of Offsite Power (12%), General Transient (11%), ATWS (8%) and Medium LOCA (5%).

The frequency of the most likely sequence, a Loss of Offsite Power in Both Units with subsequent failure of all onsite AC power (station blackout) in both units, is 8.62E-07 per year; this constitutes about 40% to the total core damage frequency. The next most likely sequence is also a DLOOP sequence. This sequence has failure of high pressure injection and contributes about 7% to the core damage frequency (1.6E-07 per year). The third most likely sequence, a single unit LOOP with failure of high pressure injection and failure of operator action to depressurize and to allow low pressure injection, contributes about 7% to overall CDF. Sequence 4 is a Medium LOCA with failure of HPCI and failure of operator action to depressurize and allow low pressure injection. This sequence contributes about 5% to CDF. Sequence 5 is a general transient with failure of high pressure injection and operator action to depressurize and contributes about 4% to CDF. The next two most likely sequence are DLOOP sequences that each contribute about 3% to CDF.

A review of the results indicates the most important hardware contributors toward total core damage frequency are failures of the diesel generators. The quantitative importance of emergency AC power sources is influenced significantly by the dependency of the plant on electrically-driven systems for long-term decay heat removal. The most significant operator-related contributions result from the failure to depressurize the reactor vessel when required and failure to initiate the SSMP. Of the three cases modeled for operator action to initiate the SSMP (i.e., initiation with suction from the CCST, initiation with suction from the fire system, and initiation with suction from the CCST and an ECCS signal present), the most important is initiation from the CCST with an ECCS signal present. Based on the low overall core damage frequency, these actions do not represent a plant vulnerability. Nonetheless, IPE and AM insights have been identified to reduce the impact of these failures and to provide accident management guidance to emergency response organizations to ensure these important actions are achieved.

A review of the modified IPE results against NUMARC Severe Accident Issue Closure Guidelines (NUMARC 91-04, January, 1992), reveals that with the exception of the Class IB sequences, all the accident sequence groups fall below the Severe Accident Closure Guidelines frequency cutoffs.

The Class IB sequences (Station Blackout) exceed the percentage cutoff value specified by the guidelines and just exceed the lowest frequency limit requiring action. Recently, additional diesel generators have been added at the station; one new diesel generator has been added at each unit. These diesel generators will significantly reduce the likelihood of station blackout and will be included in an update to the Quad Cities PRA model.

The use of realistic analyses, in conjunction with modeling the emergency procedures has shown that some accident sequences do not achieve core damage until well after 24 hours. Consistent with traditional PRA philosophy, these have not been classified as core damage sequences in the IPE results. The SAM sequences have a predicted frequency of occurrence of 1.82E-06 per year. The SAM accident sequences identified by the Quad Cities Modified IPE are primarily loss of offsite power sequences (in one or both units) in which decay heat removal ultimately fails. In these sequences actions are required after the initial 24 hours to ensure that long-term core cooling can be maintained.

Only one sequence in the top 100 sequences of the modified IPE resulted in a large early release frequency (LERF), as defined in the NEI-sponsored PSA Applications Guide (EPRI TR-105396). This sequence is an ATWS sequence with containment failure in the drywell. Sequences that result in LERF contribute less than 2% to overall CDF with a frequency of 3.74E-08. The frequency of uncontrolled release caused by high pressure and/or high temperature was calculated to be 1.29E-06 per year. This frequency consists mostly of ATWS and loss of offsite power events where containment venting is either unavailable or ineffective. Source terms in these sequences are much larger than those due to other types of accident sequences. Sequences in which the containment is vented during the event and, though vented, fails later due to high temperature contribute 8E-08 per year. Another group of sequences involve venting the containment with the containment remaining intact; these contribute 1.1E-07 per year. In yet other sequences, low pressure injection or drywell sprays are used in combination with suppression pool cooling to prevent containment failure and limit source terms to containment leakage. The interfacing systems LOCA sequence frequency of 2.6E-10 per year at Quad Cities makes ISLOCA a negligible contributor to source term and plant risk. For an inerted containment, the likelihood of plant operation with a failure to isolate is extremely remote.

The Quad Cities IPE demonstrated that MAAP is a very useful tool for plant analysis. A CECo-specific version of the MAAP code was found to be of value for system success criteria and for event timing, as well as for calculation of radioactive releases.

The Commonwealth Edison engineering staff has been intimately involved in the IPE process and has acted as both originator of IPE analyses and reviewer of all IPE analyses. In addition, the ComEd PRA staff performed all the analyses required for the Modified IPE. As a result of the Integrated IPE/AM Program, the CECo PRA staff has developed a unique understanding of the behavior of the plant under accident conditions and of the total plant capabilities to respond to accidents.

The principal purpose of the Quad Cities IPE was to develop an understanding of the response of the plant to severe accidents. It accomplished this purpose. A second purpose of the Quad Cities IPE was to serve as the basis for an Accident Management program. The insights developed during performance of the Quad Cities IPE will be reviewed and dispositioned during the Quad Cities-specific implementation of the BWR Owners Group generic SAM guidance The final results of the study support the idea that the best improvement for plant safety is a good Accident Management program.

Although accident management considerations were not required at this time, an extensive accident management evaluation program was conducted to fully integrate the IPE/AM effort.

Finally, the results of the IPE have been carefully reviewed to identify areas where plant improvements could be effectively made with emphasis on core damage prevention.

2.3 Information Assembly

A tremendous amount of information was needed to perform the detailed Quad Cities IPE/AM study. The project team reviewed and assembled information from plant specific sources, relevant plant studies, and generic sources. Plant walkdowns were an important part of the data collection effort. Information was assembled to familiarize the analysts with the plant, determine and quantify the important initiating events, determine the component and system failure rates, perform various supporting analyses (e.g., common cause failure), conduct the evaluation of internally initiated flooding events, and develop plant layout insights. Table 2-1 provides a list of the important information sources reviewed. Complete lists of individual references are documented in the project notebooks.

The Quad Cities team for the original IPE/AM study used only the latest revision of drawings, design documents such as the FSAR, and plant procedures that were available as of July 1991. Differences between these documents and systems and layout were noted and resolved or included in the models as appropriate, if found. Thus, the PRA models for the original IPE reflect the Quad Cities as-built condition as it existed in July 1991[1].

The Modified IPE addresses several issues raised in the 1994 RAI by incorporating information from additional walkdowns and updated data. Walkdowns were performed in response to an issue raised in the 1994 RAI concerning electrical switchgear threatened by pipe spray. These walkdowns resulted in water spray failure rates being incorporated into initiating event frequencies for special initiators for one 480 VAC MCC and four 4 kV buses. These initiators were added to the Modified IPE. The 1994 RAI also raised an issue concerning the failure to open at demand probability for the Electromatic Relief Valves (ERV). Specifically, the probability estimated during the original IPE was two orders of magnitude higher than the generic (NUREG/CR 4450) value actually used in the original IPE. The generic ERV fail to open probability of 3.0E-04 was replaced with the plant specific failure probability of 3.57E-02 for the Modified IPE. This plant specific value is derived from data from the time period of the original IPE (January 1985 through December 1991).

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[1] One modification to the plant, imminent in July 1991, was included in the "baseline" Quad Cities model: the installation of the hardened containment vent.

Instrument failure rates were reassessed based on a plant review of calibration events. This review of Quad Cities Licensee Event Reports (LERs), Deviation Reports (DVRs), and Problem Investigation Reports (PIRs) was performed using a computer database that included the approximate period of 1991-1995. The review verified that the use of generic failure rates, as were used in the original IPE, was conservative. However, the mission times used to calculate failure probabilities was found to be less conservative. The Modified IPE incorporated revised instrument failure probabilities based on generic failure rates multiplied by increased instrument mission times thus resulting in higher probabilities than used in the original IPE.

In a February 28, 1996 request, the NRC formally asked Quad Cities to estimate the impact on CDF of recent (1993 through 1995) unavailability and reliability data for key systems. This information was provided in a letter from the station to the NRR and is incorporated in the Modified IPE. Unavailability and reliability data was updated for the HPCI, RCIC, Safe Shutdown Makeup Pump (SSMP), RHR, and Emergency Diesel Generators (EDGs).

Additionally for the Modified IPE, selected human reliability analyses were revised. For the revised human reliability analyses, plant procedures that were available as of December, 1995 were used. Human reliability probabilities were recalculated using a newer technique, the EPRI CBDTM (Cause Based Decision Tree Methodology), utilizing nominal (mean) values from various EPRI decision trees and NUREG tables.

Detailed system notebooks were developed for 21 major systems and miscellaneous systems that were expected to have an influence on the IPE/AM results. In addition, notebooks were developed for the major elements of the IPE/AM analyses (e.g., initiating events, internal flooding, etc.). Again, the plant information sources identified in Table 2-1 were used to develop system descriptions and models. Both plant specific and generic sources were used to define component availabilities, initiating events and initiating event frequency, important accident sequences, potentially important modeling features, common cause failure rates, and human reliability data. Subsequent sections of this report provide more detailed discussions of the specific use of the information collected.

While no other Quad Cities specific PRA studies were directly consulted for use in the IPE/AM, information from plants similar to Quad Cities has been collected and incorporated where appropriately justified. NSAC 151, a report on other plant PRA's, was also reviewed for applicability.

Plant walkdowns were conducted by members of the IPE/AM team who were responsible for the evaluation of specific plant systems, or areas of special interest, e.g., internal

The plant systems and operator actions were modeled in detail including explicit modeling of all key components. A CECo-specific version of the MAAP code was used to develop realistic success criteria as well as the plant response to each important accident sequence. Accident Management assessment methodology was integrated into the steps of the PRA evaluation to develop detailed insights concerning system faults and potential recovery actions. The key tasks are discussed below.

Plant Familiarization

Plant familiarization was accomplished through a combination of a comprehensive document review and plant walkdowns. The document review included: the plant FSAR, design drawings, design descriptions, training materials, normal and emergency operating procedures, technical specifications, test procedures, location and layout drawings. Walkdowns were used to develop an appreciation for the potential environmental impact on equipment that is difficult to discern from drawings. In addition, design descriptions and drawings were checked for accuracy and completeness during the walkdown process.

Plant Information and Data Analysis

Originally, plant specific information was collected from a variety of logs, reports, and operator interviews for the period from January 1, 1985 through December 31, 1991 to examine plant specific component failure, testing, and maintenance data as well as initiating events that have led to reactor trips. As a result of a February 28, 1996 NRC request, recent (1993 through 1995) unavailability and reliability data for HPCI, RCIC, SSMP and the EDGs were collected. This recent data included in the model. In a few instances, generic data from IEEE-500, NUREG/CR-2815 Revision 1, or other sources, were used to supplement plant specific information when sufficient plant data was not available. For common cause failure data, the Multiple Greek Letter (MGL) method was used to generate failure probabilities.

Accident Initiators

The selection of accident initiating events for Quad Cities was made from the collection and analysis of plant trip data. Additionally, the plant specific data evaluation was supplemented with industry data from NUREG/CR-3862 and from WASH-1400.

Quad Cities trip data was collected from scram reports, deviation reports, LERs and plant operating logs to identify actual trip events, power level at which the trip occurred, the failure which caused the trip, and the safety equipment that operated in response to the event.

The Quad Cities accident initiating events include: large LOCA, medium LOCA, small LOCA, interfacing systems LOCA (ISLOCA), loss of offsite power, transients and special (support system based) initiators. Most transient initiators were evaluated together since they all leave the plant in a state where the reactor protection system (RPS) is challenged and demand is made of safety systems to provide the reactor with a reliable source of cooling and makeup injection. These transients include such events as reactor trips, turbine trips, loss of main feedwater, etc. Special initiators include loss of 4KV AC Buses 11, 12, 13, 14, 21, 22, 23 and 24, loss of 125 VDC Reserve Bus 1B-1, loss of Instrument Air, loss of Service Water, loss of outboard MSIV room cooling due to loss of 480V Buses 18 and 28, and loss of 480V MCC 18-2 and 28-2.

Several methods were employed to determine initiating event frequencies for the relevant initiators. For those events having sufficient plant specific data, each event was categorized as identified above and the frequency determined by the number of events in the category. For events where there were insufficient plant specific data, such as LOCAs, the initiating event frequency was developed from generic data or similar plants. In the cases of small, medium, and large break LOCAs, the initiating event frequencies were taken from WASH-1400. Loss of offsite power was determined from a detailed study of the CECo grid reliability and plant experience.

Internal flooding was treated as a special initiator. A separate analysis was performed to determine if areas in Quad Cities are susceptible to flooding, and if there is sensitive equipment in those areas that could cause a plant shutdown or result in one or more failed safety systems. If such areas and equipment were identified, the contribution to core melt was evaluated from flooding those areas. The event trees from the other internal event initiators were used to quantify the contribution of flooding to core melt frequency.

Intersystem Dependence

A detailed analysis of dependence of the various frontline safety systems and supporting systems was conducted for each of the identified initiators, and for the interactions between the two Quad Cities units. Dependency matrices were developed for the dependence of a) frontline and support systems upon the initiating events, b) frontline systems upon support systems, c) frontline systems upon other frontline systems, d) support systems upon other support systems, and e) systems in one unit upon the systems or initiators in the other unit.

Information obtained from the accident initiating event analysis, system description documents and other sources, and plant walkdowns was used to construct the dependency matrices.

Event Tree Modeling

Plant response event trees (PRTs) and support state event trees were used to develop a Quad Cities accident sequence model. A plant response tree was developed for each initiator. A support state model was developed for each major class of initiating events: Transients (including LOCAs and most Special Initiators), Loss of Offsite Power, and the loss of 125 VDC (reserve bus) Special Initiator.

Support systems were modeled in a separate support system event tree. The support system model was a dual unit model since the two Quad Cities units share important support systems. The frontline system trees were connected directly to the support system event trees by the computer code, QT, for the quantification process.

The plant response tree is a relatively unique and innovative modeling approach used by CECo for all IPE models. This tree combines considerations of core damage prevention with those of accident progression through accident mitigation or containment failure and fission product release. The trees were developed by evaluating the accident initiator to determine the relevant critical safety functions and identifying the combination of safety systems and proceduralized operator actions required to bring the plant to a safe stable condition. These were identified for each initiator and modeled together as events within the event tree structure.

The plant response trees also modeled consequential or active failures that could change plant response by transfers to other plant response trees. Thus the models appropriately accounted for events not normally associated with the response tree developed for the particular initiating event.

A CECo-specific version of the MAAP code was used to develop and validate the accident sequence assumptions. These assumptions include items like the success criteria for achieving inventory control with frontline systems and the time available to accomplish operator actions successfully. Both system success criteria and operator actions were modeled realistically so that the accident sequences represent best estimate plant response. The best estimate approach was considered to be essential for use in developing appropriate accident management strategies and useful IPE insights.

Systems Analyses

The Quad Cities systems were modeled with fault trees. For each system, the analysis included the development of detailed system notebooks describing the system, its operation, the effect of accident conditions (success criteria, initiator impact, etc.), its operating experience, the system models and assumptions, quantification, and analyst insights. The relationship between the two units was also carefully examined and discussed.

The development of the fault tree models was done from the top event down. Fault tree development was aided through the development of simplified process & instrumentation diagrams (P&IDs) and fault tree modules which simplified and standardized the fault tree layouts. The fault trees were quantified using the computer program, GRAFTER.

Accident Sequence Quantification

The plant response trees were quantified with the QT computer code to calculate the probability of the accident sequences. The plant response trees first required the quantification of each system node, taking into account the various dependencies that can affect each node. In addition to PRT nodes involving system reliability which are quantified through the use of fault trees, operator action nodes were originally quantified using the human reliability analysis model, THERP (Technique for Human Error Rate Prediction). For the modified IPE, significant operator actions were re-evaluated utilizing a newer alternate technique, the EPRI CBDTM (Cause Based Decision Tree Methodology) from EPRI TR-100259. Other events which do not fit either of the above categories (e.g., restoration of offsite power) also required quantification on the basis of generic data. Intersystem shared equipment dependencies were modeled in the trees by including conditional probabilities for the event.

TABLE 2-1
QUAD CITIES IPE/AM INFORMATION SOURCES

SOURCE	PURPOSE[1]
Plant Specific	
System Descriptions	SN, DM
Updated FSAR	SN, DM
Units 1&2 General Arrangement Plan	SN, IF
Fluid System Drawings	SN, FT
Piping & Instrumentation Drawings	SN, FT
Station Electrical Drawings	SN, FT, SS
Station Structural Drawings	SN, ES
Technical Specifications	SN, FT, Q
Abnormal Operating Procedures	SN, FT, PRT
Emergency Operating Procedures	PRT, SN
Periodic Test Procedures	FT, Q
Maintenance Procedures	FT, Q
Licensee Event Reports (LERs)	Q, IE
Deviation Reports (DVRs)	IE, Q
Plant Operating History	IE, Q
Maintenance Records	Q
Scram Reports	IE, Q
Inoperable Equipment Log	Q
Quad Cities Operating Procedures (QCOP)	SN, FT
IPE/AM System/Containment/Flooding Evaluation Walkdowns	SN, FT, DM, IF
Emergency Plan	ST
Plant Pump Head Curves for Key Pumps	ST, SC
HVAC Calculations	ES
DBA and LOCA Calculations	ST
Vendor Data/Specifications for Safety Grade Components	SN, SC
Fire Protection Studies	ID, PRT, IF
Tech Staff Surveillance Procedures	SN
Problem Investigation Reports (PIRs)	Q, IE

[1] Acronyms are defined at the end of this table.

The IPEP project team was also supported on an as needed basis by personnel from the various IPEP organizations with the requisite skills and experience such as HRA.

3.2 Project Review Process

The complete and accurate modeling of Quad Cities Station was accorded the highest priority in developing the Quad Cities IPE/AM program. Only with accurate representation of the plant and its response to the dominant accident sequences could plant design and accident management insights be meaningful. To insure that the models developed accurately represented the plant, detailed reviews were implemented. The contractor analysts were organized so that independent review of each system model was conducted by selected members of the contractor team before its submittal to CECo. The CECo IPE/AM organization, described in Section 3.1 above, performed detailed technical reviews of the submitted models. Models were assigned to individuals or several of the PRA staff according to the expertise required. The reviews examined the quantification of the models as well as their accuracy and completeness. Comments and guidance were provided in meetings with contractor analysts to insure that appropriate modeling changes were made. This process was repeated until the CECo PRA staff was satisfied with the quality of the models.

At important milestones in the study, products and results were reviewed with the SMST and the CECo program manager. These additional reviews were conducted to insure that the approach taken, the products developed, and the results obtained were reasonable and acceptable.

A final review of the IPE/AM study was conducted by CECo senior management. Decisions concerning IPE and/or AM recommendations were made as part of the CECo management review.

As described in CECo's Project Plan, and as submitted in the earlier CECo response to Generic Letter 88-20, no separate "independent review" of the base Quad Cities IPE was performed. It is CECo's view that the quality of the study is assured by the employment of knowledgeable, experienced analysts both at IPEP and at CECo; as well as the many levels of review within the CECo program. The changes made in the generation of the modified Quad Cities IPE received an independent, peer review.

In addition to reviews of the IPE/AM studies conducted for accuracy and completeness, reviews were organized to develop realistic insights for possible plant improvements or accident management strategies. Program personnel participated in review of the models, results, and products as well as relevant material from other programs to identify and suggest insights. A group of senior level engineers, designated the "Tiger Team" and familiar with all aspects of plant design, operations, licensing environment, and severe accident issues, met regularly to review and distill insights into meaningful accident management strategies and/or plant improvements. The insights were further reviewed by the contractor Senior Management Support Team for further refinement, and a final review was made by CECo senior management of the recommendations as well as all other insights not selected for further consideration. The "Tiger Team" is intended to review the insights from all future CECo IPE/AM studies. CECo participants on the Quad Cities team are Mr. Harding, Mr. Raney, Mr. Knoespel, Mr. Klopp, Mr. Christensen, and Mr. MacLennan. The mix of CECo personnel who are intimately familiar with the plant and senior engineers with broad outside experience proved especially valuable in discerning the value of suggested insights and formulating broad strategies.

4.0 ANALYSES

4.1 Accident Sequence Delineation

This section describes the three key elements in defining the accident sequences. These three elements are the initiating events, the support system modeling, and the plant response tree modeling.

4.1.1 Initiating Events

CECo has identified the initiating events relevant to the Quad Cities IPE. These initiating events include the following:

- LOCAs involving the loss of coolant from the primary system due to pipe breaks, safety/relief valve failures and interfacing system piping ruptures.
- Transient events including reactor trips, turbine trips, loss of main feedwater, and loss of the condenser.
- Special (support system based) initiators.
- Other initiators including loss of offsite power (LOSP), loss of all AC power, and anticipated transient without scram (ATWS).

Transient events were identified through BWR operating experience. The following steps were taken to create a database of transient initiating events and make them specifically applicable to Quad Cities:

- Data from NUREG/CR-3862 was used to supplement historical data for categories which had no occurrence from 1/1/85 through 12/31/91.
- The Quad Cities trip history was reviewed to identify events that have occurred at this plant. To assure that the data reflected current Quad Cities configuration and operating practices, the time frame investigated was 1/1/85 through 12/31/91.
- The Quad Cities design and abnormal operating procedures were reviewed to determine whether plant conditions were considered that may result in the addition or deletion of accident initiators.
- The results of plant systems analyses were utilized to identify potential initiating events.

The Quad Cities initiating event frequencies are listed in Table 4.1.1-1. A summary of the method to develop each initiating event frequency is provided in the following paragraphs.

Large, Medium and Small LOCA frequencies were taken from WASH-1400 for this analysis.

The interfacing system LOCA frequency analysis was determined by a Quad Cities specific calculation considering all likely flowpaths. The frequency of high-energy line breaks (steamline-feedline ruptures) was calculated based upon methodology originally presented in WASH-1400 and further developed in the "Boiling Water Reactor Individual Plant Evaluation Methodology" by Delian Corporation.

The anticipated transient frequency is the sum of anticipated transient frequencies for Quad Cities. The NUREG/CR-3862 ("Development of Transient Initiating Events Frequencies for Use in Probabilistic Risk Assessments," May 1985) anticipated transient categories relevant to Quad Cities are grouped as one initiating event, with the exception of LOSP which was considered as a special initiator. Although the transient events begin as significantly different initiators, these events break down to the same basic components, namely, removal of decay heat from the core and containment heat removal.

The loss of heating, ventilation, and air conditioning (HVAC) was evaluated for potential to initiate a reactor trip. Loss of cooling for the outboard MSIV room due to a loss of 480 V Bus 18 or 480 V MCC 18-2 was included as an initiating event in the Modified IPE (see below). Many HVAC systems at Quad Cities are independent systems each with its own power supply connections and cooling system connections. No other HVAC related initiators were identified. Where HVAC was found to be important to the success of specific equipment, it was analyzed in the system model.

The Modified IPE added the following special initiators:

- Loss of 4 kV Buses 11, 12, 13, and 14. The frequency of loss of these buses was determined by adding a water spray contribution to the failure frequency based on a fault tree analysis.
- Loss of outboard MSIV room cooling due to loss of 480 V Bus 18 or due to loss of motor control center (MCC) 18-2. The frequency of loss of this bus and MCC was determined by adding a water spray contribution to the failure frequency based on a fault tree analysis.
- Loss of 125 VDC Reserve Bus 1B-1. The frequency of loss of this bus was determined based on a fault tree analysis.
- Loss of service water. The frequency of this special initiator was determined by adding a contribution from rupture rate estimates (for large service water system piping) to the failure frequency based on a fault tree analysis.
- Loss of instrument air. The frequency of this special initiator was based on the instrument air system failure rate calculated as part of the original IPE.

The frequencies for single unit LOSP and dual unit LOSP were calculated separately. The methodology and site specific values developed in NUREG-1032 for grid related losses, weather related losses, and extreme weather related losses were used to calculate the LOSP frequencies. The values for Plant Centered Loss (PCL) were calculated from generic data presented in NSAC-147 (Nuclear Safety Analysis Center, "Losses of Off-site Power at U.S. Nuclear Power Plants All Years Through 1989," March 1990) and NSAC-166 ("Losses of Off-site Power at U.S. Nuclear Power Plants All Years Through 1990," March 1991) for LOSP at dual unit sites. The generic PCL frequency was used in the Quad Cities analysis, since no true single-unit LOSP events have been experienced at the Quad Cities site.

4.1.2 Support System Modeling

The support state methodology was used to model the key support systems and their impact on the safety systems that are required to respond to the initiating events modeled. The concept of a support state model allowed the major support systems to be modeled outside of the accident sequence event tree structures. Quad Cities contains two units which share major support systems. Shared systems were modeled to ensure that the influence on both units is captured. An example of a shared system at Quad Cities is the service water (SW) system. This system contains pumps powered from both units. Therefore, to appropriately account for the influence of Unit 2 on this system, the Unit 2 power sources were considered in the SW model. The development of the support state model for a dual-unit site consisted of the following basic steps:

TABLE 4.1.1-1
SUMMARY OF QUAD CITIES INITIATING EVENT FREQUENCIES

FREQUENCY (/Y)	INITIATING EVENT
3.0E-04	Large Loss of Coolant Accident (LLOCA)
8.0E-04	Medium Loss of Coolant Accident (MLOCA)
3.0E-03	Small Loss of Coolant Accident (SLOCA)
1.2E-07	Interfacing Systems LOCA (ISLOCA)
1.06E-02 (1)	Inadvertent Open Relief Valve (IORV)
3.87E+00*	Anticipated General Transients (GTR)
3.2E-02*	Single Unit LOSP (LOSP)
1.61E-02*	Dual Unit LOSP (DLOSP)
(2)	Loss of all AC Power (Station Blackout)
1.16E-04 (3)	Anticipated Transient Without Scram (ATWS)
2.75E-04*	Loss of Bus 11 (LB11)
2.05E-04*	Loss of Bus 12 (LB12)
4.65E-04*	Loss of Bus 13 (LB13)
5.88E-04*	Loss of Bus 14 (LB14)
4.18E-04*	Loss of Bus 18 (LB18)
1.34E-04*	Loss of MCC 18-2 (L182)
1.01E-03*	Loss of 125 VDC Reserve Bus 1B-1 (L1B1)
9.10E-03*	Loss of Service Water (LOSW)
1.43E-02*	Loss of Instrument Air (LOIA)

(1) This event includes inadvertent opening of a relief valve (7.1E-02) and transfers from other event trees in which relief valves were challenged, opened but did not close. The summation of the latter contribution from all other events is 3.5E-02.

(2) This event is treated as a consequential failure in the accident sequence analysis and thus, no frequency was calculated.

(3) This event is a consequential failure, but the likelihood of a transient occurring (3.9/yr) and the likelihood of failure to scram (3E-05) were combined manually and input to the ATWS PRT quantification (1.16E-04).

* Plant-specific calculation.

TABLE 4.1.3-3
PLANT RESPONSE SUBTREES

ATW1	ATWS - INITIAL PHASE
ATW2	ATWS - FW AND MC SUCCESS, RPT SUCCESS, ARI FAILURE
ATW3	ATWS - FW SUCCESS, MC FAILURE, RPT SUCCESS, ARI FAILURE
ATW4	ATWS - FW FAILURE, MC SUCCESS, RPT SUCCESS, ARI FAILED
ATW5	ATWS - FW AND MC FAILURE, RPT SUCCESS, ARI FAILURE
ATW6	ATWS - ARI SUCCESS
ATW7	ATWS - OIADS, SLC AND HPCI SUCCESS, RHR HEAT REMOVAL FAILURE
ATW8	ATWS - OIADS AND SLC SUCCESS, HPCI FAILURE, RHR HEAT REMOVAL SUCCESS
ATW9	ATWS - OIADS AND SLC SUCCESS, HPCI AND RHR COOLING FAILURE
ATW10	ATWS - OIADS AND SLC SUCCESS, HPCI FAILURE, LPCI FAILURE
ATW11	ATWS - ARI SUCCESS, SPC SUCCESS, HPCI AND RCIC FAILURE
ATW12	ATWS - ARI SUCCESS, HPCI AND SPC SUCCESS, INVC FAILURE
ATW13	ATWS - ARI SUCCESS, HPCI/INVC -OR- RCIC SUCCESS, SPC FAILURE
ATW14	ATWS - ARI AND HPCI SUCCESS, RHR COOLING AND INVC FAILURE
ATW15	ATWS - ARI SUCCESS, HPCI, RCIC AND RHR COOLING FAILURE
ATW16	ATWS - ARI SUCCESS, HPCI/INVC -OR- RCIC SUCCESS, RHR FAILURE
ATW17	ATWS - ARI SUCCESS, HPCI/INVC -OR- RCIC SUCCESS, RHR PUMP FAILURE
ATW18	ATWS - ARI SUCCESS, HPCI, RCIC, AND RHR PUMP FAILURE
ATW19	ATWS - ARI SUCCESS, HPCI/INVC AND SPC SUCCESS
ILOC1	ISLOCA - INTERFACING SYSTEMS LOCA
IORV1	IORV - INITIAL PHASE
IORV2	IORV - FW, CRD, AND SSMP FAILURE, HPCI/RCIC AND SPC SUCCESS
IORV3	IORV - FW, CRD, SSMP AND RHR COOLING FAILURE, HPCI/RCIC SUCCESS
IORV4	IORV - FW, CRD, SSMP AND RHR PUMP FAILURE, HPCI/RCIC SUCCESS
IORV5	IORV - FW, HPCI, RCIC, AND SSMP FAILURE, SPC SUCCESS
IORV6	IORV - FW, HPCI, RCIC, SSMP, AND RHR COOLING FAILURE
IORV7	IORV - FW, HPCI, RCIC, SSMP, AND RHR PUMP FAILURE
LLOC1	LARGE LOCA - INITIAL PHASE
LLOC2	LARGE LOCA - CS SUCCESS, RHR PUMP SUCCESS
LLOC3	LARGE LOCA - CS SUCCESS, RHR PUMP FAILURE
LLOC4	LARGE LOCA - CS FAILURE, LPCI SUCCESS
LLOC5	LARGE LOCA - CS AND RHR INJECTION VALVE FAILURE, RHR PUMP SUCCESS
LLOC6	LARGE LOCA - CS AND RHR PUMP FAILURE
LOOP1	LOSP - INITIAL PHASE
LOOP2	LOSP - HPCI, INVC AND SPC SUCCESS
LOOP3	LOSP - HPCI OR RCIC SUCCESS, SPC FAILURE
LOOP3A	LOSP - HPCI/RCIC AND ADS SUCCESS, FW, CRD, SSMP AND SPC FAILURE
LOOP3B	LOSP - HPCI/RCIC SUCCESS, ADS, FW, CRD, SSMP AND SPC FAILURE
LOOP4	LOSP - HPCI/RCIC SUCCESS, RHR HEAT REMOVAL FAILURE
LOOP5	LOSP - HPCI/RCIC SUCCESS, RHR FAILURE
LOOP6	LOSP - SPC AND HPCI SUCCESS, INVC FAILURE
LOOP7	LOSP - HPCI SUCCESS, SPC AND INVC FAILURE
LOOP8	LOSP - HPCI AND RCIC FAILURE, SPC SUCCESS
LOOP9	LOSP - HPCI, RCIC, AND SPC FAILURE
LOOP10	LOSP - HPCI, RCIC, AND RHR FAILURE
MLOA1	MEDIUM LOCA - INITIAL PHASE
MLOA2	MEDIUM LOCA - OIADS, HPCI, RHR COOLING, AND CTM SPRAY SUCCESS
MLOA3	MEDIUM LOCA - OIADS AND HPCI SUCCESS, CTM SPRAY FAILURE

TABLE 4.1.4-1 (Continued)
QUAD CITIES PRT SUCCESS CRITERIA

Notes:

1. Operator actions are required to trip the recirculation pumps, start the SLC pumps, and inhibit ADS.
2. CRD is assumed to continue to operate throughout the event.
3. Successful LPCI injection includes operation of at least one RHR pump (LPA or LPB) and an injection valve (LV).
4. Recovered following restoration of offsite power. Initial inventory makeup provided by RHR or CS pumps.
5. Recovered following restoration of offsite power. Initial inventory makeup provided by HPCI or RCIC pumps.
6. As a result of adding water to the containment with the SBCS, NPSH will be maintained to the low pressure pumps even with a failure of containment heat removal.
7. With the exception of some success sequences dependent on SSMP, recovery of power is required for all SBO success sequences.
8. During a single unit SBO, the SSMP may be powered from the other unit and prevent core damage without recovery of power to the unit experiencing the SBO.
9. This success path included only for potential sensitivities and to allow for flexibility in future modelling modifications. No credit was given for success of these paths in the base quantification.
10. Containment status (success/failure) does not affect the success of the core.
11. The majority of decay heat is directed to the main condenser, therefore, containment pressure/temperature control is not required during first 24 hours.
12. Depressurization and low pressure inventory control is not required if high pressure injection is available.
13. If the operators successfully isolate the break (OIB) or the RHR relief valves reclose after successful depressurization, the event essentially becomes a transient and these sequences are assumed to terminate in a SAM endstate. Because of the extremely low initiating event frequency in conjunction with the variety of systems available to mitigate the accident, little error is introduced.
14. Success of SBCS includes operation of the feedwater system.
15. Model was not developed due to the low frequency of occurrence of this event.
16. The RPV inventory control function was not developed because termination of ISLOCA flow either by isolation of the break (OIB) or reclosure of the relief valves (CRV) results in an event very similar to a General Transient. The low initiating event frequency in conjunction with the many systems available to maintain inventory indicated that additional development of this function was unwarranted.

High Efficiency Afterfilter

The next component in the filter train is the high efficiency afterfilter, 1/2-A(B)-7504, which removes particulates and carbon dust which may be carried from the charcoal bed. The afterfilter is identical to the high efficiency prefilter.

Fans

The final component of the filter train is the fan. The fan provides the motive force for the system. It is located downstream of the process filters to minimize contamination. The fan design speed is 1800 rpm with a 20 horsepower, 3-phase, 60 Hz, 230/460V motor.

Augmented Primary Containment Vent System

The APCV system is used if the SBGT system is unable to produce adequate primary containment pressure control. If the SBGT flow capacity is not sufficient to control and maintain pressure below the primary containment pressure limit, then venting is performed through the 18-inch air-operated valves (AO1(2)-1601-60 or AO1(2)-1601-23) and valve AO1(2)-1601-24 to the APCV system. As shown in Figure 4.2.1.16-3, the inlet valve to the reactor building exhaust system, AO1(2)-1699-7, is normally open and must be closed before venting to the APCV system. The inlet valve to the APCV system, AO1(2)-1699-6, is normally closed and must be opened to permit venting. The APCV vent lines from Units 1 and 2 combine and a single line ties directly into the ventilation exhaust duct. The exhaust duct ultimately vents directly to the main chimney.

4.2.1.17 Instrument/Service Air Systems

The air systems of interest include the Instrument Air, Service Air, and Drywell Pneumatic systems. The Instrument Air system supplies clean, dry, compressed air to air-operated control devices and instruments throughout the plant. The Service Air system supplies a regulated source of compressed air throughout the station for breathing air and plant air drops. The Service Air system is of primary interest because it also provides an emergency backup supply of compressed air to the Instrument Air system. The Drywell Pneumatic system provides clean, dry, compressed air or nitrogen to pneumatically operated control devices in the drywell.

Instrument Air System

Instrument Air is a non-safety-related system that directly supports the operation of safety- and non-safety-related systems. The Instrument Air systems draw air from the Turbine Building using four rotary compressors connected in parallel. Two compressors supply Unit 1 loads (1-4709A and 1-4709B), while one supplies Unit 2 loads (2-4709). A shared compressor supplies both units (1/2-4709). The systems are connected through a cross-tie line that includes a normally closed and locked valve. Figure 4.2.1.17-1 shows a simplified diagram of the Instrument Air systems of Units 1 and 2, and Figure 4.2.1.17-2 shows a representative arrangement of the Service Water cooling supply to the Instrument Air compressor heat exchangers. Table 4.2.1.17-1 lists some of the important loads on the Instrument Air system.

Each of the four compressor trains dries and filters the air before storage in air receivers for distribution to system loads. Each compressor can supply 300 cfm at 105 psig. Clean Demineralized Water seals and cools each compressor, and also acts as the compressing agent. A separator removes entrained moisture from the compressor's output, and returns it to the compressor via a heat exchanger. The Service Water system supplies cooling water to the heat exchanger. A solenoid-operated valve controlled by the separator's water level allows makeup to each compressor-separator pair from the Clean Demineralized Water system.

An unloader valve on each compressor controls system pressure by opening when receiver pressure reaches 105 psig, thus equalizing pressure between the suction and discharge of the compressor. When receiver pressure drops to 95 psig, the unloader valve closes.

A low-pressure signal (85 psig) from either unit causes the shared 1/2 compressor to start automatically. When both units have maintained pressure above 110 psig for 15 minutes, the 1/2 compressor shuts off. The automatic stop feature of the 1/2 compressor occurs regardless of whether the compressor is operating due to a manual or automatic start signal, and is a feature of the 1/2 compressor only.

The outlet line of each separator contains a filter to remove moisture and particulates from the compressed air. The two cross-ties from the Service Air system connect with the 1A and 2 compressor trains just downstream of these separator outlet filters.

Each compressor train has two prefilters arranged in parallel, with one normally valved in service. The prefilter removes oil vapors and dirt particles using disposable cartridge filters. Next, the compressed air passes through a dual-chamber dryer filled with alumina beads. The chambers operate on a 10-minute drying/regeneration cycle. An automatic bypass feature on the dryers actuates when system pressure drops to 80 psig.

A pair of afterfilters arranged in parallel removes alumina dust from the compressed air stream. One of the two is normally valved in service. The air then flows to the 86-ft^3 air receiver associated with each compressor train (except for the 1/2 compressor train, which has no receiver).

STEP 3 Quantify Fault Tree

The fault trees developed in the previous task were quantified using the GRAFTER Code System to determine the system failure probability and to obtain the minimal cutsets. The calculational methods for quantifying the basic event probabilities that were input into the fault tree quantification are presented in the respective technical guideline. Calculational methods were described for hardware failures (both demand and time dependent), maintenance outages, test outages, human errors and common cause failures. A discussion of system mission times was provided and a component ID format was provided to maintain consistency within the analyses.

Step 3.1

Calculate basic event probabilities - Utilizing the component failure rates, test and maintenance unavailabilities and other basic event data, the basic event probabilities defined in the fault tree were quantified using the equations provided in the technical guidelines.

Step 3.2

Calculate human error probabilities - The human errors considered in the development of the fault trees and the human error probabilities used in the original IPE quantification of the fault trees were developed using the THERP methodology. As part of the Modified IPE, significant human error probabilities were re-quantified using the EPRI Cause Based Decision Tree methodology.

For the Modified IPE, "pre-initiator" type failures of standby instruments were reconsidered. Plant-specific data for a five-year time period was reviewed for instances of instrument miscalibration and other types of pre-initiator human errors. For all instruments, generic failure rates, together with appropriate standby mission times (one-half of the instrument calibration interval) were judged to appropriately represent failures, including out-of-calibration failure due to human errors.

Step 3.3

Calculate common cause failure probabilities - Once a fault tree for a system was developed, which includes random hardware failures, test outages, maintenance outages and human errors, the important common cause component groups were identified for inclusion in the fault trees. The common cause attributes that were used for the identification of common cause failures are:

- Component Type
- Component Use/Function (system isolation, flow modulation, etc.)
- Component initial conditions (i.e., normally closed, initially running, etc.)
- Component failure mode

For each common cause component group identified, common cause events were added to the fault tree at the component level. Once all important common cause failures were identified, the Multiple Greek Letter method was used to calculate the common cause failure probability.

With the common cause failure probabilities input into the fault tree, the fault tree was quantified to determine the total system failure probability and to obtain the dominant contributors (cutsets) for the system.

STEP 4 Document Process

The entire process of fault tree development including key assumptions, boundary conditions, and other important information was documented in the fault tree section of the system notebook. The quantification of the fault tree was also documented in the system notebook in the quantification section (including the computer code used and its input and output). The dominant contributors to system failure were identified and documented in the system notebook. The key insights were also documented in the system notebook.

Containment Venting

The Torus/Drywell Vent is designed to provide primary containment pressure control to prevent failure at the containment pressure boundary and uncontrolled releases of radioactive fission products to the environment. The Torus/Drywell Vent contains 18" vent pipes from both the drywell and torus gas spaces. Flow can be diverted from the 18" lines through a 2" vent valve to limit the discharge rate. Once extracted from containment, the vent flow can be directed through the Standby Gas Treatment (SBGT) System located in the reactor building at the 666'-6" elevation, or the Augmented Primary Containment Vent (APCV) system. The APCV or "hardened vent" provides an alternative 8-inch nominal diameter flow path from the 18" vent line (normally used for the reactor building exhaust system) to the Radwaste Ventilation Duct discharging directly into the main chimney. The flow diagram for various containment vent paths is illustrated in Figures 4.3-6 through 4.3.-6b. The emergency procedures instruct the operator to vent the primary containment if the primary containment pressure cannot be maintained below the Primary Containment Pressure Limit. According to this limit, venting must occur if the primary containment water level exceeds 93 feet or the torus bottom pressure exceeds 46 psig.

The SBGT system consists of a demister, an electric heater, an activated carbon iodine absorber, rough and high efficiency prefilters, a high efficiency afterfilter and a fan. The gas enters the demister which removes moisture and then is heated to lower the gas relative humidity. The gas is then filtered and passed through the activated charcoal bed which is capable of removing 100 grams of iodine. The gas stream is then filtered once again and blown out through the chimney.

If the SBGT flow is not sufficient to control or maintain torus gas space pressure below 20 psig, then the use of the 8" APCV line is directed, when torus gas space pressure reaches 46 psig. The selection of whether to vent from the drywell or the wetwell is dependent on the torus water level with the torus vent line being the preferred vent path.

4.3.1.3 Containment Data

A CECo-specific version of the Modular Accident Analysis Program, MAAP, (see Section 4.3.2) was used in the Quad Cities IPE to provide an integrated approach to the modeling of plant and containment thermal hydraulic response and fission product behavior during severe core damage accidents. All MAAP versions require plant specific input data, which is compiled into a MAAP parameter file. The parameter file provides a complete, realistic description of the plant for a MAAP simulation, and its data is consistently maintained for all accident sequences.

4.3.2 Plant Models and Methods for Physical Processes

The Quad Cities containment and source term analysis are part of an integral approach to the assessment of total plant response to accident initiators. The analysis looks at the whole spectrum of accident sequences. Since this integrated approach encompasses the traditional Level I and Level II analyses, it includes plant models which reflect the overall plant behavior prior to, and following core damage. This is accomplished by coupling a

4.4 Supporting Analysis

The following sections describe several analyses that support the quantification of the fault trees and the plant response trees. These supporting analyses include the generation of plant specific and the compilation of generic component data, the generation of human error probabilities, the generation of plant specific common cause failure probabilities, the identification of any internal flooding initiating events, and the identification of any equipment that may not survive the expected accident conditions. These analyses were completely documented in separate Quad Cities IPE/AM project notebooks.

4.4.1 Data Analysis

The purpose of the data analysis task was to collect data and obtain reasonable estimates of the failure rates and unavailabilities of basic components significant to the IPE. Random failure rates (including failure probabilities per demand), unavailabilities due to maintenance and testing, and common cause failure rates are the basic quantities that were evaluated extensively in the data analysis task. Testing was found to affect the unavailability of only a few systems analyzed (the High Pressure Coolant Injection and Reactor Core Isolation Cooling systems). These unavailabilities were calculated from the test frequencies and their average durations based on Quad Cities-specific experience.

At the onset of the data collection task, important key components were identified as likely to dominate or have an important impact on core damage frequencies, based on knowledge of previous PRAs. The list of key components for the Quad Cities IPE defined the scope of the major phase of the plant-specific data collection effort. The key component approach permitted resources to be focused on the most important failures and unavailabilities and enabled investigations to be made in greater depth when needed. Failure and unavailability data for key components for which plant-specific data was not available was obtained from generic data sources. Table 4.4.1-1 is a list of key components and failure modes. This list was used during the process of collecting plant-specific data.

Comparisons were made between key component failure data and generic failure data, and comparisons were also made between key component failure rates and failure rates and maintenance unavailabilities provided in the Dresden Nuclear Power Station Individual Plant Evaluation (IPE) initiating event and component data. Generally, the Quad Cities IPE data was comparable to both generic and the Dresden IPE.

The failure and component unavailability data collected for the original Quad Cities IPE spanned the period of January 1, 1985 through December 31, 1991. This period included seven years of plant operating experience. In general, the use of the seven years, particularly the most recent seven years, as the basis for data analysis is believed to be optimal. Seven years is a sufficient period of time for the calculation of most failure rates. More significantly, the most recent 7 year period (i.e., 1985 through 1991) should produce failure rate and unavailability results that come closest, in most cases, to the current true state of unreliability of the key components.

For the Modified IPE, new failure data was collected and analyzed for five key plant systems (HPCI, RCIC, Emergency Diesel Generators, Safe Shutdown Makeup Pump, and RHR) for the three year period of 1993 through 1995. Maintenance unavailability data for these same five systems was collected for the most recent two year interval (1994 and 1995). These time spans were judged to provide the best indication of recent performance for these key systems. Selection of these time spans is also consistent with the methodology being used for the ComEd plant PRA model updates.

Additional data for the Modified IPE was also gathered to search for pre-initiator type failures involving calibration of pertinent instruments. Records of events as preserved in the DVR and PIF databases were gathered for a five year period through the end of 1995. No "pre-initiator" type failures were identified. The review for the Modified IPE concluded that use of generic failure rates and mission times based on calibration interval was appropriate for standby instruments and adequately accounts for calibration failures due to human error.

TABLE 4.4.1-1
SUMMARY OF KEY COMPONENTS AND FAILURE/UNAVAILABILITY MODES

KEY COMPONENT	SYSTEM CODE[1]		FAILURE AND UNAVAILABILITY MODE[2]
Electrical Components			
Diesel Generators	DG	(66)	FTS, FTR, Unavailable due to Maintenance
Diesel Generator Output Breakers	DG	(66)	FTO, FTC, Spurious Opening, Unavailable due to Maintenance
4KV Breakers	AC	(67)	FTO, FTC, Spurious Opening, Unavailable due to Maintenance
Batteries (125/250VDC)	DC	(83)	FTF, Unavailable due to Maintenance
Battery Chargers (125/250VDC)	DC	(83)	FTF, Unavailable due to Maintenance
Mechanical Components			
Valves			FTO, FTC, Unavailable due to Maintenance
Motor Operated	All		
Air Operated	All		
Safety Related	All		
Pumps			FTR, FTS, Unavailable due to Maintenance
Control Rod Drive Hydraulic	CR	(03)	
Core Spray	CS	(14)	
Diesel Generator Cooling Water Pump	DG	(66)	
Feedwater and Condensate	FW	(32, 33, 34)	
Fire Protection	FP	(41)	
High Pressure Coolant Injection	HI	(23)	
Reactor Building Closed Cooling Water	RB	(37)	
Reactor Core Isolation Cooling	RC	(13)	
Residual Heat Removal	RH	(10)	
Residual Heat Removal Service Water	RS	(10)	
Safe Shutdown Makeup	SS	(29)	
Service Water	SW	(39)	
Standby Liquid Control	SL	(11)	
Turbine Building Closed Cooling Water	TB	(38)	
Miscellaneous			FTF, Unavailable due to Maintenance
Instrument Air	IA	(47)	
Residual Heat Removal Heat Exchanger	RS	(10)	
Standby Gas Treatment Fans & Dampers	GT	(75)	
Turbine Building Closed Cooling Water Heat Exchanger	TB	(38)	
HPCI Room Cooler	HI	(23)	
Service Water Strainer	SW	(39)	
Core Spray Room Cooler	CS	(14)	

Notes:

1. System Code Represent
   - The Code used in the database
   - The Quad Cities Plant System Code
2. Fails to Open (FTO), Fails to Close (FTC), Fails to Start (FTS), Fails to Run (FTR), Fails to Function (FTF).

Plant-specific data was collected from the operating records of both units and was combined to form one data base. The advantage of this approach is that the length of record available for plant-specific failure rate estimation is double what the record would be if the units were treated separately. The dual-unit data base results in better average (point estimate) failure rates for the key components. Another reason for the combined data base was that no significant differences between the components of Unit 1 and Unit 2 were identified. Therefore, no basis was found for pursuing the hypothesis that the unreliability of Unit 1 components could be different from the unreliability of Unit 2 components. For key components which lacked enough operating experience during the 7-year period to provide a usable estimate of failure, generic data was used.

In accordance with the CECo Guidelines, failure rates were calculated as point-estimate values. One type of point estimate failure rate (hourly failure rate) is simply the number of failures that occur during a particular period of component operation divided by the operating hours of the component. This type of point estimate was used to calculate failure to run of components such as pumps and diesel generators. Another type of point-estimate failure rate is the number of failures during a particular period of time divided by the number of component demands that occurred during the same period. This type of point estimate was used to calculate the failure rates of components failing to start, and motor-operated valves failing to open or close.

Criteria was established for the plant-specific analysis. The purpose of the criteria was to distinguish failures from non-failures, and to screen out maintenance events that would not contribute to maintenance unavailability. The type of maintenance unavailability desired was the unavailability that would be applicable during times when components could be required to operate for all modes of plant operation. Failures that occurred during cold shutdown were not arbitrarily screened out for two reasons. First, to get statistically significant records for components which are normally operating only during cold shutdown, cold shutdown periods must be included. Second, equipment failures are assumed to occur randomly in time. They are assumed to occur no more or no less frequently during plant operation than during plant shutdown. Therefore, the failure data base considered failures during all modes of operation and shutdown.

The boundaries of each component were also considered in the screening of failures and maintenance events. A table of component boundaries in the CECo guidelines indicated which subcomponents to include with the main component. For example, circuit breakers and handswitches were included within the boundaries of pumps, and failures of the subcomponents were counted as failures of the pump.

The sources of plant-specific data are presented in annotated form in Table 4.4.1-2.

TABLE 4.4.1-2
SOURCES OF PLANT-SPECIFIC DATA UTILIZED IN DATA ANALYSIS

PLANT SPECIFIC DATA SOURCE DESCRIPTIONS	
1.	Deviation Reports (DVRs - prior to mid-1993): Events such as malfunctions of safety-related components, forced shutdowns, and scrams are generally reported in DVRs. DVRs include the date, time, plant operating mode, description of the event, and the date and time when the equipment was returned to service. Each DVR is assigned a number which indicates the unit, year of occurrence, and sequence number. A DVR index is maintained for each unit.
2.	Problem Identification Forms and Problem Investigation Reports (PIFs and PIRs - after mid-1993): Replaced the DVRs but serve the same function and record the same information. PIFs have a lower threshold for reporting than did DVRs, so equipment malfunctions and human errors of all levels are recorded. PIF and PIR indexes are maintained for the site.
3.	Licensee Event Reports (LERs): LERs are prepared for events that are determined to be reportable after CECo evaluation. The LER describes the event and includes the cause, corrective action, safety analysis, and brief history of previous occurrences. The initial submittal of an LER is sometimes followed up with later supplemental reports that provide further clarification, analysis, or corrective action. LERs are filed at the Quad Cities site as attachments to the DVRs.
4.	Nuclear Plant Reliability Data System (NPRDS): Queries of this computer data base were used as comparison only since the major source for this database (work requests) were reviewed.
5.	Procedures: All applicable Quad Cities Procedures (QGP, QIS, QMPM, QTP, QTS, QOS, QOP, QCOP, QEPM, QCOS, QCOA) were reviewed to obtain demands and unavailabilities on key components. The procedures generally describe, in a step-by-step manner, actions taken during testing/surveillances. The unavailabilities due to testing were accounted for from these procedural reviews.
6.	Monthly Operating Reports: These reports give an overall monthly status of each plant. Included in the status report are the monthly operating hours for each plant and any unusual occurrences which took place during the reporting period.
7.	Emergency Diesel Generator Reliability Program: This is a compilation of Emergency Diesel Generator operability experience and was used to determine the diesel operating time.
8.	Total Job Maintenance Data System (TJM): Queries of this computer database were made for all key components, and information was retrieved by system. This information included the equipment part number, event description, cause, corrective action, and repair time.

NUREG/CR-2815 was the primary source of generic failure rate data. NUREG/CR-2815 was the first source consulted and was used except in cases where it did not provide data for the particular failure mode needed or where some other source was determined to provide more relevant data. NUREG/CR-4550 was the primary source of generic maintenance unavailability data. Generic component failure data from NUREG/CR-4550 was also used for producing a Bayesian update estimate for several pieces of equipment that had not experienced a failure during the three year data collection period for the Modified IPE. The following sources of generic data were also used in the data analysis task:

- IEEE Std. 500-1984, "IEEE Guide to Collection and Presentation of Reliability Data for Nuclear Power Generating Stations," IEEE Power Engineering Society, March 21, 1984.

- NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide," January 1983.

The plant-specific failure rates and maintenance/test unavailabilities are given in Tables 4.4.1-3 and 4.4.1-4 respectively. Table 4.4.1-5 contains the plant-specific common cause failure rates and probabilities. A discussion of the common cause analysis is presented in Section 4.4.3 of this report.

The generic failure rates and unavailabilities are presented in Table 4.4.1-6.

TABLE 4.4.1-3

QUAD CITIES-SPECIFIC COMPONENT FAILURE RATES

(IN VOLUME 2)

TABLE 4.4.1-4

SUMMARY OF QUAD CITIES-SPECIFIC MAINTENANCE UNAVAILABILITIES

(IN VOLUME 2)

TABLE 4.4.1-5

COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

(IN VOLUME 2)

TABLE 4.4.1-6

QUAD CITIES IPE GENERIC FAILURE AND UNAVAILABILITY DATA

(IN VOLUME 2)

4.4.2 Human Reliability Analysis

The human reliability analysis (HRA) for the Quad Cities probabilistic risk assessment (PRA) provides human error probabilities (HEPs) for use in quantifying the fault trees and plant response trees (PRTs) which comprise the model.

The original HRA was performed in two phases. The first phase of the analysis used the THERP (Technique for Human Error Rate Prediction) method to determine the HEPs for operator actions in the PRA model. The second phase consisted of verifying assumptions made during phase one through the use of simulator exercises and interviews with Quad Cities operations and training personnel. The significant operator actions were re-evaluated utilizing a newer alternate technique. The selected technique was the EPRI CBDTM (Cause Based Decision Tree Methodology) from EPRI TR-100259.

4.4.2.1 Description of the HRA Methodology

The fault tree and PRT (plant response tree) analysts identified operator actions during the development of the trees representing the Quad Cities PRA model. These operator actions were then analyzed and quantified by human reliability analysts. The process was an iterative one, to ensure that the assumptions made by the PRT analysts and the HRA analysts were consistent.

The HEPs within each of the Plant Response Trees were analyzed on a sequence-by-sequence basis to identify the conditions of stress, dependency, and availability of recovery opportunities. This sequence by sequence evaluation generated multiple cases for each of the operator actions.

Only one case is considered for operator actions modeled in the fault trees since the various conditions under which the action may occur can not be predetermined.

4.4.2.1.1 Quad Cities Characteristics and Assumptions Important to the HRA

Shift Manning

Shift manning as well as the role of each member of the shift during accident conditions was discussed with members of the Quad Cities Operations Department. This information was used to determine the applicable opportunities for error recovery and to determine the level of dependency between members of the crew. Error recovery and dependency are further discussed below.

Applicable Procedures

The procedures which are used during the course of an accident or transient at Quad Cities include the Generating Station Emergency Plan (GSEP), the Quad Cities Emergency Operating Procedures (QGAs), annunciator procedures, abnormal procedures, and operating procedures.

It should be noted that the QGAs are in a flow-chart format, and provide very general guidance for the operators to follow based upon plant symptoms. This is consistent with the emergency operating procedures at other BWRs. Systems to be used to fulfill a desired function are specified. The Quad Cities QGAs do not provide step-by-step, proceduralized actions necessary to accomplish the desired action. Therefore, line up of systems directed by the QGAs is accomplished from memory by the operators, without initial reliance on procedures. Since many of the QGA actions are simple (i.e., start pump, open valve), these activities are considered well within the skill of the craft.

Use of the abnormal and operating procedures during QGA execution is discouraged; it is considered that the QGA actions are well practiced and use of these procedures would tend to slow down implementation of the required function. However, the operators are expected to consult the procedures to verify proper system line up as time permits (for example, when the accident has reached a stable condition). This represents a recovery opportunity which is dependent upon enough time being available, and is included in the model as a possible recovery.

The operators are trained to perform most actions directed in the QGAs from memory. Job Performance Measures are used to accomplish this training. For this reason, the Job Performance Measures were used to determine the subtasks for actions which are performed from memory.

Command and Control

During an accident situation, each member of the shift staff has a definite role to perform. As of August 1996, the shift staffing consists of:

- one Shift Engineer (SRO)
- two Unit Supervisors (SRO)
- one Shift Technical Advisor (STA - SRO)
- one Field Supervisor (SRO)
- four Nuclear Station Operators (NSO - RO) - two on each unit
- four High Voltage Operators
- three Equipment Attendants

The normal control room complement consists of the two Unit Supervisors and the four Nuclear Station Operators.

During an event, the Unit Supervisor on the affected unit will direct all unit control room and in-plant activities. The STA and the Shift Engineer will enter the control room and perform an overview of plant conditions and activities. The Shift Engineer is responsible for the coordination of the GSEP and, ultimately, has overall responsibility for the plant. The NSO on the affected unit will initially operate the control rods and nearby equipment (i.e., feedwater, recirculation control, etc.). The extra NSO will assist on the ECCS panels or BOP panels, wherever needed. One of the unaffected unit's NSO may assist if the need arises. The remaining NSO will maintain control of the unaffected unit and provide control of the unaffected unit's equipment that may be needed to support the affected unit.

During Emergency Operating Procedure use, the Unit Supervisor is the designated procedure reader and is responsible for ensuring actions required by the Emergency Operating Procedures are properly implemented. This action verification was considered a possible recovery.

Control Room Human Factors

The Quad Cities control room is well designed from a human factors standpoint. The main control board layout is based upon functional groupings. Color coding is used to separate trains and systems. Mimic buses are used to aid the operator in the selection of pumps, valves, electrical power sources, and breakers. This greatly facilitates establishing flow paths or providing electrical power.

Indicator lights are provided adjacent to the control switches for pumps, valves, breakers, and other components. A "green board" concept is employed, such that the indicator lights are green for normal or expected conditions and red to indicate deviation from the norms.

4.4.2.1.2 General Approach

The analysis of a given operator action requires knowledge of the detection, diagnosis, decision, and action execution steps associated with the accomplishment of the action. Additionally, performance shaping factors, recovery, and dependency must be incorporated. To properly address these factors, it is necessary to consider the operator action in the context of the accident sequence.

Determination of Human Error Probabilities

Each human interaction can be divided into a cognitive phase and an execution phase, namely.

$$HEP = Pc + Pe$$

where

Pc = the probability of failure to initiate the correct response

Pe = the probability of failure to execute the response correctly.

Each part (Pc & Pe) utilizes a common approach: a search for error modes that could result in the error, and a search for potential recovery.

Estimation of Pc

The methodology used to estimate Pc is explained in detail in EPRI TR-100259. The methodology assesses Pc by evaluating a series of decision trees that evaluate failure mechanisms such as: data availability, attention failure, miscommunication and misreading of data, misleading information, missing or misreading procedure steps, misinterpretation of instructions or decision logic, and deliberate violations.

Use of the EPRI decision trees requires an understanding of the quality of information, procedures, training, and similar attributes. The time variable enters the analysis through application of "recovery factors" such as extra crew, STA review, staffing of the Technical Support Center and shift change. Specific allowable values for recovery factors are provided in EPRI TR-100259.

Estimation of Pe

The action execution portion of a given operator action encompasses those steps which must be accomplished to complete the desired action. The steps required to accomplish each operator action were determined based upon what the operator must do to meet the specific success criteria utilized in the PRA. It should be noted that a procedure may specify more than just the minimum required steps to meet the PRA success criteria. For the purposes of determining human error probabilities, only the steps in the procedure which were important to accomplishing the function based upon the success criteria were considered. Verification steps were considered as a potential for error recovery.

For QGA actions which do not explicitly provide direction to the operator, it is assumed that the operator is acting from memory. In this case, it is not the operating procedure steps which are important, but those steps which the operator recalls. Therefore, it is assumed that the operator will respond to the requirements of the QGAs based upon how they have practiced the evolutions during training. For this reason, the Job Performance Measures (JPM) were utilized instead of the operating procedures for these types of actions.

Action execution errors in the Quad Cities HRA are of two types: Omission errors and commission errors. The omission errors involve missing procedure steps or, in the case of operating from memory, the failure to recall non-written instructions. Commission errors include selection of the wrong switch, control, or gage, as well as improper operation of controls given that the correct one has been selected. The nominal HEPs for errors of omission and commission were taken from the appropriate table in NUREG/CR-1278 Chapter 20.

Determination of Cases

In general, a different case of a given operator action is evaluated to account for differences in stress level, differences in available recovery factors, or differences pertaining to the success or failure of preceding operator actions (dependency).

Stress

One of the major performance shaping factors in the analysis is stress. Factors such as lighting, noise levels, control board ergonomics, and administrative controls at Quad Cities are considered nominal.

The factors considered when determining the stress level include the time available to perform the action, the amount of activity during that time, and the availability of systems and components. In general, greater workload, more equipment failures and shorter time frames for performing an action were considered to result in higher stress.

The stress levels employed in the analysis were taken from NUREG/CR-1278, Table 20-16. Stress levels were taken to be either optimal, moderate, or high. When optimal stress applied, the nominal HEPs from the appropriate NUREG/CR-1278 table were used for each identified error opportunity. If stress was considered to be moderate, the nominal HEPs were multiplied by a factor of 2; if stress was high, the nominal HEPs were multiplied by 5. The stress factors were only applied to the Pe portion of the action, the CBDTM technique includes stress in the determination of Pc.

Recovery

When errors are made in diagnosing or performing an action, there is the possibility that the error will be detected in time to recover the error and continue with the accident mitigation. Some of the recovery opportunities credited in this analysis include the following:

- a procedure step directing the operator to verify that the system is performing its intended function
- a second alarm at another location in the control room that a different crew member may acknowledge
- an alarm that would indicate that the action had not been performed correctly, assuming that there would still be sufficient time to accomplish the action

The non-recovery probabilities associated with these recovery opportunities are taken from the appropriate table in NUREG/CR-1278 Chapter 20.

Additional recovery opportunities are addressed within the CBDTM, which credits the review of the event progression by a person or persons who had not been intimately involved with mitigation of the event up to that point. Recoveries of this type include the STA and Shift Engineer becoming available to focus on the event in progress, manning of the Technical Support Center, arrival of off-duty personnel to assist in accident mitigation and recovery, and/or arrival of a relieving crew. It was deemed that these recoveries, as identified in EPRI TR-100259, were appropriate for both the cognitive and execution portion of the HEPs.

Dependence between Operator Actions

Since the same crew must detect, diagnose, decide, and act upon all actions which take place in the scenario, it is reasonable to expect that operator actions are not necessarily independent events. If an operator action within a PRT sequence was preceded by an earlier failed operator action, this was evaluated and any impact on the calculated human error probability (HEP) for the latter event was determined. The method for adjusting the HEP was to assign a dependency of the second event upon the first.

The methodology used in assessing dependencies is discussed below.

There are two places where dependency must be considered. The first is when the crew must perform more than one operator action in a given event sequence. The second involves the dependency among the subtasks which comprise any single operator action.

When a given event sequence involved more than one operator action, the dependency of each operator action in a plant response tree (PRT) upon the previous operator action was considered.

Within a given operator action, the subtasks may also be interdependent. Dependency between subtasks of an operator action was determined and the HEPs modified accordingly.

In assessing dependencies, the five categories presented in NUREG/CR-1278 were utilized: Complete, high, moderate, low, and zero dependency. The formulas for the conditional probability of failure on task "n," given failure of previous task "n-1" for each level of dependence, are presented in Table 20-17 of NUREG/CR-1278. These were utilized to appropriately modify the HEP for any given operator action or subtask.

Quantification

For the Modified IPE, significant operator actions were re-evaluated. Only those actions which had a Risk Achievement Worth (using the original IPE model) of greater than 2.5 and those actions which were added as a result of changes to the Plant Response Trees and Fault Trees, received a complete evaluation utilizing the EPRI CBDTM. The remaining HEPs were reviewed for reasonableness of values and for the selection of the appropriate value for each branch of the PRTs.

The EPRI CBDTM technique provided values for two operator actions, OHX (Operator action to align cooling to RHR) and OSPC (Operator action to initiate suppression pool cooling), which appeared overly conservative. The action OHX contains two pump starts and two valve manipulations. The action OSPC contains a single valve manipulation. These actions are practiced simultaneously during simulator exercises and multiple times during the year to keep the suppression pool cooled. Also, these actions are directed to be performed early in the events (5-10 minutes), but the need does not arise until much later (5-24 hours) into the event. There are multiple indications and alarms on suppression temperature and containment pressure so there are multiple opportunities to recover the actions. The CBDTM technique does not adequately treat actions which can be delayed for hours and have multiple recovery opportunities. After reviewing similar actions from other IPEs and discussions with the operators and instructors, a value of 1.0E-5 (for both OHX & OSPC) was assumed for all initiating events except for ATWS, SBO and Loss of DC. For these three initiators a value of 1.0E-4 (for both OHX & OSPC) was assumed to properly reflect the added stress and the possibly shortened response times.

4.4.2.2 HRA Results

All of the HEPs after the re-evaluation have values above 1.0E-4, except for OHX and OSPC as discussed above. As can be seen in the summary of the HEP evaluations (Tables 4.4.2-1 and 4.4.2-2 below), some of the actions changed significantly, some increasing and some decreasing. Table 4.4.2-1 lists the Quad Cities Fault Tree Operator Actions and Table 4.4.2-2 lists the Quad Cities Plant Response Tree Operator Actions. Listed for each operator action are the node case name, the description, the Risk Achievement Worth, the original HEP value, the revised HEP value and the percent difference in the HEP. (Note: after each set of operator action cases there is a summary line that gives the combined Risk Achievement Worth for all the cases.)

The results are presented in Tables 4.4.2-1 and 4.4.2-2. Note that more than one case was identified for many of the operator actions; as discussed above, this was to account for differences in stress, dependency, and recovery on a sequence-by-sequence basis.

4.4.2.3 Time Considerations

During the Human Reliability Analysis, the time available to perform an action was compared to the time required to complete the action. This comparison was performed to determine the feasibility of the action and to assess the feasibility of recovery given an initial operator error.

The times available for the action were determined from MAAP (Modular Accident Analysis Program) runs which established the times for the cues and the time for action completion. The time required for the detection, analysis, diagnosis, decision and action was determined by discussions with operators and operator trainers, observation of simulator runs, and judgment by the analyst.

Each modeled operator action and the associated performance time (Pt) and available times (At) are presented below.

1. **OAD1** Initiate Automatic Depressurization

   This action is to depressurize the RPV (Reactor Pressure Vessel) to permit injection by the low pressure injections systems and is required for all initiating events except for Inadvertent Open Relief Valve and Large LOCA. Depressurization is required prior to the RPV level reaching the top of active fuel (TAF).

   Pt < 1 minute
   At ~ 5 minutes for a Medium and Interfacing System LOCA without any high pressure injection
   > 30 minutes during other initiators (LOOP, SLOCA, Transient, LOIA, LOSW, Loss of AC buses, ATWS) without any high pressure injection
   > 60 minutes for a medium LOCA with HPCI or Feedwater success
   > 4 hours during other initiators (LOOP, SLOCA, Transient, LOIA, LOSW, Loss of AC buses, ATWS) with HPCI, RCIC, or Feedwater success

2. **OAD2** Depressurize per the Containment Heat Capacity Limit

   This action is to depressurize the RPV to remain within the limits of the containment HCL curve and is required for all initiating events except for Inadvertent Open Relief Valve, Medium and Large LOCA. This action is not required if heat is being removed from containment by the RHR system or the main condenser.

   Pt < 1 minute
   At > 4 hours during all applicable scenarios

3. OAL Reduce and Control Reactor Pressure Vessel Water Level (ATWS)

This action is to lower reactor water level to limit reactor power thus reducing the heat input to the Suppression Pool during an ATWS event. This action is only required if feedwater is operating and controlling reactor water level. If feedwater is not available, then HPCI will actuate and control water at a lower level. If the main condenser is not available this action is not credited.

Pt ~ 2 minutes
At ~ 10 minutes if the main condenser is available (insufficient time if the main condenser is not available)

4. OAT Manually Initiate Alternate Rod Insertion (ATWS)

This is an immediate action in an ATWS event. After noting that the control rods have not successfully scrammed, the operator manually initiates alternate rod insertion. This action is only credited if the main condenser is available. If the condenser is unavailable, there is insufficient time to prevent over-pressurization of the primary system by actuating the ARI (Alternate Rod Insertion) system.

Pt < 1 minute (about 1 minute into the event)
At ~ 10 minutes if the main condenser is available (insufficient if the main condenser is not available)

5. OCD Cooldown with the Main Condenser

This action is to cooldown to the main condenser at a rate of 100°F/hr using the main turbine bypass valves and is only modeled in the small LOCA event tree. The operators initiate a cooldown in a LOCA situation to limit the mass and energy releases to the containment and to minimize the RPV makeup requirements. This action would be initiated after RPV level and pressure have been stabilized near their normal bands.

Pt ~10 minutes to establish a steady cooldown rate
At >1 hour

6. **OCNTS** Initiate Containment Spray

This action is modeled in all initiators except for Interfacing System LOCA. This action is required to maintain containment pressure below the primary containment pressure limit so that containment venting is not required. The sprays also provide scrubbing of radionuclides from the containment atmosphere. The cue to start the spray happens at the initiator with the Small, Medium and Large LOCAs. The condition for starting sprays for the other initiators usually does not occur until hours into the accident sequence.

Pt ~ 2 minutes
At > 4 hours if high pressure systems were not successful
> 6 hours if HPCI was successful
> 20 hours to maintain containment pressure below the primary containment pressure limit

7. **OCRD** Restore CRD Injection

This action to restart a single CRD pump as an injection source is only modeled following restoration of power after a Loss Of Offsite Power event. This action can be performed entirely from the control room.

Pt < 1 minute
At > 4 hours after event initiation. CRD flow is only credited after other injection sources have operated successfully for several hours

8. **OCST** Align Low Pressure Pump Suction to the CCST

This action is to align at least one RHR or core spray pump to take suction from the CCST (Contaminated Condensate Storage Tank). This action would occur if torus temperature increased to the point that minimum NPSH requirements for the low pressure pumps were not met and is modeled in the transient, IORV, SBO, LOOP and ATWS PRTs. The action is not credited in the LOCA events.

Pt ~20 minutes
At > 4 hours

9. **OFW1** Restart Main Feedwater Pumps

This action is to first restart feedwater pumps and restore feedwater flow after a high level trip following a reactor scram then restore and maintain RPV water level in the normal band. OFW1 is modeled in the transient, small break LOCA, and IORV PRTs.

Pt ~ 2 minutes
At > 30 minutes without any other high pressure injection source

10. OFW2 Manually Scram and Control Feedwater

This event models the operator actions needed to operate the feedwater system to control RPV level and then initiate a manual scram following indications of a LOCA outside containment. These operator actions are modeled in the interfacing systems LOCA PRT for scenarios where the RHR piping remains intact and coolant loss is through RHR system relief valves.

Pt < 5 minutes
At ~ 1 hour for the scenarios in which OFW2 is modeled

11. OFW3 Restore Feedwater Injection

Restore Feedwater Injection

This action is only modeled in the LOOP and SBO events and includes those actions necessary to restart the feedwater and condensate systems following an extended shutdown. Due to the time required to fill and vent the condensate system after a complete system shutdown, this action is assumed to have a failure probability of 1.0 in the time period of concern in the PRA models.

12. OHX Align Cooling to RHR

OHX is used in all PRTs except the ISLOCA event. The action is required whenever there is an initiation of LPCI, shutdown cooling, suppression pool cooling, or containment sprays. Success of the action requires that one RHR service water pump be started in an operable RHR loop and the associated RHR heat exchanger bypass valve be closed. The action must be completed while there is still sufficient NPSH available to the low pressure pumps.

Pt ~ 5 minutes (start ~ 5 minutes into the event)
At > 4 hours if high pressure injection systems failed
> 6 hours if high pressure injection systems were successful
> 20 hours to avoid containment venting

13. OIADS Inhibit Automatic Depressurization

This node refers to operator action to inhibit automatic actuation of ADS. The operator is instructed to inhibit ADS if unable to maintain RPV level above -59 inches. If the operators do not inhibit ADS the subsequent RPV blowdown will cause steam pressure to drop below that required to support operation of RCIC. ADS is inhibited because it imposes a severe transient on the RPV and can complicate efforts to restore and maintain RPV level. This node is modeled in the medium and small LOCA and ATWS PRTs.

Pt < 1 minute
At ~ 2 minutes after the simultaneous high containment pressure (>2 psig) and low-low RPV level (-59 inches) alarms during LOCA or ATWS events

14. OIB Isolate the Break

This action is used only in the Interfacing System LOCA event and refers to the actions to close the normally open valves 1001-28A and 1001-28B. The ISLOCA is assumed to take place in the LPCI system cooldown line with reactor coolant release to the reactor building. No training is conducted on ISLOCA scenarios at power and personnel do not believe the scenario is credible. Therefore, this action is assumed to fail with a probability of 1.0.

15. ORP Trip Recirculation Pumps

During ATWS sequences, this action reduces the reactor power to limit the heat input into the Suppression Pool. ORP is only credited if the main condenser is available -- if the condenser is unavailable, there is insufficient time to trip the ARI system to prevent over-pressurization of the Primary system.

Pt ~ 1 minute (about 1 minute into the event)
At ~ 10 minutes if the main condenser is available (insufficient time if the main condenser is not available)

16. OSBCS Initiate Standby Coolant Supply System

The action to initiate standby coolant supply (SBCS) is modeled in the large, medium, and interfacing system LOCAs. The actions include the pump and valve manipulations necessary to supply raw service water to the condenser and to use the condensate system to inject this water into the reactor vessel. The cue to initiate SBCS is given in the QGAs, when it is determined that the RPV level cannot be maintained above the top of active fuel or that RPV level cannot be determined.

Pt ~ 5 minutes
At insufficient time to prevent core damage for Large LOCA without other ECCS pumps initially injecting
~ 5 minutes for Interfacing System LOCA with no other injection
> 4 hour if a Low pressure ECCS pump can inject water initially

17. OSDC Initiate Shutdown Cooling

This action is modeled in the transient, SBO, and LOOP PRTs after plant pressure has been reduced to less than 100 psig. With pressure less than 100 psig, the operators align the RHR pumps to take suction from and discharge to the RPV recirculation piping. Actions to align cooling to the RHR heat exchangers is modeled separately in the OHX node.

Pt ~ 2 hours
At > 5 hours in those sequences where SDC is modeled

18. OSL1 Initiate one Standby Liquid Control Pump

This action is to initiate standby liquid control (inject boron) with at least one injection pump and is modeled in an ATWS event. The operators are required to start initiating the injection prior to the Suppression Pool reaching 110 degrees. Additional time for completion of this action would be available if feedwater flow is terminated or controlled to lower the RPV level to the TAF.

Pt < 1 minute (about 1 minute into the event)
At < 1 minute if main condenser is unavailable and Recirc pump trip failed
1.3 minutes if main condenser is unavailable and Recirc pump trip is successful
4.2 minutes if main condenser is available and Recirc pump trip is successful

19. OSL2 Initiate two Standby Liquid Control Pumps

This action is to initiate standby liquid control (inject boron) with both injection pumps and is modeled in an ATWS event. The operators are required to start initiating the injection prior to the Suppression Pool reaching 110 degrees. However, if two injection pumps are used, additional time is available without exceeding the heat capacity limit of the suppression pool. Additional time for completion of this action would be available if feedwater flow is terminated or controlled to lower the RPV level to TAF, then reactor power is reduced thereby reducing the heat input to the suppression pool. If the main condenser is available, the reactor recirculation pumps were tripped and the RPV level were lowered to the TAF, then the reactor power would be reduced to the point that no heat is being added to the Suppression Pool

Pt < 1 minute (about 1 minute into the event)
At ~ 5 minutes if main condenser is unavailable and Recirc pump trip failed
> 5 minutes if main condenser is unavailable and Recirc pump trip is successful
~ 10 minutes if main condenser is available and Recirc pump trip is successful

20. OSMP1 Initiate Safe Shutdown Makeup Pump with Suction Aligned to the CCST - No HPCI Injection Signal Present

This action includes the steps necessary to provide RPV injection with the SSMP and is included in the transient, IORV, SBO, LOOP, and ATWS PRTs. With the SSMP normally aligned to take suction from the CCST, the operators must manually align the SSMP to inject through the HPCI discharge line then start the pump and control RPV level in the -142 to +48 inch band.

Pt < 5 minutes
At > 30 minutes (Assumes all other injection sources have failed - additional time would be available if other injection sources were previously operating)

21. OSMP2 Initiate Safe Shutdown Makeup Pump with Suction Aligned to Fire Protection - No HPCI Injection Signal Present

This action is similar to OSMP1 with the inclusion of additional steps to align pump suction to the fire system. This action is included in the transient, SBO, LOOP, and ATWS PRTs.

Pt ~ 15 minutes
At > 30 minutes (Assumes all other injection sources have failed - additional time would be available if other injection sources were previously operating)

22. OSMP3 Initiate Safe Shutdown Makeup Pump with Suction Aligned to the CCST - With HPCI Injection Signal Present

This action is similar to OSMP1 with the inclusion of additional steps needed to align the injection flowpath through the HPCI lines while an injection signal is present. This action is included in the transient, small LOCA, SBO, LOOP, and ATWS PRTs.

Pt ~ 10 minutes
At > 30 minutes without any high pressure injection
> 4 hours with HPCI, RCIC, or Feedwater initial success

23. OSPC Initiate Suppression Pool Cooling

This action is modeled in all initiators except for Interfacing System LOCA and is required to remove containment heat and to prevent the loss of the low pressure ECCS pumps due to insufficient NPSH. The cue to initiate suppression pool cooling happens early in the event (about 5 minutes), but the action can be delayed for hours.

Pt ~ 5 minutes (start ~ 5 minutes into the event)
At > 4 hours if high pressure injection systems failed
> 6 hours if high pressure injection systems were successful
> 20 hours to avoid containment venting

24. OSS Restore Support Systems

This action is modeled in the LOOP and SBO events and includes the steps necessary to restart the service water and turbine building closed cooling water systems following a loss and subsequent recovery of offsite power. This action is required to be completed prior to the restart of the feedwater, condensate, or CRD systems.

Pt < 5 minutes
At > 4 hours if any high pressure injection source was successful
> 6 hours if depressurization and any low pressure injection source was successful

25. OVNT Initiate Containment Venting

The containment is vented to avoid over-pressure failure. The need for venting can be precluded by successful containment heat removal. The need for venting is indicated by containment pressure approaching 46 psig. Containment failure is not expected to occur until containment pressure approaches 124 psig, therefore, venting may be delayed for a significant time period. Containment venting is modeled in all PRTs except the interfacing systems LOCA.

Pt ~ 30 minutes (at more than 20 hours into the event)
At > 24 hours

26. 1ACBS13-1-13-H-- Energize Bus 13 From 13-1

This is a recovery action during LOOP events when the only source of power for 4KV Bus 13 is from the DG Bus 13-1. The only loads on Bus 13 modeled in the PRA are the containment cooling service water and CRD pumps. The times involved are the same as those addressed in OCRD, OHX, and OSPC discussed above.

27. 1ACBS14-1-14-H-- Energize Bus 14 From 14-1

This is a recovery action during LOOP events when the only source of power for 4KV Bus 14 is from the DG Bus 14-1. The only loads on Bus 14 modeled in the PRA are the containment cooling service water and CRD pumps. The times involved are the same as those addressed in OCRD, OHX, and OSPC discussed above.

28. 1ACBS14-124-1H-- Energize Bus 14-1 from Cross Tie Bus

This is a recovery action during LOOP events when power to bus 14-1 is not available from either the normal sources or the diesel generators. When this situation exists, the operators are directed to energize the bus using the cross-tie to the opposite unit by closing breakers 1421 and 2429. The time available to perform the actions is driven by the need for core spray and RHR. The action takes less than one minute to complete.

29. 1ACBS18-19---H-- Energize Bus 18 From 19

This is a recovery action that cross-ties the 480VAC Buses 18 and 19. The action is only required if Bus 18 cannot be supplied power from the DG Bus 13-1. Bus 18 supplies the following loads: SBLC pump 1A, Div I core spray and LPCI valves, DG1/2 cooling water pump and normal supply for the instrument bus. Time considerations are driven by the need for both SBLC pumps.

30. 1ACBS19------H-- Energize Bus 19 From 18

This is a recovery action that cross-ties the 480VAC Buses 18 and 19. The action is only required if Bus 19 cannot be supplied power from the DG Bus 14-1. Bus

19 supplies the following loads: SBLC pump 1B, Div II Core Spray and LPCI valves, DG1 cooling water pump, and normal supply for the Essential Service Bus. Time considerations are driven by the need for both SBLC pumps.

31. 1ACCB1517----H-- Energize Bus 15 From 17

This is a recovery action that supplies the 480VAC Bus 15 from Bus 17 which is an alternate supply. The action is only required if Bus 15 cannot be supplied power from Bus 14. Bus 15 acts as an alternate supply for the Essential Service Bus. Time considerations are driven by the need for both SBLC pumps.

32. 1ATAD22A-----H-- Master Trip Unit Not Restored After Test/Maint

Actions 32 through 35 are pre-initiators. No time constraints were considered.

33. 1ATAD22B-----H-- Master Trip Unit Not Restored After Test/Maint

34. 1ATAD22C-----H-- Master Trip Unit Not Restored After Test/Maint

35. 1ATAD22D-----H-- Master Trip Unit Not Restored After Test/Maint

36. 1CRPMOPERATORH-- Start CRD Pump B

This action models the operator starting CRD pump B after failure of the running CRD pump A. For this event, the operator must start CRD pump B and open the discharge valve. The timing considerations for this action are the same as for the OCRD node discussed above.

37. 1CSSWMANSTRT-H-- Initiate Core Spray Following Failure of Automatic Initiation

This action is a backup to the automatic start of the core spray system. The action is modeled for cases where the core spray system receives a start signal but the pump does not start or the discharge valve does not open.

38. 1HISY--------H-- Control HPCI in Accordance with QCOP 300-3

This action models the operator action to manually throttle the High Pressure Coolant Injection system. This action prevents the system from continuously tripping and restarting on RPV level. The system will function without the intervention, however the cycling of the system can be reduced if the operator takes control. Usually this action is accomplished shortly after the system initiates (within 2-3 minutes).

39. 1HISYMANSTARTH-- Initiate HPCI Following Failure of Auto Initiation

This action is a backup to the automatic start of the HPCI system. The action is modeled for cases where the HPCI system receives a start signal but the pump does not start or the discharge valve does not open.

40. 1INSYH-5KGPM-H-- Miss QCOP 2300-6 5000 gpm Caution (High Stress)

Actions 40 through 55 are modeled as part of the CCST inventory control function which considers preventing transfer of CCST water to the torus via the HPCI minimum flow valve. Within the inventory control function modeling, time considerations for CCST inventory control are included in the fault tree model of the overall function. Evaluation of time constraints for an individual action within the overall function is not considered meaningful. The operators have several hours to take action to control CCST inventory.

41. 1INSYH-ALTINJH-- Switch to an Alternate Injection Source (High Stress)

42. 1INSYH-COLLDNH-- Decide to Cool Down (High Stress)

43. 1INSYH-LEVEL-H-- Acknowledge +/-.5" Torus Level Alarm (High Stress)

44. 1INSYH-MINFLWH-- Miss QCOP 2300-6 Caution Minimize Min Flow (High Stress)

45. 1INSYH-PRESS-H-- Place HPCI In Pressure Control (High Stress)

46. 1INSYH-TORUS-H-- Switch HPCI Suction to Torus (High Stress)

47. 1INSYH-X-FER-H-- Defeat HPCI Auto Suction X-Transfer (High Stress)

48. 1INSYO-5KGPM-H-- Miss QCOP 2300-6 5000 GPM Caution (Opt Stress)

49. 1INSYO-ALTINJH-- Switch to an Alternate Injection Source (Opt Stress)

50. 1INSYO-COLLDNH-- Decide to Cool Down (Opt Stress)

51. **1INSYO-LEVEL-H--** Acknowledge +/-.5" Torus Level Alarm (Opt Stress)

52. **1INSYO-MINFLWH--** Miss QCOP 2300-6 Caution Minimize Min Flow (Opt Stress)

53. **1INSYO-PRESS-H--** Place HPCI In Pressure Control (Opt Stress)

54. **1INSYO-TORUS-H--** Switch HPCI Suction to Torus (Opt Stress)

55. **1INSYO-X-FER-H--** Defeat HPCI Auto Suction X-Transfer (Opt Stress)

56. **1RISYMANSTARTH--** Initiate RCIC Following Failure of Auto Initiation

This action is a backup to the automatic start of the RCIC system. The action is modeled for cases where the RCIC system receives a start signal but the pump does not start or the discharge valve does not open.

57. **1TBPM1-3801B-H--** Start TBCCW Pump after the Running Pump has Failed (Fault/Loss of Power)

This action is to start the standby turbine building closed cooling water pump after the running pump fails due to random faults or support system failures. The time available to perform this action is driven by the need for the feedwater or CRD systems which require TBCCW for operation.

58. **BFPPD12-4101AH--** Fail to Restore Diesel Fire Pump 1/2-4101A following Test or Maintenance

Actions 58 and 59 are pre-initiators. No time constraints were considered.

59. **BFPPD12-4101BH--** Fail to Restore Diesel Fire Pump 1/2-4101B following Test or Maintenance

60. **BSWPM1/2-3901H--** Start the Standby Service Water Pump

This is the operator action to start the standby service water pump after a loss of the normally running pump. The loss may be caused by either random failures or a single unit LOOP. The time available to perform this action is driven by the need for CRD, FW and Instrument Air.

61. BSWPM2/5-3901-H-- Start Two Service Water Pumps Following a Dual Unit LOOP

This is the operator action to start one service water pump on each unit following a dual unit LOOP. The time available to perform this action is driven by the need for CRD, FW and Instrument Air.

Table 4.4.2-1
SUMMARY of QUAD CITIES FAULT TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW*	Orig. HEP	Rev. HEP	%CHG
1ACBS13-1-13-H--	Manually Transfer Power to Bus 13	1.450	5.3E-3	1.2E-3	-77.2%
1ACBS14-1-14-H--	Manually Transfer Power to Bus 14	4.800	5.3E-3	1.2E-3	-77.2%
1ACBS14-124-1H--	Energize Bus 14-1 from Cross Tie Bus	1.450	1.0E-2	1.0E-2	
1ACBS18-19---H--	Energize Bus 18 from 19		1.0E-2	1.0E-2	
1ACBS19------H--	Energize Bus 19 from 18		1.0E-2	1.0E-2	
1ACCB1516----H--	Energize Bus 15 from 16		5.3E-3	5.3E-3	
1ACCB1517----H--	Energize Bus 15 from 17		5.3E-3	5.3E-3	
1ATAD22A-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.080	4.5E-3	4.5E-3	
1ATAD22B-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.090	4.5E-3	4.5E-3	
1ATAD22C-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.080	4.5E-3	4.5E-3	
1ATAD22D-----H--	Indicator Trip Unit not Restored after Test/Maintenance	1.090	4.5E-3	4.5E-3	
1CRPMOPERATORH--	Start CRD Pump B		4.0E-3	4.0E-3	
1CSSWMANSTRT-H--	Init Core Spray Following Failure of Auto Initiation		7.5E-3	7.5E-3	
1HISY--------H--	Manually Controlling HPCI in Accordance with QCOP 2300-3	6.540	2.4E-3	1.3E-3	-43.8%
1HISYMANSTARTH--	Init HPCI Following Failure of Auto Initiation		1.2E-3	1.2E-3	
1INSYH-5KGPM-H--	Miss QCOP 2300-6 5000gpm Caution (High Stress)	1.000	6.0E-3	6.0E-3	
1INSYH-ALTINJH--	Switch To An Alternate Inj Source (High Stress)	1.030	4.3E-2	4.3E-2	
1INSYH-COLLDNH--	Decide To Cool Down (High Stress)	1.000	3.7E-3	3.7E-3	
1INSYH-LEVEL-H--	Acknowledge +/-.5" Torus Lvl Alarm (High Stress)	1.000	1.3E-3	1.3E-3	
1INSYH-MINFLWH--	Miss QCOP 2300-6 Caution Minimize Min Flow (High Stress)	1.000	6.0E-3	6.0E-3	
1INSYH-PRESS-H--	Place HPCI In Pressure Control (High Stress)	1.010	1.8E-1	1.8E-1	
1INSYH-TORUS-H--	Switch HPCI Suction To Torus (High Stress)	1.020	4.0E-2	4.0E-2	
1INSYH-X-FER-H--	Defeat HPCI Auto Suction X-Fer (High Stress)	1.000	3.1E-2	3.1E-2	
1INSYO-5KGPM-H--	Miss QCOP 2300-6 5000gpm Caution (Opt Stress)		1.2E-3	1.2E-3	
1INSYO-ALTINJH--	Switch To An Alternate Inj Source (Opt Stress)		7.1E-3	7.1E-3	
1INSYO-COLLDNH--	Decide To Cool Down (Opt Stress)		3.7E-3	3.7E-3	
1INSYO-LEVEL-H--	Acknowledge +/-.5" Torus Lvl Alarm (Opt Stress)		2.7E-4	2.7E-4	
1INSYO-MINFLWH--	Miss QCOP 2300-6 Caution Minimize Min Flow (Opt Stress)		1.2E-3	1.2E-3	
1INSYO-PRESS-H--	Place HPCI In Pressure Control (Opt Stress)		3.6E-2	3.6E-2	
1INSYO-TORUS-H--	Switch HPCI Suction To Torus (Opt Stress)		7.9E-3	7.9E-3	
1INSYO-X-FER-H--	Defeat HPCI Auto Suction X-Fer (Opt Stress)		7.9E-3	7.9E-3	

Table 4.4.2-1
SUMMARY of QUAD CITIES FAULT TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW*	Orig. HEP	Rev. HEP	%CHG
1RISYMANSTARTH--	Init RCIC Following Failure of Auto Initiation		1.2E-3	1.2E-3	
1TBPM1-3801B-H--	Start TBCCW Pmp after the Running Pmp has Failed (Fault/Loss of Pwr)		1.4E-3	1.4E-3	
BFPPD12-4101AH--	Fail to Restore Diesel Fire Pmp 1/2-4101A following Test or Maint		5.5E-2	5.5E-2	
BFPPD12-4101BH--	Fail to Restore Diesel Fire Pmp 1/2-4101B following Test or Maint		5.5E-2	5.5E-2	
BSWPM1/2-3901H--	Start the Standby Service Water Pump	1.060	5.3E-3	5.3E-3	
BSWPM2/5-3901-H--	Start Two Service Water Pumps Following a Dual Unit LOOP	1.020	2.7E-2	2.7E-2	

* The RAW value was not determined for those operator actions whose contribution was below the quantification cutoff. These operator actions were deemed to be insignificant contributors and were not evaluated further.

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OAD1-CS01	Init ADS (MLOCA) - [Opt-stress, Zero-depend, No-recov]	24.991	4.9E-3	1.3E-2	165.3%
OAD1-CS02	Init ADS (TRAN, SBO, LOOP, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.742	5.4E-4	1.3E-3	140.8%
OAD1-CS04	Init ADS (TRAN, SBO, ATWS) - [Opt-stress, Low-depend, Yes-recov]	1.000	6.0E-3	5.1E-2	753.9%
OAD1-CS09	Init ADS (TRAN, MLOCA, SLOCA, LOOP, ATWS) - [Mod-stress, Zero-depend, No-recov]	5.996	9.8E-3	1.6E-2	63.3%
OAD1-CS10	Init ADS (SLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.000	1.1E-3	5.2E-2	4583.7%
OAD1-CS11	Init ADS (TRAN, MLOCA, LOOP, ATWS) - [Mod-stress, Zero-depend, No-recov]	2.602	9.8E-3	6.5E-2	565.3%
OAD1-CS12	Init ADS (SLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.000	6.5E-3	5.2E-2	692.6%
OAD1-CS17	Init ADS (TRAN, MLOCA, SLOCA, SBO, LOOP, ATWS) - [High-stress, Zero-depend, No-recov]	1.083	2.5E-2	1.6E-2	-36.0%
OAD1-CS18	Init ADS (SLOCA, LOOP) - [High-stress, Zero-depend, Yes-recov]	4.738	2.7E-3	1.6E-3	-40.7%
OAD1-CS19	Init ADS (TRAN, SLOCA, SBO, LOOP, ATWS) - [High-stress, Low-depend, No-recov]	1.018	7.4E-2	6.5E-2	-11.9%
OAD1-CS20	Init ADS (LOOP) - [High-stress, Zero-depend, Yes-recov]	1.265	2.7E-3	5.2E-2	1808.2%
OAD1-CS21	Init ADS (SLOCA) - [High-stress, Low-depend, No-recov]	1.008	7.4E-2	1.6E-1	111.6%
OAD1-SUM	Init ADS	36.443			
OAD2-CS01	Depress per the HCL Curve - [Op-stress, Zero-depend, No-recov]	1.000	3.2E-2	1.3E-2	-59.4%
OAD2-CS02	Depress per the HCL Curve (TRAN, SLOCA, ISLOCA, SBO, LOOP, ATWS) - [Op-stress, Zero-depend, Yes-recov]	1.383	3.5E-3	1.3E-3	-62.9%
OAD2-CS04	Depress per the HCL Curve (SLOCA) - [Opt-stress, Low-depend, Yes-recov]	1.000	8.8E-3	5.1E-2	482.2%
OAD2-CS08	Depress per the HCL Curve (SBO) - [Opt-stress, High-depend, Yes-recov]		1.8E-2	5.0E-1	2681.4%
OAD2-CS09	Depress per the HCL Curve (ATWS) - [Mod-stress, Zero-depend, No-recov]	1.005	6.4E-2	1.3E-2	-79.7%
OAD2-CS10	Depress per the HCL Curve (TRAN, SLOCA, SBO, LOOP, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	7.137	7.0E-3	1.3E-3	-81.4%
OAD2-CS11	Depress per the HCL Curve (ATWS) - [Mod-stress, Low-depend, No-recov]	1.013	1.1E-1	6.2E-2	-43.3%
OAD2-CS12	Depress per the HCL Curve (LOOP) - [Mod-stress, Low-depend, Yes-recov]	1.000	1.2E-2	5.1E-2	327.0%
OAD2-CS17	Depress per the HCL Curve (ISLOCA) - [High-stress, Zero-depend, No-recov]	1.000	1.6E-1	1.3E-2	-91.9%
OAD2-CS18	Depress per the HCL Curve (SBO) - [High-stress, Zero-depend, Yes-recov]	1.000	1.8E-2	1.3E-3	-92.8%
OAD2-CS19	Depress per the HCL Curve (ISLOCA) - [High-stress, Low-depend, No-recov]	1.001	2.0E-1	6.2E-2	-68.8%

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OAD2-SUM	Depress per the HCL Curve	7.539			
OAL-CS01	Lower Water Level - [Opt-stress, Zero-depend, No-recov]	1.000	4.7E-2	4.7E-2	
OAL-CS09	Lower Water Level (ATWS) - [Mod-stress, Zero-depend, No-recov]	1.007	9.4E-2	9.4E-2	
OAL-CS11	Lower Water Level (ATWS) - [Mod-stress, Low-depend, No-recov]	1.014	1.4E-1	1.4E-1	
OAL-SUM	Lower Water Level	1.021			
OAT-CS01	Manually Init ARI - [Opt-stress, Zero-depend, No-recov]	1.000	1.2E-3	1.2E-3	
OAT-CS17	Manually Init ARI (ATWS) - [High-stress, Zero-depend, No-recov]	1.000	6.0E-3	6.0E-3	
OAT-CS21	Manually Init ARI (ATWS) - [High-stress, Mod-depend, No-recov]	1.000	1.5E-1	1.5E-1	
OAT-SUM	Manually Init ARI	1.000			
OCD-CS01	Cooldown With the Main Condenser - [Opt-stress, Zero-depend, No-recov]	1.000	5.0E-3	5.0E-3	
OCD-CS02	Cooldown With the Main Condenser (SLOCA) - [Opt-stress, Zero-depend, Yes-recov]	1.013	5.5E-4	5.5E-4	
OCD-CS04	Cooldown With the Main Condenser (SLOCA) - [Opt-stress, Low-depend, Yes-recov]	1.032	6.0E-3	6.0E-3	
OCD-SUM	Cooldown With the Main Condenser	1.045			
OCNTS-CS01	Init Containment Sprays (MLOCA, SLOCA) - [Opt-stress, Zero-depend, No-recov]	1.232	1.2E-2	1.2E-2	
OCNTS-CS03	Init Containment Sprays (MLOCA, SLOCA) - [Opt-stress, Low-depend, No-recov]	1.000	6.1E-2	6.1E-2	
OCNTS-CS09	Init Containment Sprays (LLOCA, SLOCA) - [Mod-stress, Zero-depend, No-recov]	1.006	2.5E-2	2.5E-2	
OCNTS-CS10	Init Containment Sprays (TRAN, LLOCA, SBO, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.003	2.7E-3	2.7E-3	
OCNTS-CS11	Init Containment Sprays (SLOCA) - [Mod-stress, Low-depend, No-recov]	1.011	7.4E-2	7.4E-2	
OCNTS-CS13	Init Containment Sprays (LLOCA) - [Mod-stress, Mod-depend, No-recov]	1.000	1.6E-1	1.6E-1	
OCNTS-CS17	Init Containment Sprays - [High-stress, Zero-depend, No-recov]	1.000	6.2E-2	6.2E-2	
OCNTS-CS18	Init Containment Sprays (TRAN, SBO, LOOP, ATWS, IORV) - [High-stress, Zero-depend, Yes-recov]	1.000	6.8E-3	6.8E-3	
OCNTS-CS20	Init Containment Sprays (TRAN, SBO, ATWS) - [High-stress, Low-depend, Yes-recov]	1.000	1.2E-2	1.2E-2	
OCNTS-CS21	Init Containment Sprays - [High-stress, Mod-depend, No-recov]	1.000	2.0E-1	2.0E-1	

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OCNTS-CS22	Init Containment Sprays (LLOCA) - [High-stress, Mod-depend, Yes-recov]	1.000	2.2E-2	2.2E-2	
OCNTS-SUM	Init Containment Sprays	1.252			
OCRD-CS01	Restore CRD Injection (SBO) - [Opt-stress, Zero-depend, No-recov]	1.000	6.2E-3	6.2E-3	
OCRD-CS02	Restore CRD Injection (SBO, LOOP) - [Opt-stress, Zero-depend, Yes-recov]	1.000	6.4E-4	6.4E-4	
OCRD-CS04	Restore CRD Injection - [Opt-stress, Low-depend, Yes-recov]	1.000	6.1E-3	6.1E-3	
OCRD-CS10	Restore CRD Injection (SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	1.000	1.3E-3	1.3E-3	
OCRD-CS12	Restore CRD Injection - [Mod-stress, Low-depend, Yes-recov]	1.000	6.8E-3	6.8E-3	
OCRD-CS17	Restore CRD Injection (SBO) - [High-stress, Zero-depend, No-recov]	1.000	3.1E-2	3.1E-2	
OCRD-CS18	Restore CRD Injection (LOOP) - [High-stress, Zero-depend, Yes-recov]	1.000	3.4E-3	3.4E-3	
OCRD-SUM	Restore CRD Injection	1.000			
OCST-CS01	Align Low Pressure Pump Suction to the CCST (IORV) - [Opt-stress, Zero-depend, No-recov]	1.000	8.4E-2	1.4E-2	-83.1%
OCST-CS02	Align Low Pressure Pump Suction to the CCST (IORV) - [Opt-stress, Zero-depend, Yes-recov]	1.002	9.2E-3	1.4E-3	-84.6%
OCST-CS09	Align Low-Press Pump Suct to the CCST - [Mod-stress, Zero-depend, No-recov]	New	New	1.5E-2	
OCST-CS10	Align Low-Press Pump Suct to the CCST (TRAN, IORV, LOOP, SBO, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.415	1.9E-2	1.5E-3	-91.9%
OCST-CS12	Align Low Pressure Pump Suction to the CCST (SBO, LOOP) - [Mod-stress, Low-depend, Yes-recov]	6.174	2.3E-2	5.1E-2	123.8%
OCST-CS18	Align Low Pressure Pump Suction to the CCST (TRAN, IORV, ATWS) - [High-stress, Zero-depend, Yes-recov]	1.018	4.6E-2	1.9E-3	-95.9%
OCST-CS20	Align Low-Press Pump Suct to the CCST (TRAN, SBO, LOOP, ATWS) - [High-stress, Low-depend, Yes-recov]	1.017	4.9E-2	5.2E-2	5.7%
OCST-CS25	Align Low Press Pmp Suct to the CST	New	New	1.0E+0	
OCST-SUM	Align Low Pressure Pump Suction to the CCST	6.627			
OFW1-CS00	Restart Main Feedwater Pumps or Recover Hotwell Level with Manual Makeup Valves (LOIA) [Opt-stress, Zero-depend, Yes-recover]	New	New	8.4E-3	
OFW1-CS01	Restart Main Feedwater Pumps (TRAN, SLOCA, IORV) - [Opt-stress, Zero-depend, No-recov]	4.988	5.0E-3	1.4E-2	172.4%
OFW1-CS02	Restart Main Feedwater Pumps (TRAN, SLOCA, IORV) - [Opt-stress, Zero-depend, Yes-recov]	New	New	1.4E-3	

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OFW1-SUM	Restart Main Feedwater Pumps	4.988			
OFW2-CS01	Manually Scram and cntrl Feedwater - [Opt-stress, Zero-depend, No-recov]	1.000	8.8E-3	8.8E-3	
OFW2-CS09	Manually Scram and cntrl Feedwater (ISLOCA) - [Mod-stress, Zero-depend, No-recov]	1.019	1.7E-2	1.7E-2	
OFW2-SUM	Manually Scram and cntrl Feedwater	1.019			
OFW3-CS01	Restore Feedwater Injection - [Opt-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OFW3-CS17	Restore Feedwater Injection (SBO, LOOP) - [High-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OFW3-SUM	Restore Feedwater Injection	1.000			
OHX-CS01	Align Cooling to RHR (ATWS) - [Opt-stress, Zero-depend, No-recov]	1.000	8.2E-3	1.0E-4	-98.8%
OHX-CS02	Align Cooling to RHR (TRAN, MLOCA, SLOCA, IORV, LLOCA) - [Opt-stress, Zero-depend, Yes-recov]	5.174	9.0E-4	1.0E-5	-98.9%
OHX-CS06	Align Cooling to RHR (IORV) - [Opt-stress, Mod-depend, Yes-recov]	1.001	1.6E-2	1.6E-2	
OHX-CS09	Align Cooling to RHR (ATWS) - [Mod-stress, Zero-depend, No-recov]	2.386	1.7E-2	1.7E-2	
OHX-CS10	Align Cooling to RHR (MLOCA, SLOCA, IORV, SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	54.048	1.9E-3	1.9E-3	
OHX-CS14	Align Cooling to RHR (SLOCA, IORV) - [Mod-stress, Mod-depend, Yes-recov]	1.000	1.7E-2	1.7E-2	
OHX-CS17	Align Cooling to RHR (SBO) - [High-stress, Zero-depend, No-recov]	59.647	4.5E-2	4.5E-2	
OHX-CS18	Align Cooling to RHR (SLOCA, LLOCA) - [High-stress, Zero-depend, Yes-recov]	1.007	4.9E-3	4.9E-3	
OHX-CS22	Align Cooling to RHR (SLOCA) - [High-stress, Mod-depend, Yes-recov]	1.001	2.0E-2	2.0E-2	
OHX-SUM	Align Cooling to RHR	118.264			
OIADS-CS01	Inhibit ADS (MLOCA) - [Opt-stress, Zero-depend, No-recov]	3.770	1.7E-4	1.3E-2	7547.1%
OIADS-CS02	Inhibit ADS (MLOCA) - [Opt-stress, Zero-depend, Yes-recov]	New	New	3.3E-3	
OIADS-CS09	Inhibit ADS (SLOCA) - [Mod-stress, Zero-depend, No-recov]	1.000	3.8E-4	1.3E-2	3321.1%
OIADS-CS10	Inhibit ADS (SLOCA) - [Mod-stress, Zero-depend, Yes-recov]	New	New	3.3E-3	
OIADS-CS26	Inhibit ADS (ATWS) - [Mod-stress, Zero-depend, No-recov]	New	New	3.0E-3	
OIADS-SUM	Inhibit ADS	3.770			
OIB-CS01	Isolate the Break - [Opt-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS09	Isolate the Break (ISLOCA) - [Mod-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS10	Isolate the Break (ISLOCA) - [Mod-stress, Zero-depend, Yes-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS13	Isolate the Break (ISLOCA) - [Mod-stress, Mod-depend, No-recov]	1.000	1.0E+0	1.0E+0	

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OIB-CS14	Isolate the Break (ISLOCA) - [Mod-stress, Mod-depend, Yes-recov]	1.000	1.0E+0	1.0E+0	
OIB-CS17	Isolate the Break (ISLOCA) - [High-stress, Zero-depend, No-recov]	1.000	1.0E+0	1.0E+0	
OIB-SUM	Isolate the Break	1.000			
ORP-CS01	Trip Recirculation Pumps - [Opt-stress, Zero-depend, No-recov]	1.000	1.0E-2	1.0E-2	
ORP-CS17	Trip Recirculation Pumps (ATWS) - [High-stress, Zero-depend, No-recov]	1.007	5.1E-2	5.1E-2	
ORP-SUM	Trip Recirculation Pumps	1.007			
OSBCS-CS01	Init Standby Coolant Supply - [Opt-stress, Zero-depend, No-recov]	1.000	1.4E-2	1.4E-2	
OSBCS-CS02	Init Standby Coolant Supply (ISLOCA) - [Opt-stress, Zero-depend, Yes-recov]	1.000	1.5E-3	1.5E-3	
OSBCS-CS09	Init Standby Coolant Supply (MLOCA) - [Mod-stress, Zero-depend, No-recov]	1.014	2.8E-2	2.8E-2	
OSBCS-CS10	Init Standby Coolant Supply (LLOCA, MLOCA) - [Mod-stress, Zero-depend, Yes-recov]	1.143	3.1E-3	3.1E-3	
OSBCS-CS12	Init Standby Coolant Supply (LLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.200	8.4E-3	8.4E-3	
OSBCS-CS17	Init Standby Coolant Supply (LLOCA, MLOCA) - [High-stress, Zero-depend, No-recov]	1.002	7.2E-2	7.2E-2	
OSBCS-CS19	Init Standby Coolant Supply (LLOCA, MLOCA, ISLOCA) - [High-stress, Low-depend, No-recov]	1.001	1.2E-1	1.2E-1	
OSBCS-SUM	Init Standby Coolant Supply	1.360			
OSDC-CS01	Init Shutdown Cooling - [Opt-stress, Zero-depend, No-recov]	1.000	8.4E-2	8.4E-2	
OSDC-CS02	Init Shutdown Cooling (TRANS, SBO, LOOP, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.002	9.2E-3	9.2E-3	
OSDC-SUM	Init Shutdown Cooling	1.002			
OSL1-CS01	Init one SLC pump (ATWS) - [Opt-stress, Zero-depend, No-recov]	1.501	8.7E-3	8.7E-3	
OSL1-CS17	Init one SLC pump (ATWS) - [High-stress, Zero-depend, No-recov]	1.739	4.5E-2	4.5E-2	
OSL1-SUM	Init one SLC pump	2.240			
OSL2-CS01	Init two SLC pump - [Opt-stress, Zero-depend, No-recov]	1.000	8.5E-3	8.5E-3	
OSL2-CS16	Init two SLC pump (ATWS) - [Mod-stress, High-depend, Yes-recov]	1.645	5.6E-2	5.6E-2	
OSL2-SUM	Init two SLC pump	1.645			
OSMP1-CS01	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Opt-stress, Zero-depend, No-recov]	1.001	1.1E-2	1.4E-2	29.1%
OSMP1-CS02	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.009	1.2E-3	1.4E-3	18.3%

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OSMP1-CS03	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Opt-stress, Low-depend, No-recov]	1.000	6.0E-2	6.3E-2	5.8%
OSMP1-CS04	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, LOOP, ATWS) - [Opt-stress, Low-depend, Yes-recov]	1.000	6.6E-3	5.1E-2	678.0%
OSMP1-CS09	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Mod-stress, Zero-depend, No-recov]	1.001	2.3E-2	1.5E-2	-33.0%
OSMP1-CS10	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	6.292	2.5E-3	1.5E-3	-38.4%
OSMP1-CS11	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (IORV) - [Mod-stress, Low-depend, No-recov]	1.000	7.2E-2	6.5E-2	-10.2%
OSMP1-CS12	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [Mod-stress, Low-depend, Yes-recov]	1.837	7.9E-3	5.1E-2	551.4%
OSMP1-CS17	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (SBO with No HPCI or RCIC success) - [High-stress, Zero-depend, No-recov]	New	New	1.9E-2	
OSMP1-CS18	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (TRAN, SBO, LOOP, ATWS) - [High-stress, Zero-depend, Yes-recov]	1.176	7.3E-3	1.9E-3	-74.0%
OSMP1-CS19	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal (SBO) - [High-stress, Low-depend, No-recov]	1.000	1.1E-1	6.8E-2	-38.1%
OSMP1-CS20	Init SSMP with Suct Aligned to the CCST - No HPCI Inj Sig (TRAN, SBO, LOOP, ATWS) - [High-stress, Low-depend, Yes-recov]	1.062	1.2E-2	5.2E-2	331.7%
OSMP1-SUM	Init SSMP with Suct Aligned to the CCST - No HPCI Inject Signal	7.378			
OSMP2-CS01	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal - [Opt-stress, Zero-depend, No-recov]	1.000	9.0E-3	9.0E-3	
OSMP2-CS10	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (TRAN, SBO, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.000	2.0E-3	2.0E-3	
OSMP2-CS12	Init SSMP with Suct Aligned to the FP - No HPCI Inj Sig (TRAN, SBO, ATWS) - [Mod-stress, Low-depend, Yes-recov]	1.000	7.4E-3	7.4E-3	
OSMP2-CS17	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (SBO) - [High-stress, Zero-depend, No-recov]	1.000	6.3E-2	6.3E-2	
OSMP2-CS18	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (LOOP) - [High-stress, Zero-depend, Yes-recov]	1.023	6.9E-3	6.9E-3	

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OSMP2-CS20	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal (LOOP) - [High-stress, Low-depend, Yes-recov]	1.005	1.2E-2	1.2E-2	
OSMP2-SUM	Init SSMP with Suct Aligned to the FP - No HPCI Inject Signal	1.028			
OSMP3-CS01	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal - [Opt-stress, Zero-depend, No-recov]	1.000	7.8E-3	7.8E-3	
OSMP3-CS03	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (ATWS) - [Opt-stress, Low-depend, No-recov]	1.299	5.6E-2	5.6E-2	
OSMP3-CS09	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (TRAN, SLOCA, LOOP, ATWS) - [Mod-stress, Zero-depend, No-recov]	2.083	1.4E-2	1.4E-2	
OSMP3-CS10	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (SLOCA) - [Mod-stress, Zero-depend, Yes-recov]	1.000	1.5E-3	1.5E-3	
OSMP3-CS11	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA, ATWS) - [Mod-stress, Low-depend, No-recov]	1.004	6.3E-2	6.3E-2	
OSMP3-CS12	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA) - [Mod-stress, Low-depend, Yes-recov]	1.000	7.0E-3	7.0E-3	
OSMP3-CS13	Init SSMP with Suct Aligned to the CCST or with HPCI Injt Sig (TRAN, ATWS) - [Mod-stress, Mod-depend, No-recov]	1.000	1.5E-1	1.5E-1	
OSMP3-CS17	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (TRAN, SLOCA, SBO, LOOP, ATWS) - [Hi-stress, Zero-depend, No-recov]	1.061	4.8E-3	4.8E-3	
OSMP3-CS18	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal (SLOCA) - [High-stress, Zero-depend, Yes-recov]	1.000	5.3E-3	5.3E-3	
OSMP3-CS19	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA, LOOP, ATWS) - [High-stress, Low-depend, No-recov]	1.015	9.6E-2	9.6E-2	
OSMP3-CS20	Init SSMP with Suct Aligned to the CCST or with HPCI Inj Sig (SLOCA) - [High-stress, Low-depend, Yes-recov]	1.000	1.1E-2	1.1E-2	
OSMP3-SUM	Init SSMP with Suct Aligned to the CCST or with HPCI Inject Signal	2.462			
OSPC-CS01	Init Supp Pool Cooling (ATWS) - [Opt-stress, Zero-depend, No-recov]	1.000	4.7E-4	1.0E-4	-78.7%
OSPC-CS02	Init Supp Pool Cooling (TRAN, MLOCA, SLOCA, IORV, LLOCA) - [Opt-stress, Zero-depend, Yes-recov]	1.008	5.2E-5	1.0E-5	-80.8%
OSPC-CS04	Init Supp Pool Cooling (MLOCA) - [Opt-stress, Low-depend, Yes-recov]	1.000	5.5E-3	5.5E-3	
OSPC-CS09	Init Supp Pool Cooling (ATWS) - [Mod-stress, Zero-depend, No-recov]	2.155	1.3E-3	1.3E-3	

Table 4.4.2-2
SUMMARY of QUAD CITIES PLANT RESPONSE TREE OPERATOR ACTIONS

NAME	DESCRIPTION	RAW	Orig. HEP	Rev. HEP	%CHG
OSPC-CS10	Init Supp Pool Cooling (SLOCA, IORV, SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	4.817	1.4E-4	1.4E-4	
OSPC-CS17	Init Supp Pool Cooling (SBO) - [High-stress, Zero-depend, No-recov]	1.000	6.3E-3	6.3E-3	
OSPC-CS18	Init Supp Pool Cooling (SLOCA) - [High-stress, Zero-depend, Yes-recov]	1.000	6.9E-4	6.9E-4	
OSPC-SUM	Init Supp Pool Cooling	5.979			
OSS-CS01	Rest Supp Systems (SBO) - [Opt-stress, Zero-depend, No-recov]	1.000	4.0E-2	4.0E-2	
OSS-CS02	Rest Supp Systems (SBO, LOOP) - [Opt-stress, Zero-depend, Yes-recov]	1.000	4.4E-3	4.4E-3	
OSS-CS04	Rest Supp Systems (SBO, LOOP) - [Opt-stress, Low-depend, Yes-recov]	1.000	9.7E-3	9.7E-3	
OSS-CS09	Rest Supp Systems (SBO) - [Mod-stress, Zero-depend, No-recov]	1.000	8.0E-2	8.0E-2	
OSS-CS10	Rest Supp Systems (SBO, LOOP) - [Mod-stress, Zero-depend, Yes-recov]	1.000	8.8E-3	8.8E-3	
OSS-CS11	Rest Supp Systems (SBO) - [Mod-stress, Low-depend, No-recov]	1.000	1.3E-1	1.3E-1	
OSS-CS12	Rest Supp Systems (SBO, LOOP) - [Mod-stress, Low-depend, Yes-recov]	1.000	1.4E-2	1.4E-2	
OSS-CS17	Rest Supp Systems (SBO) - [High-stress, Zero-depend, No-recov]	1.000	2.0E-1	2.0E-1	
OSS-CS18	Rest Supp Systems (SBO, LOOP) - [High-stress, Zero-depend, Yes-recov]	1.000	2.2E-2	2.2E-2	
OSS-CS19	Rest Supp Systems (SBO) - [High-stress, Low-depend, No-recov]	1.000	2.4E-1	2.4E-1	
OSS-CS20	Rest Supp Systems (SBO, LOOP) - [High-stress, Low-depend, Yes-recov]	1.000	2.6E-2	2.6E-2	
OSS-SUM	Rest Supp Systems	1.000			
OVNT-CS01	Init Cont Vent - [Opt-stress, Zero-depend, No-recov]	1.000	9.2E-2	9.2E-2	
OVNT-CS02	Init Cont Vent (TRAN, SBO, ATWS) - [Opt-stress, Zero-depend, Yes-recov]	1.000	1.0E-2	1.0E-2	
OVNT-CS10	Init Cont Vent (TRAN, LLOCA, SBO, LOOP, ATWS) - [Mod-stress, Zero-depend, Yes-recov]	1.009	2.0E-2	2.0E-2	
OVNT-CS18	Init Cont Vent (TRAN, LLOCA, MLOCA, SLOCA, IORV, SBO, LOOP, ATWS) - [High-stress, Zero-depend, Yes-recov]	1.238	5.1E-2	5.1E-2	
OVNT-SUM	Init Cont Vent	1.247			

(Note: after each set of operator action cases there is a summary line that gives the combined Risk Achievement Worth for all the cases.)

4.4.3 Common Cause Analysis

"Common cause" describes multiple failures of functionally identical components due to a single, shared cause. Common cause analysis (CCA) evaluates the effects of these dependencies that may affect the ability of a system to prevent or mitigate a severe accident.

The Quad Cities CCA modeled common cause failures at the basic event level, employing the Multiple Greek Letter (MGL) method as defined in NUREG/CR-4780, "Procedures for Treating Common Cause Failure in Safety and Reliability Studies." The MGL method uses the parameters beta (β), gamma (γ), and delta (δ), defined as follows:

β conditional probability that the common cause of a component failure will be shared by one or more additional components

γ conditional probability that a common cause failure of two components will be shared by one or more additional components

δ conditional probability that a common cause failure of three components will be shared by one or more additional components

The MGL method allows one to continue defining parameters as far as desired. Consideration of more than three parameters generally reduces conservatism but can become unwieldy. The Quad Cities CCA assumed that common cause groups with greater than four components were adequately represented by the three-parameter analysis without being overly conservative.

The evaluation of Quad Cities failure data indicated that there had been no common cause events at the Quad Cities site applicable to current maintenance and operating practices. As a result, to more realistically model current experience at Quad Cities, a Quad Cities-specific evaluation of common cause failure events was performed. Quad Cities-specific common cause parameters were developed for components that had data available, including the following:

- Circuit Breakers
- Check Valves
- Diesel Generators
- Motor-operated Valves
- Fans
- Low-head Pumps
- HVAC Chillers
- Relief Valves
- Pumps

A generic common cause failure database was developed from EPRI NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," supplemented with events from the September 1990 EPRI draft report, "A Database of Common Cause Events for Risk and Reliability Evaluations."

An expert judgement panel reviewed data from the generic common cause failure database for applicability to CECo plants. IPEP and CECo each provided members to the panel representing the Quad Cities and Byron IPEs. The members from IPEP were common cause data specialists and the members from CECo were plant experts. CECo plant experts had previously worked as operators and were familiar with current plant practices and procedures as well as those in practice at the time of the events in the generic common cause failure database.

The common cause failure database was reviewed for events applicable to Quad Cities. The expert panel came to a consensus opinion on each generic common cause event's applicability to Quad Cities, based upon current Quad Cities system configuration, and maintenance and operating practices. Events involving known common cause mechanisms addressed by specific programs in place at Quad Cities were discarded from the database as were common cause events that occurred due to specific system configurations not present at Quad Cities. Events involving common cause mechanisms that have been addressed in general by maintenance or operating practices at Quad Cities were assigned a lesser probability of occurrence based on judgement of the panel. Furthermore, as part of the Modified IPE, a "floor" of 0.01 was imposed on beta values. Table 4.4.3-1 shows the Quad Cities-specific MGL parameters resulting from this analysis.

An average common cause component group was quantified from a composite of all the common cause failures for all components in the database (ALL in Table 4.4.3-1). Use of the parameters calculated for this average common cause group was extended to components that have no history of common cause failure, but were judged by the analyst to have some potential for common cause failure. The common cause contribution for the following components was calculated using the average MGL values:

- Air compressors
- Batteries and DC power supplies
- Dampers
- Explosive, manual, and solenoid valves
- Heat exchangers
- Manual pushbutton switches
- Relays, including contacts and coils
- Strainers and filters
- Switches, including temperature, level, and pressure switches
- Temperature and pressure transmitters and controllers
- Signal comparators
- Transformers
- Buses
- Electrical filters
- Inverters

In general, the components included in this list were judged to be less complex than the components in the database and thought to have less potential for common cause failure mechanisms. Therefore, assignment of the average common cause parameters is judged to be realistic.

The following general equation expresses the probability (Q_k) of a subset of multiple component failures of size k within a common cause group of size m due to common cause in terms of the MGL parameters defined earlier:

TABLE 4.4.3-1
MGL PARAMETERS FOR QUAD CITIES

COMMON CAUSE COMPONENT GROUP	FOUR-COMPONENT SYSTEM			THREE-COMPONENT SYSTEM		TWO-COMPONENT SYSTEM
	β	γ	δ	β	γ	β
Circuit Breakers	7.2E-2	9.9E-1	8.6E-1	5.6E-2	9.3E-1	3.9E-2
Check Valves	4.8E-2	6.8E-1	3.7E-1	4.3E-2	4.4E-1	3.1E-2
Diesel Generators	(a)	(a)	(a)	1.0E-2	(b)	1.0E-2
Motor-operated Valves	1.9E-2	5.4E-1	5.2E-1	1.6E-2	4.3E-1	1.1E-2
Fans	5.8E-2	(b)	7.8E-1	5.8E-2	8.5E-1	5.4E-2
Low-head Pumps	(a)	(a)	(a)	(a)	(a)	8.1E-2
HVAC Chillers	(a)	(a)	(a)	2.4E-2	1.3E-1	1.3E-2
Safety/Relief Valves	3.2E-1	5.4E-1	(b)	2.7E-1	6.4E-1	2.2E-1
Pumps	(a)	(a)	(a)	1.3E-2	3.5E-1	1.0E-2
ALL	2.1E-2	6.8E-1	6.4E-1	1.9E-2	5.8E-1	1.5E-2

Notes:

a. MGL values not calculated for this size system.

b. Value not calculated. The average value for all component failures ("ALL") given in this table is used as the value for these components.

$$Q_{k/m} = \frac{1}{\binom{m-1}{k-1}} \left(\prod_{i=1}^{k} \rho_i \right) (1 - \rho_{k+1})\ Q_t$$

where

$$\begin{aligned} \rho_1 &= 1 \\ \rho_2 &= \beta \\ \rho_3 &= \gamma \\ \rho_4 &= \delta \\ &\vdots \\ \rho_{m+1} &= 0 \end{aligned}$$

and where Q_t represents the random portion of the failure rate of the component for the given failure mode. Again, the Quad Cities CCA used only three MGL parameters.

The denominator of the fraction in the equation for $Q_{k/m}$ contains the binomial expansion coefficient:

$$\binom{m-1}{k-1} = \frac{(m-1)!}{(k-1)!\,(m-k)!}$$

This represents the number of ways m-1 components can be grouped into sets of size k-1. The fault tree should include this number of similar common cause events. The coefficient assures that each common cause event gets a fraction of the common cause probability for a system of size m and order k such that the sum of the common cause events equals the total common cause contribution.

We can represent the above equation for $Q_{k/m}$ in the following manner:

$$Q_{k/m} = (\text{MGL Factor})_{k/m}\ Q_t$$

Strict adherence to the above methodology requires that fault trees include all common cause event combinations, including those that exceed the failure criterion for the given set of components. For example, a fault tree representing a system of four pumps in which only two must succeed has to include an event representing common cause failure of all four pumps. The reason for this lies in the $(1-\rho_{k+1})$ factor in the equation for $Q_{k/m}$. In essence, this factor eliminates the contribution of higher-order common cause events (e.g., a second-order common cause failure for a set of three check valves represents failure of two *and only two* check valves). Because the factor eliminates higher-order common cause contributions, higher-order terms must be explicitly included in the fault trees to properly account for all common cause events that cause system failure.

This presents a difficulty, especially for the process of modeling systems in various degraded support state conditions. In general, system models for cases in which one or more trains are unavailable due to degraded support systems were created by trimming out the unavailable trains from the more general all-support-systems-available fault trees, thus eliminating some non-minimal common cause events.

Rather than adding back all common cause events that were trimmed in each degraded support state model, the Quad Cities CCA eliminated the (1-ρ) factor from the MGL factor equations. This yields a slightly higher (i.e., conservative) unavailability because with this alteration, a given common cause event for failure of k out of m components actually represents the failure of *at least* k out of m components. Generally, this conservatism affects fault trees that model systems with all supports available more than it affects those modeling degraded support states. An additional benefit of this innovation is that common cause failures need only be modeled up to the failure criteria for the component set. Non-minimal common cause events are represented in each lower-order common cause event.

Table 4.4.3-2 shows the equations used in the Quad Cities CCA process to generate the MGL factors, and Table 4.4.3-3 shows the component-specific MGL factors derived from Tables 4.4.3-1 and 4.4.3-2 for the Quad Cities IPE. The first column of Table 4.4.3-3 shows both the Quad Cities-specific component code and the applicable component category from Table 4.4.3-1. To duplicate a common cause event probability, multiply the failure rate of the random event on which the common cause event is based by the MGL factor shown in Table 4.4.3-3.

TABLE 4.4.3-2
EQUATIONS FOR GENERATING MGL FACTORS FROM β, γ, δ

COMPONENTS IN SYSTEM (m)	NUMBER OF FAILED COMPONENTS (k)		
	2	3	4
2	β	-	-
3	$\frac{1}{2}\times\beta$	$\beta\times\gamma$	-
4	$1/3\times\beta$	$1/3\times\beta\times\gamma$	$\beta\times\gamma\times\delta$

TABLE 4.4.3-3
MGL FACTORS FOR QUAD CITIES

COMPONENT TYPE	COMPONENTS IN SYSTEM (m)	NUMBER OF FAILED COMPONENTS (k)		
		2	3	4
Circuit Breakers (CB)	2	3.90E-2	-	-
	3	2.80E-2	5.21E-2	-
	4	2.40E-2	2.38E-2	6.13E-2
Check Valves (CV)	2	3.10E-2	-	-
	3	2.15E-2	1.89E-2	-
	4	1.60E-2	1.09E-2	1.21E-2
Diesel Generators (DG)	2	1.00E-2	-	-
	3	5.00E-3	5.80E-3	-
Motor-operated Valves (MV)	2	1.10E-2	-	-
	3	8.00E-3	6.88E-3	-
	4	6.33E-3	3.42E-3	5.34E-3
Fans (FN)	2	5.40E-2	-	-
	3	2.90E-2	4.93E-2	-
	4	1.93E-2	1.31E-2	3.08E-2
Low-head Pumps (PM)	[1] 2	8.10E-2	-	-
HVAC Chillers (RF)	2	1.30E-2	-	-
	3	1.20E-2	3.12E-3	-
Relief Valves (AM, AS, AV)	2	2.20E-1	-	-
	3	1.35E-1	1.73E-1	-
	[2] 4	1.07E-1	5.76E-2	1.11E-1
Pumps (PD, PM, PT)	[3] 2	1.00E-2	-	-
	[4] 3	6.50E-3	4.55E-3	-
ALL	2	1.50E-2	-	-
	3	9.50E-3	1.10E-2	-
	[5] 4	7.00E-3	4.76E-3	9.14E-3

Notes:
[1] Applied to Core Spray Pumps
[2] Applied to Electromatic Relief Valves (Main Steam Lines)
[3] Applied to Standby Liquid Control, Control Rod Drive, and Diesel Fire Pumps
[4] Applied to Diesel Generator Cooling Water and Feedwater Pumps
[5] Applied to Residual Heat Removal (RHR), RHR Service Water, Condensate, and non-safety related Service Water Pumps

4.4.4 Internal Flooding Analysis

The internal flooding analysis was performed to identify potential sources of flooding and spraying internal to Quad Cities, and the event sequences associated with these sources that could potentially lead to core damage. Pipe, tank, and valve ruptures, etc., could lead to flooding and/or spraying of plant equipment, resulting in failures that could trip the reactor and impair the operation of equipment needed to safely shutdown the plant. The impact of the potential flooding/spraying was assessed to assure that all potential core damage sequences of high probability would be identified.

4.4.4.1 Information Collection

Much information needed for the analysis was taken from the Safe Shutdown Report (SSR) prepared in response to the requirements of 10CFR Part 50, Appendix R. The fire zones developed for the SSR were generally found to be acceptable for use as flooding zones. The list of equipment necessary for safe shutdown developed for the SSR was used for the internal flooding analysis as well.

Additional information necessary to the analysis was collected during plant walkdowns. This included investigation of the potential flooding and spraying sources, the equipment that would be affected by these sources, the potential for flooding propagation between areas, and flood mitigation features in the various areas. The walkdowns encompassed those areas judged to be of possible significance in terms of core damage potential in a flooding zone screening process.

4.4.4.2 Screening Process

Core damage results from the combination of an initiating event and failure of systems or components necessary to the safe shutdown of the plant. The safe shutdown systems are sufficiently reliable to reduce the core damage frequency to within acceptable levels for trip initiators on the order of several per year. Flooding events such as pipe, valve, and tank breaks or ruptures are sufficiently infrequent to be unimportant as trip initiators alone. Only if the same flooding event also degrades safe shutdown capability will the potential for core damage become significant.

The flooding zones judged to be of possible significance were, therefore, those containing both safe shutdown equipment and equipment whose failure would result in a reactor trip. These zones were investigated during the plant walkdowns. Other flooding zones were eliminated from further analysis as possible contributors to core damage.

4.4.4.3 Qualitative Analysis

The information gathered as described in Section 4.4.4.1 was used to analyze the flooding zones with the potential for core damage. The potential for flooding to result in equipment failure was investigated. Many zones were found to have drainage adequate to mitigate the effects of any flooding that could affect the zone. The potential for flood propagation to other zones was investigated. The potential for water spray to result in equipment failure was investigated. Shielding and distance from potential spray sources was considered. Qualification of equipment for operation in adverse environments was considered.

Although electrical switchgear was not identified as being threatened by flooding, water spray from nearby pressurized piping is possible should a pipe leak occur. An electrical bus or motor control center was conservatively assumed to fail if sprayed by a pipe leak. Potential water spray of electrical switchgear was included in estimating initiating event frequencies for the pertinent support system based initiators.

4.4.4.4 Results

All of the flooding zones except for the Unit 1 and 2 Turbine Building Condensate Pump Rooms were eliminated from consideration during the qualitative analysis. The frequency for flooding occurring in the Condensate Pump rooms is approximately 1.3E-02 per year. This event would be similar to a loss of feedwater transient which is already considered in the evaluation of transient events. This contribution to the transient initiator is probabilistically insignificant in comparison with the transient initiator frequency.

Water spray was found to be a contributor to the initiating event frequencies for support system based initiators involving the following electrical switchgear:

480 VAC Motor Control Centers 18-2 and 28-2; and

4 kV Buses 11, 12, 13, 14, 21, 22, 23, and 24.

The highest importance loss of AC special initiator, loss of Bus 13, contributed 0.24% of the total CDF in the Modified IPE. None of the other loss of AC special initiators contributed more than 0.1 percent.

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4.4.5 Equipment Survivability

As part of the Quad Cities IPE, equipment important for prevention of core damage and/or containment failure was evaluated for survivability during the range of postulated accident conditions. To accomplish this task, the Quad Cities equipment survivability study was divided into three phases:

Phase I	-	Support State and Fault Tree Assumptions
Phase II	-	IPE Conditions
Phase III	-	Accident Management/Core Damage Conditions

For Phase I, the assumptions regarding support equipment in the support state and fault tree models were reviewed. Analyses were then completed, as necessary, to verify the assumptions.

Phase II of the study involved reviewing all Plant Response Trees (PRTs) to determine which initiators and events can potentially produce a harsh environment. Also, equipment (including instrumentation) important in achieving successful endstates was determined. The limiting conditions, with respect to the PRTs, were then identified for each piece of equipment and a survivability evaluation was completed.

Phase III of the study will consider the equipment identified for accident management and post-core damage purposes. This will include the equipment needed for post-24-hour accident management to maintain the plant in a safe, stable state (see also SAM discussion in 4.1.3.4); the equipment needed for containment accident management following a core damage event; and any other equipment identified for the overall CECo accident management program.

4.4.5.1 Phase I - Support State and Fault Tree Assumptions

<u>Equipment Identification</u>

The Phase I evaluation is used to support assumptions used in the system fault tree analyses that certain equipment will operate effectively to mitigate a severe accident when system-specific support sub-systems, such as room cooling or ventilation, are unavailable. The support state model and the fault tree models of systems used in the PRTs were reviewed to determine assumptions which impact equipment survivability. The results of the review indicated that assumptions regarding room/area cooling for the RCIC pumps, Core Spray Pumps, HPCI pumps, TBCCW pumps, the Reactor Feed pumps, RHR pumps, RHRSW pumps, and the diesel generators affected the survivability of this equipment.

Assumptions

The support state and fault tree models made the following assumptions regarding equipment survivability:

1. Lack of room coolers will not impact operability of the RCIC pumps, the Core Spray pumps, the HPCI pumps, the TBCCW pumps, and RHR Pumps 1(2)A, 1(2)B, 1C and 1D.

2. Lack of room cooling will result in failure of the Reactor Feed pumps, the RHRSW pumps, RHR Pumps 2C and 2D, and the diesel generators. This assumption is predicated on the supposition that the rooms are in their normal operating configuration (i.e., access doors closed).

Evaluation Methodology

Each component was evaluated assuming a loss of applicable support equipment to determine survivability. The evaluations utilized, but were not limited to, pump/motor data, plant-specific environmental qualification (EQ) data, generic EQ data, equipment design reports, various test data, and specific component analysis. The intent of the evaluation was to compare the environmental conditions the component would be subjected to following a loss of room cooling against the conditions to which the component was designed and/or tested.

1. Nuclear Fuel Services Department report RSA-Q-90-02 was reviewed for applicability for the Core Spray/RCIC pump rooms, the RHR pump rooms, and the HPCI pump rooms. This analysis indicates that the equipment in these rooms will survive the environment posed by loss of room cooling, with the exception of the equipment located in RHR Room 2B (Unit 2 RHR pumps C and D), which will eventually fail unless the room equipment hatch (i.e., removable slab) is removed.

2. Because room cooling failure was assumed to fail the diesels generators, the Reactor Feed pumps, and the RHRSW pumps, no further analysis is required for these components.

3. The TBCCW pumps are located in a mild environment and the fluid in the system is low-temperature and low-pressure. Therefore, the loss of room/area cooling is not expected to significantly increase their rooms' temperature and further analysis is unwarranted.

Conclusions

The support state and fault tree assumptions regarding room cooling are consistent or conservative with respect to the results of the survivability evaluations.

4.4.5.2Phase II - IPE Conditions

The Initiating Events notebook was reviewed to determine the initiators with a potential for producing a harsh environment. The initiating events with that potential were identified as follows:

- Transients (e.g., HELBs outside containment)
- IORVs
- ATWSs
- Large LOCA
- Interfacing Systems LOCA
- Medium and Small LOCA

From an equipment survivability perspective, the ATWS event is considered to be bounded by the large LOCA event; the IORV event is considered to be bounded by the small and medium LOCA equipment survivability evaluations. Transients that result in a harsh environment (e.g., feedwater line rupture or main steam line break outside the containment) have a low probability of occurring. Therefore, HELBs outside containment are removed from further consideration.

Those events within any PRT with a potential for producing a harsh environment were also evaluated. Rupture of high-energy lines in the HPCI, Feedwater, and Main Steam systems could produce such an environment. The Initiating Events notebook documents an analysis discounting this potential on the basis of low probability. As a result, these events will not be considered in this analysis.

The basic event list was reviewed for each of the systems associated with the fault trees supporting the top nodes to determine the equipment assumed to be required for operation. For those top nodes that were human actions, and for the human actions within the fault trees supporting the top nodes, instruments required by the operators were identified. Components identified as necessary to mitigate a large, medium or small LOCA, except those components located in the drywell, are listed in Table 4.4.5-1. The components necessary to mitigate an Interfacing Systems LOCA (ISLOCA), except those located in the drywell, are listed in Table 4.4.5-2. Components located in the drywell were verified to be qualified to a design basis accident in the equipment qualification effort.

<u>Evaluation Methodology</u>

To perform an evaluation of the important equipment, the limiting IPE conditions (i.e., temperature, pressure, humidity) for each piece of equipment were identified based on consideration of important equipment for each initiating event. The limiting IPE conditions were then evaluated versus plant-specific EQ data to determine equipment availability. For those instances in which the equipment is not included in the Quad Cities EQ program, the survivability evaluation was completed by comparing the limiting IPE condition with data excerpted from, but not limited to, generic EQ data, test data, design reports, and specific component analysis.

Results for Phase II

For the Quad Cities Phase II evaluation, all important components have been shown to be available via survivability evaluation.

TABLE 4.4.5-1
COMPONENTS NECESSARY TO MITIGATE A LARGE, MEDIUM OR SMALL LOCA

COMPONENT	LOCATION
CS Pump 1-1401A	NW Corner Room, RB 554'
CS Pump 1-1401B	SW Corner Room, RB 554'
MOV 1-1402-25A	RB 623'
MOV 1-1402-25B	RB 623'
Press SW 263-52A	Inst Rack 2201-5, RB 623'
Press SW 263-52B	Inst Rack 2201-6, RB 623'
RHR Pump 1A-1001	NE Corner Rm, RB 554'
RHR Pump 1B-1002	NE Corner Rm, RB 554'
RHR Pump 1C-1002	SE Corner Rm, RB 554'
RHR Pump 1D-1002	SE Corner Rm, RB 554'
dPIS 261-34A/B	Inst Rack 2201-5, RB 623'
dPIS 261-34C/D	Inst Rack 2201-6, RB 623'
MOVs 1001-5A/B	RB 554'
MOVs 1001-7A/B/C/D	RB 554'
MOVs 1001-16A/B	RB 554'
MOVs 1001-19A/B	RB 554'
MOVs 1001-23A/B	RB 595'/RB 623'
MOVs 1001-26A/B	RB 595'/RB 623'
MOVs 1001-29A/B	RB 591'
MOVs 1001-34A/B	RB 591'
MOVs 1001-36A/B	RB 591'
MOVs 1001-37A/B	RB 591'
MOVs 1001-43A/B/C/D	RB 554'
HPCI TURBINES 1-2301,-2303	HPCI Rm, RB 554'
MOV 1-2301-3	HPCI Rm, RB 554'
MOV 1-2301-8	HPCI Rm, RB 554'
MOV 1-2301-35	HPCI Rm, RB 554'
MOV 1-2301-36	HPCI Rm, RB 554'
PRESS SW 1-2391-10A/B/C/D	HPCI Rm, RB 554'
dPIS 1-2391-02/-03/-6/-07	HPCI Rm, RB 554'
PRESS SW 1-2360	HPCI Rm, RB 554'
PRESS SW 1-2368A/B	HPCI Rm, RB 554'
PRESS SW 1-2303-16	HPCI Rm, RB 554'
TEMP SW 1-2370A/B/C/D	HPCI Rm, RB 554'

TABLE 4.4.5-1 (Continued)
COMPONENTS NECESSARY TO MITIGATE A LARGE, MEDIUM OR SMALL LOCA

COMPONENT	LOCATION
RCIC TURBINE 1-1303	NW Corner Rm, RB 554'
MOV 1-1301-26	NW Corner Rm, RB 554'
MOV 1-1301-60	NW Corner Rm, RB 554'
MOV 1-1301-61	NW Corner Rm, RB 554'
MOV 1-1301-62	NW Corner Rm, RB 554'
PRESS SW 1-1360-9A/B/C/D	NW Corner Rm, RB 554'
PRESS SW 1-1360-21	NW Corner Rm, RB 554'
PRESS SW 1-1360-26A/B	NW Corner Rm, RB 554'
dPIS 1-1360-1A/B	NW Corner Rm, RB 554'
TEMP SW 1-1350-14A/B/C/D	NW Corner Rm, RB 554'
AIR OPERATED DAMPER 1601-23	RB 647'
AIR OPERATED DAMPER 1601-24	RB 647'
AIR OPERATED DAMPER 1601-60	RB 647'
MCC 18-1A	RB 623'
MCC 18-1B	RB 623'
MCC 18/19-5	RB 595'
MCC 19-1	RB 623'
MCC 19-4	RB 623'
RB 125VDC MCC 1A	RB 623'
RB 250VDC MCC 1B	RB 623'

TABLE 4.4.5-2
COMPONENTS NECESSARY TO MITIGATE AN INTERFACING SYSTEMS LOCA

COMPONENT	LOCATION
CS PUMP 1-1401A	NW Corner Rm, RB 554'
CS PUMP 1-1401B	SW Corner Rm, RB 554'
MOV 1-1402-25A	RB 623'
MOV 1-1402-25B	RB 623'
PRESS SW 263-52A	Inst Rack 2201-5, RB 623'
PRESS SW 263-52B	Inst Rack 2201-6, RB 623'
LIS 263-72A/C	Inst Rack 2201-5, RB 623'
LIS 263-72B/D	Inst Rack 2201-6, RB 623'
MCC 18-1A	RB 623'
MCC 19-1	RB 623'

4.4.5.3 Phase III - Accident Management/Core Damage Sequences

Preliminary lists of important components and instruments for Phase III are presented in Tables 4.4.5-3 and 4.4.5-4, respectively. The identification and survivability evaluations for this portion of the Quad Cities equipment survivability program is beyond the scope of this report; these evaluations will be included as part of the implementation phase of an Accident Management program.

TABLE 4.4.5-3
QUAD CITIES PHASE III IMPORTANT EQUIPMENT

EQUIPMENT IMPORTANT TO PHASE III	
1.	Residual Heat Removal Pump(s) (All modes of operation)
2.	Core Spray Pump(s)
3.	Residual Heat Removal Heat Exchangers
4.	ADS Valves
5.	Condensate Pump(s) (Standby Coolant Supply)
6.	Residual Heat Removal Service Water Pump(s)
7.	Control Rod Drive Pump(s)
8.	Clean Demin Pump(s)
9.	Torus/Drywell Vent Valves
10.	Reactor Building/Wetwell Vacuum Breakers
11.	Atmospheric Containment Atmosphere Dilution System Components
12.	Fire Protection Pump(s)
13.	Drywell Coolers

TABLE 4.4.5-4
QUAD CITIES PHASE III IMPORTANT INSTRUMENT FUNCTIONS

INSTRUMENTATION IMPORTANT TO PHASE III	
1.	RPV Pressure
2.	RPV Level
3.	RHR Pump Flow
4.	Core Spray Pump Flow
5.	Condensate Pump Flow
6.	Suppression Pool Temperature
7.	Suppression Pool Level
8.	Containment Pressure
9.	Containment Temperature
10.	Containment Radiation
11.	Containment Hydrogen
12.	Containment Water Level

4.5 Sequence and Source Term Quantification

Accident sequence quantification for the Quad Cities IPE began with the support system event trees, which were loaded into the QT code system using CADET, the event tree editor. Fault trees were developed to address each node of the support system event trees consistent with the operating states defined for each support system. The support system fault trees were quantified using the GRAFTER code system.

Typically, the support system event trees are evaluated to determine which paths have similar impact on the frontline systems. This impact analysis results in a smaller set of support states (combinations of support system event tree paths) for which the accident sequence event trees are quantified. Therefore, each plant response tree is quantified for each significant support state and the results are combined with the initiating event probability. For the Quad Cities IPE study, support state grouping through frontline system impact analysis is not necessary since computer codes that can analyze the models completely in a timely manner are readily available. Unique support system event trees are developed in the Support State Model notebook for the following events:

1. transient events, LOCAs, and special initiators,
2. loss of offsite power at Unit 1, and
3. loss of offsite power at both units.

The plant response trees were developed and loaded into the QT code system using CADET. Analysis was then performed to provide failure probabilities for the nodes of each plant response tree. Fault trees consistent with the system success criteria were developed using the GRAFTER code system for a majority of the nodes. These fault trees were quantified for the case with all supporting equipment available and for various degraded cases representing loss of specific support equipment. Fault tree results are presented in Section 4.5.2. The operator action nodal failure probabilities were determined using the THERP methodology. Also, hand calculations were used, in several cases, to determine failure probabilities for nodes which were not system or human error related.

Some nodes were determined to be dependent upon other nodes which preceded them on the plant response trees. In order to account for these dependencies, and ensure a correct quantification of the accident sequence, conditional failure probabilities were calculated and used in place of the fault tree quantification results as appropriate.

Sequence quantification was then performed on the entire plant model with a calculational cutoff of 1.0E-12. This quantification was done by appending each PRT to each support system event tree path. Therefore, PRT sequences include the support system model nodes. Each plant response tree was quantified with its associated initiating event frequency for each path of its associated support system event tree. The QT code multiplied each accident sequence and placed the resulting probability in the designated damage state bin. If the multiplicative probability for a specific sequence dropped below the calculational cutoff during quantification, quantification of that path was halted and the probability was placed in the "residual" bin which was reported as a plant damage state. Plant response tree quantification results are presented in Section 4.5.3.

The support system event tree used in the quantification of each plant response tree is shown in Table 4.5-1.

4.5.1 Support System States and Probabilities

Typically, the support system event trees are grouped according to impact on the frontline systems and quantified. For the Quad Cities IPE study, support state grouping through frontline system impact analysis is not necessary since computer codes that can analyze the models completely in a timely manner are readily available. Tables 4.5.1-1 through 4.5.1-3 list the most frequent support system combinations which were identified and their probabilities, according to initiating event.

4.5.2 Unavailability of Systems and Plant Functions

Table 4.5.2-1 (located in Volume 2) provides a listing of failure probabilities for systems and plant functions used in the Quad Cities IPE. The unavailabilities are grouped by system and function. Within each group unavailabilities are delineated by success criteria and initiating event as appropriate. Each listing provides a description of the success criteria and the failure probability which resulted from fault tree analysis or hand calculation.

4.5.3 Accident Sequence Frequencies

Table 4.5.3-1 (located at the end of this subsection) individually lists the top 100 accident sequences for the modified IPE model quantification. These account for more than 94% of the total core damage frequency. For each of these sequences the following information is provided:

- Accident sequence frequency
- Percentage contribution to total core damage frequency
- Plant damage state (bin)
- Initiating event name and frequency
- Failed support state event tree nodes, probabilities, and descriptions
- Failed plant response tree nodes, probabilities, and descriptions

The total plant damage state frequency is 2.17E-06 for the modified IPE model quantification.

As an example of dominant sequence generation, sequence #3 from the original IPE submittal is described in detail. Sequence #3 is a loss of offsite power; the initiating event frequency is 3.2E-02. To follow the accident progression, it is necessary to address the availability of the support systems in the support state event tree (SSET) for a loss of offsite power. The values assigned each node are from the original IPE quantification and are shown in Table 4.5.3-2. "Success" as used in Table 4.5.3-2 means that the quantification code assigns a value of one minus the node failure probability. The probabilistic value associated with each SSET node is described below:

TABLE 4.5-1
CORRELATION BETWEEN SUPPORT SYSTEM EVENT TREES AND PRTs

SUPPORT SYSTEM EVENT TREE	PLANT RESPONSE TREE
Transient, Loss of Coolant Accidents (LOCA), and Special Initiators	Small LOCA (SLOCA) Medium LOCA (MLOCA) Large LOCA (LLOCA) Interfacing Systems LOCA (ISLOCA) Transient (GTR)[1] Anticipated Transient Without Scram (ATWS) Inadvertently Open Relief Valve (IORV)
Single Unit Loss of Offsite Power (LOOP)	Loss of Offsite Power (LOOP) Station Blackout (SBO)
Dual Unit Loss of Offsite Power (DLOOP)	Loss of Offsite Power (LOOP) Station Blackout (SBO)

[1] The Transient PRT was also used to quantify the following Special Initiators: Loss of Instrument Air, Loss of Service Water, Loss of Bus 11, Loss of Bus 12, Lossof Bus 13, Loss of Bus 14, Loss of Bus 18, Loss of MCC 18-2, and Loss of 125VDC Bus 1B-1.

TABLE 4.5.1-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL QUANTIFICATION RESULTS

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	EVENT (4)	VALUE (5)	DESCRIPTION (6)	MODIFIED IPE NUMBER(S) (7)
1	3.86E+00	96.11	GTR	3.87E+00	GENERAL TRANSIENT IE	5, 8, 12, 16, 69, 86, 88, 94
3	1.43E-02	0.36	LOIA	1.43E-02	LOSS OF INSTRUMENT AIR IE (EXCL LOSW CONT.)	39
			1IA	1.00E+00	EVENT FAILS	
5	9.07E-03	0.23	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)	9, 37, 46, 68
			SW	1.00E+00	EVENT FAILS	
			1IA	1.00E+00	EVENT FAILS	
11	7.97E-04	0.02	MLOCA	8.00E-04	MLOCA IE	4, 60, 61, 93
20	2.99E-04	0.01	LLOCA	3.00E-04	LLOCA IE	31
25	1.61E-04	0	GTR	3.87E+00	GENERAL TRANSIENT IE	96
			1IA	4.17E-05	IA FAILS	
27	1.16E-04	0	ATWS	1.16E-04	ATWS INITIATOR	10, 14, 15, 17, 23, 25, 28, 30, 36, 43, 48, 50, 53, 57, 59, 62, 63, 80, 84
47	8.82E-06	0	GTR	3.87E+00	GENERAL TRANSIENT IE	91
			131	1.84E-04	LOSS OF BUS 13-1, 13 AVAIL	
			141	1.26E-02	LOSS OF BUS 14-1 AFTER 13-1, 14 AVAIL	
77	1.89E-06	0	MLOCA	8.00E-04	MLOCA IE	92
			14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)	
*	6.11E-10	0.03	IORV	1.06E-01	IORV + OTHER IEs x RVC	90
			1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A	
			1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1	
*	1.31E-08	0.6	GTR	3.87E+00	GENERAL TRANSIENT IE	20, 22
			1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A	
			1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1	
*	4.08E-08	1.88	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE	11, 13, 51, 65
			1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A	
			1R1	1.00E+00	EVENT FAILS	
*	6.04E-09	0.28	LLOCA	3.00E-04	LLOCA IE	29, 45
			14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)	
*	1.12E-09	0.05	LLOCA	3.00E-04	LLOCA IE	66
			1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A	
*	9.47E-10	0.04	LLOCA	3.00E-04	LLOCA IE	72
			14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)	
			DG1	1.37E-01	LOP FROM DG1 TO BUS 14-1 (24 HRS)	
*	6.38E-10	0.03	LLOCA	3.00E-04	LLOCA IE	89
			1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1	

* The Support State sequence is not in the support state Top 100 report, but is in the Dominant Sequence Top 100 report.

Notes:

1. "Number" refers to support state model sequence.
2. "Frequency" is the frequency per year that this initiator/support combination is expected to occur.
3. "Percent" is the percent of off-normal conditions for the subject initiators that would involve this state.
4. "Event" is the model top event label.
5. "Value" is frequency (for initiators) or probability (for failures) that the event would occur.
6. "Description" defines the event label.
7. "Modified IPE Number(s)" refers to dominant accident sequence numbers listed in Table 4.5.3-1.

TABLE 4.5.1-2
LOOP SUPPORT MODEL QUANTIFICATION RESULTS

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	EVENT (4)	VALUE (5)	DESCRIPTION (6)	MODIFIED IPE NUMBER(S) (7)
1	2.47E-02	77.15	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	3, 21, 33, 35, 38, 70, 81
			1TB	1.00E+00	EVENT FAILS	
2	2.42E-03	7.57	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	18, 19, 67
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
3	2.41E-03	7.53	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	34, 49, 71, 77, 78, 95
			DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
4	1.21E-03	3.79	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	24
			SW	4.59E-02	FAILURE OF SW (LOOP)	
			1IA	1.00E+00	EVENT FAILS	
5	4.94E-04	1.54	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	40
			1TB	1.00E+00	EVENT FAILS	
			1IA	1.96E-02	IA FAILS (LOOP, DLOOP)	
6	2.55E-04	0.8	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	55, 99
			DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)	
			DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
7	1.19E-04	0.37	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	79
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			SW	4.59E-02	FAILURE OF SW (LOOP)	
			1IA	1.00E+00	EVENT FAILS	
8	1.18E-04	0.37	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	85
			DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)	
			SW	4.59E-02	FAILURE OF SW (LOOP)	
			1IA	1.00E+00	EVENT FAILS	
14	2.75E-05	0.09	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	64
			DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)	
			18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL	
			1TB	1.00E+00	EVENT FAILS	
22	2.95E-06	0.01	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT	27, 100
			DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)	
			DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)	
			141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL	
			SBO?	1.00E+00	SBO OCCURS IN UNIT 1	

Notes:

1. "Number" refers to support state model sequence.
2. "Frequency" is the frequency per year that this initiator/support combination is expected to occur.
3. "Percent" is the percent of off-normal conditions for the subject initiators that would involve this state.
4. "Event" is the model top event label.
5. "Value" is frequency (for initiators) or probability (for failures) that the event would occur.
6. "Description" defines the event label.
7. "Modified IPE Number(s)" refers to dominant accident sequence numbers listed in Table 4.5.3-1.

TABLE 4.5.1-3
DUAL UNIT LOOP SUPPORT MODEL QUANTIFICATION RESULTS

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	EVENT (4)	VALUE (5)	DESCRIPTION (6)	MODIFIED IPE NUMBER(S) (7)
1	1.16E-02	72.08	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	7, 33, 42, 52
			1TB	1.00E+00	EVENT FAILS	
2	1.14E-03	7.08	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	27, 41, 74
			DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
3	1.09E-03	6.8	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	26
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
4	9.61E-04	5.98	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	32
			DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
5	3.47E-04	2.16	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	44
			SW	2.85E-02	FAILURE OF SW (DLOOP), 23 UNAVAIL	
6	2.32E-04	1.44	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	56
			1TB	1.00E+00	EVENT FAILS	
			1IA	1.96E-02	IA FAILS (LOOP, DLOOP)	
7	1.20E-04	0.74	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	82
			DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)	
			DGB	1.12E-01	LOSS OF DG1/2 AFTER DG2, (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
8	1.20E-04	0.74	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	54, 83
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)	
			SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1	
9	1.16E-04	0.72	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	87
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)	
			1TB	1.00E+00	EVENT FAILS	
17	1.90E-05	0.12	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	1, 2, 6
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)	
			DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)	
			SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2	
21	1.30E-05	0.08	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	95
			DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)	
			18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL	
			1TB	1.00E+00	EVENT FAILS	
37	1.37E-06	0.01	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	98
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)	
			19	1.13E-02	LOSS OF BUS 19, 14-1 UNAVAIL	
			SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1	
39	1.30E-06	0.01	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	75
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)	
			141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL	

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	EVENT (4)	VALUE (5)	DESCRIPTION (6)	MODIFIED IPE NUMBER(S) (7)
			SBO?	1.00E+00	SBO IN UNIT 1, NO SBO IN UNIT 2	
*	1.31E-09	0.06	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	58
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)	
			DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)	
			1ES	1.51E-03	LOSS OF ESS BUS (901-49), 17 & 18 UNAVAIL	
			SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2	
*	8.26E-10	0.04	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS	76
			1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A	
			DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)	
			DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)	
			SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2	

* The Support State sequence is not in the support state Top 100 report, but is in the Dominant Sequence Top 100 report.

Notes:

1. "Number" refers to support state model sequence.
2. "Frequency" is the frequency per year that this initiator/support combination is expected to occur.
3. "Percent" is the percent of off-normal conditions for the subject initiators that would involve this state.
4. "Event" is the model top event label.
5. "Value" is frequency (for initiators) or probability (for failures) that the event would occur.
6. "Description" defines the event label.
7. "Modified IPE Number(s)" refers to dominant accident sequence numbers listed in Table 4.5.3-1.

TABLE 4.5.2-1

UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS
(IN VOLUME 2)

TABLE 4.5.3-1
DOMINANT ACCIDENT SEQUENCES FOR MODIFIED IPE MODEL QUANTIFICATION

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
1	8.62E-07	39.8	BLAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
2	1.56E-07	7.2	BEAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				ROP2	1.00E+00	FAILURE TO REC OSP; SBO, SHORT TIME AVAILABLE
3	1.46E-07	6.72	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
4	1.03E-07	4.74	MEFGS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
5	8.75E-08	4.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
6	7.04E-08	3.25	BLAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				ROP2	5.09E-02	FAILURE TO REC OSP
7	6.84E-08	3.15	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
8	5.79E-08	2.67	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
9	5.35E-08	2.47	TEABS	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				CRD	1.00E+00	EVENT FAILS
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
10	5.22E-08	2.41	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
11	4.08E-08	1.88	TEFEB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
12	2.85E-08	1.31	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
13	2.72E-08	1.25	TEFEB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
14	2.43E-08	1.12	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
15	2.04E-08	0.94	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
16	1.89E-08	0.87	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
17	1.56E-08	0.72	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
18	1.43E-08	0.66	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
19	1.35E-08	0.62	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
20	1.31E-08	0.6	TEFEB	GTR	3.87E+00	GENERAL TRANSIENT IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
21	1.05E-08	0.48	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
22	8.73E-09	0.4	TEFEB	GTR	3.87E+00	GENERAL TRANSIENT IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
23	7.27E-09	0.34	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
24	7.15E-09	0.33	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
25	6.92E-09	0.32	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
26	6.46E-09	0.3	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
27	6.38E-09	0.29	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
28	6.09E-09	0.28	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
29	6.04E-09	0.28	ALCEB	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
30	5.80E-09	0.27	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
31	5.71E-09	0.26	AEGGA	LLOCA	3.00E-04	LLOCA IE
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
				LV	2.89E-03	LV FAILS; ALL SUPPORTS AVAILABLE
32	5.67E-09	0.26	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
33	5.28E-09	0.24	LLABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)
34	5.21E-09	0.24	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
35	5.04E-09	0.23	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
36	4.31E-09	0.2	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
37	3.86E-09	0.18	TLABS	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
38	3.29E-09	0.15	LLCOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
39	3.17E-09	0.15	TEFBS	LOIA	1.43E-02	LOSS OF INSTRUMENT AIR IE (EXCL LOSW CONT.)
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				OFW1	8.40E-03	OPTR FAILS TO RSTRT A FW PMP OR RCVR HW LVL W/ MNL MU VLV
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
40	2.91E-09	0.13	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				1IA	1.96E-02	IA FAILS (LOOP, DLOOP)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
41	2.64E-09	0.12	LLBOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
42	2.54E-09	0.12	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
43	2.07E-09	0.1	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
44	2.05E-09	0.09	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				SW	2.85E-02	FAILURE OF SW (DLOOP), 23 UNAVAIL
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
45	2.04E-09	0.09	ALCEB	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	1.00E+00	EVENT FAILS
				SBCS	1.00E+00	EVENT FAILS

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
46	1.85E-09	0.09	TEABS	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
47	1.83E-09	0.08	BLASB	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO OCCURS IN UNIT 1
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
48	1.82E-09	0.08	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				SLC	2.35E-04	SLC FAILS; 1/2 PUMPS; ALL SUPPORTS AVAILABLE
49	1.79E-09	0.08	LLCOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP B SUCCESS
50	1.73E-09	0.08	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
51	1.70E-09	0.08	TEFSB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
52	1.66E-09	0.08	LLCOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
53	1.62E-09	0.07	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				RPT1	1.99E-04	AUTO RPT FAILS; ALL SUPPORTS AVAILABLE
54	1.45E-09	0.07	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.00E+00	EVENT FAILS
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
55	1.43E-09	0.07	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
56	1.37E-09	0.06	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				1IA	1.96E-02	IA FAILS (LOOP, DLOOP)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
57	1.34E-09	0.06	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				AT1	1.28E-02	ATWS1 ACTUATION FAILS
				AT2	1.28E-02	ATWS2 ACTUATION FAILS (GIVEN AT1 SUCCESS)
58	1.31E-09	0.06	BLAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				1ES	1.51E-03	LOSS OF ESS BUS (901-49), 17 & 18 UNAVAIL
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
59	1.29E-09	0.06	TEERF	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
60	1.24E-09	0.06	MEFBS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				OCNTS	1.20E-02	OPTR FAILS TO INITIATE CONT SPR (1)
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
61	1.24E-09	0.06	MEFGS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
				LV	2.89E-03	LV FAILS; ALL SUPPORTS AVAILABLE
62	1.21E-09	0.06	TEFGS	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
63	1.18E-09	0.05	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	8.70E-03	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (1)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				OAL	1.40E-01	OPTR FAILS TO CONTROL RV LEVEL AFTER ATWS (11)
64	1.14E-09	0.05	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	1.00E+00	EVENT FAILS
65	1.13E-09	0.05	TEFSB	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)
66	1.12E-09	0.05	AEGGA	LLOCA	3.00E-04	LLOCA IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				LV	1.00E+00	EVENT FAILS
67	1.03E-09	0.05	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
68	9.87E-10	0.05	TLBSB	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
69	9.86E-10	0.05	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
70	9.74E-10	0.04	LEAHD	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
71	9.71E-10	0.04	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)
72	9.47E-10	0.04	ALCEB	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				DG1	1.37E-01	LOP FROM DG1 TO BUS 14-1 (24 HRS)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
73	9.21E-10	0.04	LLABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
				RCIC2	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				HP2	2.30E-01	HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE
				OAD1	1.30E-03	OPTR FAILS TO INITIATE ADS (2)
74	9.09E-10	0.04	LLCOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP B SUCCESS

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
75	8.59E-10	0.04	BLASB	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO IN UNIT 1, NO SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.99E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 1T2 AVAILABLE
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
76	8.26E-10	0.04	BEAYF	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; 1R1, 1T2 AVAILABLE
				RCIC	1.00E+00	EVENT FAILS
				ROP2	1.00E+00	FAILURE TO REC OSP; SBO, SHORT TIME AVAILABLE
77	8.00E-10	0.04	LEACD	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	9.60E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (19)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				ROP1	1.00E+00	EVENT FAILS
78	7.43E-10	0.03	LEAHD	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
79	7.02E-10	0.03	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
80	6.98E-10	0.03	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	8.70E-03	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (1)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
81	6.83E-10	0.03	LLCOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP A SUCCESS

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
82	6.70E-10	0.03	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)
				DGB	1.12E-01	LOSS OF DG1/2 AFTER DG2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
83	6.69E-10	0.03	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
84	6.65E-10	0.03	TEEQC	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				AT1	1.28E-02	ATWS1 ACTUATION FAILS
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
85	6.62E-10	0.03	LEABS	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
86	6.53E-10	0.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	2.05E-03	FW FAILS; ALL SUPPORTS AVAILABLE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)
87	6.49E-10	0.03	LEABS	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
88	6.40E-10	0.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				OAD1	1.60E-03	OPTR FAILS TO INITIATE ADS (10)

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
89	6.38E-10	0.03	AEGGA	LLOCA	3.00E-04	LLOCA IE
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				CS	5.24E-02	CS FAILS; 13-1, 18, 1M1 AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				LV	5.01E-01	LV FAILS; 18, 19, 1M1 AVAILABLE
90	6.11E-10	0.03	IEBOG	IORV	1.06E-01	IORV + OTHER IEs x RVC
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
91	6.01E-10	0.03	TIGSB	GTR	3.87E+00	GENERAL TRANSIENT IE
				131	1.84E-04	LOSS OF BUS 13-1, 13 AVAIL
				141	1.26E-02	LOSS OF BUS 14-1 AFTER 13-1, 14 AVAIL
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				CST	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
92	6.00E-10	0.03	MLCSB	MLOCA	8.00E-04	MLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)
93	5.87E-10	0.03	MEFGS	MLOCA	8.00E-04	MLOCA IE
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				ADS	9.14E-06	ADS FAILS; ALL SUPPORTS AVAILABLE
94	5.85E-10	0.03	TEFHS	GTR	3.87E+00	GENERAL TRANSIENT IE
				OFW1	1.40E-03	OPTR FAILS TO RESTART A FW PUMP (2)
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	6.30E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
95	5.79E-10	0.03	LLBOG	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	1.00E+00	EVENT FAILS

TABLE 4.5.3-1 (Continued)

Seq. (1)	Freq. (2)	Percent (3)	Damage State (4)	Node (5)	Value (6)	Description (7)
96	5.78E-10	0.03	TEFBS	GTR	3.87E+00	GENERAL TRANSIENT IE
				1IA	4.17E-05	IA FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.40E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
97	5.75E-10	0.03	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	5.75E-03	CCST SUCTION VALVES FAIL, CS SUCCESS
98	5.64E-10	0.03	LEBGA	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				19	1.13E-02	LOSS OF BUS 19, 14-1 UNAVAIL
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 13-1, 18, 1M1 AVAILABLE
				LV	1.00E+00	EVENT FAILS
99	5.52E-10	0.03	LLBOG	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
100	5.48E-10	0.03	BLATF	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO OCCURS IN UNIT 1
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)

Notes:
1. "Seq." refers to accident sequence ranking in the top 100 sequences.
2. "Freq." is the frequency per year that this sequence is expected to occur.
3. "Percent" is the percent of total core damage represented by this single sequence.
4. "Damage State" is the plant damage state to which this sequence belongs. The fifth character presents the release associated with this type of sequence and is manually assigned at the end of the analysis in presentations of dominant sequences.
5. "Node" is the list of PRT and support system event tree top events which have failed in this sequence.
6. "Value" is frequency (for initiators) or probability (for failures) associated with each node.
7. "Description" defines the "Node" label.

TABLE 4.5.3-2

UNAVAILABILITY OF PLANT SYSTEMS & FUNCTIONS FOR SEQUENCE NUMBER 3 OF THE ORIGINAL IPE

SSET NODE	VALUE	EXPLANATION
1M1	9.99927E-01	Success of Unit 1 Main 125VDC Bus
1R1	9.99916E-01	Success of Unit 1 Reserve 125VDC Bus
DGB	9.29E-01	DG 1/2 Starts and Runs
DG1	9.219E-01	DG 1 Starts and Runs
131	9.99816E-01	Success of Unit 1 Bus 13-1
141	9.99816E-01	Success of Unit 1 Bus 14-1
13	9.944E-01	Success of Unit 1 Bus 13
14	9.945E-01	Success of Unit 1 Bus 14
1CA	9.99923E-01	Success of Unit 1 Common Actuation
15	9.99998E-01	Success of Unit 1 Bus 15
16	9.99998E-01	Success of Unit 1 Bus 16
17	9.99998E-01	Success of Unit 1 Bus 17
18	9.99997E-01	Success of Unit 1 Bus 18
19	9.99997E-01	Success of Unit 1 Bus 19
SW	9.541E-01	Success of Service Water (Shared by Both Units)
1TB	1.00E+00	Failure of Unit 1 TBCCW due to initiating event
1IA	9.804E-01	Success of Unit 1 Instrument Air
1T2	9.99989E-01	Success of Unit 1 Turbine Building 250VDC Bus

TABLE 4.5.3-2 (Continued)

PRT NODE	VALUE	EXPLANATION
1IB	9.99996E-01	Success of Unit 1 120VAC Instrument Bus
1ES	9.99996E-01	Success of Unit 1 120VAC Essential Services Bus
SBO	1.00E+00	Station Blackout Does NOT Occur

Accident sequence #3 continues with the loss of offsite power (LOOP) PRT. The probabilistic value associated with each PRT node is described below:

RC	9.9997E-01	Success of Reactivity Control (Reactor Scram)
RVO	9.99999E-01	Relief Valves/Safety Valves Open
RVC	9.73E-01	Relief Valves/Safety Valves Close
HP1	9.582E-01	Success of Automatic Initiation/Operation of the HPCI System
INVC	9.99597E-01	Successfully Maintaining CCST Inventory
LPA	9.9594E-01	Success of RHR Pump - Train A
LPB	9.9623E-01	Success of RHR Pump - Train B
OHX	1.90E-03	Failure of Operator to Align Cooling to RHR
RHRHX	1.00E+00	RHR Heat Exchanger (Not asked in this scenario; no branch at this node.)
OSPC	1.00E+00	Operator Action to Initiate Suppression Pool Cooling (Not asked in this scenario; no branch at this node.)
SPC	1.00E+00	Suppression Pool Cooling Operation (Not asked in this scenario; no branch at this node.)
RCIC	1.00E+00	Reactor Core Isolation Cooling (Not asked in this scenario; no branch at this node.)
OAD2	9.941E-01	Operator Successfully Initiates Depressurization

TABLE 4.5.3-2 (Continued)

PRT NODE	VALUE	EXPLANATION
ADS	9.99999E-01	Success of the Automatic Depressurization System
OSMP1	9.975E-01	Operator Successfully Initiates SSMP Aligned to CCST
SSMP1	2.82E-02	Failure of SSMP (Aligned to CCST)
ROP1	1.00E+00	Failure to Recover Offsite Power
OSS	1.00E+00	Operator Action to Recover SW/TBCCW (Not asked in this scenario; no branch at this node.)
OCRD	1.00E+00	Operator Action to Restore CRD (Not asked in this scenario; no branch at this node.)
CRD	1.00E+00	Control Rod Drive Injection (Not asked in this scenario; no branch at this node.)
OFW3	1.00E+00	Operator Action to Restart (fill/vent) FW Following LOOP (Not asked in this scenario; no branch at this node.)
FW	1.00E+00	Feedwater/Condensate (Not asked in this scenario; no branch at this node.)
OCST	2.30E-02	Failure of Operator to Align to CCST Source
CST	1.00E+00	CCST Suction Valves (Not asked in this scenario; no branch at this node.)
CS	1.00E+00	Core Spray (Not asked in this scenario; no branch at this node.)
LV	1.00E+00	RHR Injection Valves (Not asked in this scenario; no branch at this node.)
OCNTS	1.00E+00	Operator Action to Initiate Containment Sprays (Not asked in this scenario; no branch at this node.)

TABLE 4.5.3-2 (Continued)

PRT NODE	VALUE	EXPLANATION
CNTS	1.00E+00	Hardware Required for Containment Sprays (Not asked in this scenario; no branch at this node.)
OVNT	9.49E-01	Operator Successfully Takes Action to Vent the Containment
LVW	9.9407E-01	Success of Hardware Required for the 8-Inch Wetwell Vent
LVD	1.00E+00	Hardware Required for the 8-Inch Drywell Vent (Not asked in this scenario; no branch at this node.)
WW/DW	1.00E+00	Location of Containment Failure (Not asked in this scenario; no branch at this node.)

The quantification code, QT, multiplies all of the probabilistic values associated with sequence #3 together to yield the sequence frequency. The product of the column of probabilistic values above and the initiating event frequency is 2.70E-08.

A quantification was also performed in which the SAM endstate was set to a damage state. The total plant damage state frequency for this run is 1.82E-06 and indicates the need for accident management which allows these SAMs to be success states.

Quantification of Special Initiators

The special initiators were quantified using the General Transient plant response tree using initiator specific initiating event frequencies. The conditions caused by the individual initiators were imposed upon the SSET and PRT as appropriate through the quantification code. The structure of the transient event tree was not changed. The following paragraphs describe the conditions caused by the initiators and considered in the quantification of the events:

Loss of Instrument Air (LOIA)
The components/systems of concern given this special initiator are the MSIVs, the containment vents, the hotwell level control valves and the service water strainers. Loss of instrument air causes 1) the MSIVs to close, 2) the inability to operate the containment vents, 3) the requirement for manual hotwell level control and 4), during periods of high likelihood of service water strainer fouling, loss of service water. These events were modeled in the quantification of LOIA using appropriate values for nodes on the general transient tree, without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of Service Water (LOSW)
The components/systems of concern, given this special initiator, are the instrument air system, the TBCCW system and those components cooled by service water. Loss of service water results in the loss of the instrument air system (and the conditions caused by that event) and the loss of the TBCCW system. These events were modeled in the quantification of LOSW using appropriate values for nodes on the general transient tree, without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of 125VDC Bus 1B-1 (L1B1)
Loss of this bus results in the loss of one of the two available control power buses for plant equipment. This event has broad plant implications. The event was modeled in the quantification of L1B1 by setting the node representing this bus to a failure probability of 1.0 and quantifying the general transient tree, without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems

that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of Bus 11
Loss of this bus impacts the feedwater pump supplied by the bus and thus impacts the availability of the feedwater system as a makeup source of water. The event was modeled in the quantification of LB11 by incorporating fault tree results for the feed system in which this bus was unavailable and quantifying the general transient tree, without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of Bus 12
Loss of this bus impacts the feedwater pump supplied by the bus and thus impacts the availability of the feedwater system as a makeup source of water. The event was modeled in the quantification of LB12 by incorporating fault tree results for the feed system in which this bus was unavailable and quantifying the general transient tree, without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of Bus 13
Loss of this bus impacts several frontline systems and, therefore, this bus was included in the support system event tree explicitly as a node. The Loss of Bus 13 event was modeled in the quantification of LB13 by quantifying the general transient tree with this bus set to "fail", without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of Bus 14
Loss of this bus impacts several frontline systems and, therefore, this bus was included in the support system event tree explicitly as a node. The Loss of Bus 14 event was modeled in the quantification of LB14 by quantifying the general transient tree with this bus set to "fail", without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of Bus 18
Loss of this bus impacts several frontline systems and, therefore, this bus was included in the support system event tree explicitly as a node. The Loss of Bus 18 event was modeled in the quantification of LB18 by quantifying the general transient tree with this bus (and

PCS) set to “fail”, without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

Loss of MCC 18-2

Loss of this MCC impacts both instrument power and essential services power. These power supplies impact the containment vents, the PCS and the capability to provide cooling to the RHR heat exchanger. The Loss of MCC 18-2 event was modeled in the quantification of L182 by quantifying the general transient tree with this MCC set to “fail”, without changing the structure of this SSET/PRT. The component dependencies described in the dependency matrices identify the systems that are impacted by the loss of this support system and that are modeled in the quantification of the SSET/PRT.

4.5.4 Accident Sequence Sensitivity Analysis

4.5.4.1 Sensitivity Analyses on the Original Baseline Model

The following Accident Sequence Sensitivity Analysis discussion was submitted in the original IPE Submittal Report and is presented below for illustrative purposes only.

The following Quad Cities IPE Model "parameters" were evaluated in the sensitivity analysis studies. The purpose of conducting these sensitivity analyses was to evaluate assumptions or data which if varied could have a significant impact on the analysis results, and thereby address model uncertainty.

Human Error Probabilities

Several operator actions are significant in the IPE results. These operator actions (OA) are listed below:

OAD1 (All events): Operator action to depressurize the reactor vessel
OHX (All events): Operator action to align cooling to RHR
OCST (All events): Operator action to align low pressure pumps to the CCST

The human error probabilities (HEPs) for these operator actions were varied (increased or decreased by an order of magnitude) one at a time to determine the sensitivity of the overall IPE model results to these values. Each operator action has several probabilities due to sequence timing and/or dependencies of other operator actions. If the probability is greater than 0.1, that specific case was not varied because its value is determined primarily by dependencies on other actions.

The results of these sensitivities are summarized in Table 4.5.4-1. Included in the table are the different sets of values used for the operator action probabilities. Beside each set of values is the respective calculated core damage frequency.

Recovery of Offsite Power

The base IPE model was quantified taking no credit for recovery of offsite power during those events in which some onsite source of AC power was available. This sensitivity analysis was performed to determine the impact of this assumption. The results of this sensitivity analysis are provided in Table 4.5.4-1.

Conclusions

From the results of the sensitivity analysis presented in Table 4.5.4-1, several conclusions can be drawn. Increasing the human error probabilities for the significant operator actions by an order of magnitude does not result in significant changes in the core damage frequency. The model is sensitive to changes in the human error probability for the operator actions to: a) depressurize the reactor pressure vessel, b) establish RHR cooling, and c) align low pressure pump suctions to the CCST. Increasing the value of OAD1 by one order of magnitude changed the core damage frequency from 1.2E-6 to 3.5E-6, an increase of about 190%. Increasing OHX by an order of magnitude changed the core damage frequency from 1.2E-6 to 2.4E-6, an increase of approximately 98%. Increasing OCST by an order of magnitude changed the core damage frequency from 1.2E-6 to 2.4E-6, an increase of about 98%. Although some of these changes appear

TABLE 4.5.4-1
SENSITIVITY ANALYSIS RESULTS

PRT Node	BASE IPE MODEL		SENSITIVITY CASES	
	Value	Core Damage Frequency	Value	Core Damage Frequency
OAD1				
OD1-CS1	4.90E-03	1.20E-06	4.90E-02	3.47E-06
OD1-CS2	5.40E-04		5.40E-03	
OD1-CS4	6.00E-03		6.00E-02	
OD1-CS9	9.80E-03		9.80E-02	
OD1-CS10	1.10E-03		1.10E-02	
OD1-CS11	9.80E-03		9.80E-02	
OD1-CS12	6.50E-03		6.50E-02	
OD1-CS17	2.50E-02		2.50E-01	
OD1-CS18	2.70E-03		2.70E-02	
OD1-CS19	7.40E-02		7.40E-01	
OD1-CS20	2.70E-03		2.70E-02	
OD1-CS21	7.40E-02		7.40E-01	
OAD1				
OD1-CS1	4.90E-03	1.20E-06	4.90E-04	9.84E-07
OD1-CS2	5.40E-04		5.40E-05	
OD1-CS4	6.00E-03		6.00E-04	
OD1-CS9	9.80E-03		9.80E-04	
OD1-CS10	1.10E-03		1.10E-04	
OD1-CS11	9.80E-03		9.80E-04	
OD1-CS12	6.50E-03		6.50E-04	
OD1-CS17	2.50E-02		2.50E-03	
OD1-CS18	2.70E-03		2.70E-04	
OD1-CS19	7.40E-02		7.40E-03	
OD1-CS20	2.70E-03		2.70E-04	
OD1-CS21	7.40E-02		7.40E-03	
OHX				
OHX-CS1	8.20E-03	1.20E-06	8.20E-02	2.37E-06
OHS-CS2	9.00E-04		9.00E-03	
OHX-CS6	1.60E-02		1.60E-01	
OHX-CS9	1.70E-02		1.70E-01	
OHX-CS10	1.90E-03		1.90E-02	
OHX-CS14	1.70E-02		1.70E-01	
OHX-CS18	4.90E-03		4.90E-02	
OHX-CS22	2.00E-02		2.00E-01	
OHX				
OHX-CS1	8.20E-03	1.20E-06	8.20E-04	1.09E-06
OHS-CS2	9.00E-04		9.00E-05	
OHX-CS6	1.60E-02		1.60E-03	
OHX-CS9	1.70E-02		1.70E-03	
OHX-CS10	1.90E-03		1.90E-04	
OHX-CS14	1.70E-02		1.70E-03	
OHX-CS18	4.90E-03		4.90E-04	
OHX-CS22	2.00E-02		2.00E-03	
OCST				
OCS-CS1	1.0	1.20E-06	1.0	2.37E-06
OCS-CS2	9.20E-03		9.20E-02	
OCS-CS10	1.90E-02		1.90E-01	
OCS-CS12	2.30E-02		2.30E-01	
OCS-CS18	4.60E-02		4.60E-01	
OCS-CS20	4.90E-02		4.90E-01	
OCST				
OCS-CS1	1.0	1.20E-06	1.0	1.09E-06
OCS-CS2	9.20E-03		9.20E-04	
OCS-CS10	1.90E-02		1.90E-03	
OCS-CS12	2.30E-02		2.30E-03	
OCS-CS18	4.60E-02		4.60E-03	
OCS-CS20	4.90E-02		4.90E-03	
ROP1 (LOSP Only)				
ROP-1	1.0	1.20E-06	5.09E-02	1.05E-06

significant initially, all of the core damage frequencies associated with the sensitivities are within the range of core damage frequencies reported in previous BWR probabilistic risk assessments. However, the increases in the core damage frequencies highlight the importance of maintaining good procedures and effective training.

Decreasing human error probabilities by an order of magnitude did not result in any significant decrease in plant risk. For each of the sensitivity analyses in which the human error was reduced by an order of magnitude, the core damage frequency was not less than 9.8E-7, virtually unchanged from the base IPE model quantification. The conclusion from these analyses is that improvements in procedures or training associated with these operator actions (OAD1, OHX, and OCST), without accompanying improvements in the hardware, would not result in any significant reduction in plant risk.

In the base IPE model it was assumed that, provided an onsite source of AC power was available, no credit would be taken for recovering offsite power. It can be concluded from the results of this sensitivity analysis that this was a reasonable assumption. The core damage frequency was relatively unchanged when credit was taken for the recovery of offsite power (from 1.2E-6 to 1.1E-6). It can be concluded, given that no other plant enhancements are considered, improvements in procedures and training associated with recovering offsite power in non-station blackout scenarios would not result in a significant decrease in plant risk.

4.5.4.2 Sensitivity Analyses on the Modified IPE Model

A sensitivity analysis was performed on the modified Quad Cities IPE model to estimate the impact on CDF of the recently modified procedures to operate the RCIC system manually during a long-term station blackout (SBO).

If the RCIC is operated successfully, more time is available to restore AC power from either offsite or onsite and to restore operability of the wide variety of systems available to provide makeup to the vessel. This time also allows the decay heat generation rate to decrease to a level that is within the capability of other low flow plant systems. If RCIC is not operated successfully in manual mode, the scenario remains analogous to the currently modeled plant response subsequent to the challenge of HPCI.

For the purposes of this sensitivity analysis, this scenario was modeled considering only the RCIC hardware and the operator action to operate RCIC manually; the analysis did not include recovery of power and systems available subsequent to power recovery. The human reliability analysis conducted for this operator action resulted in an HEP of 6.4E-4 for failure to operate the RCIC successfully. Detailed plant response tree modeling of manual operation of RCIC will be conducted during a Quad Cities PRA model update. It is recognized that the detailed models will include the effects of using additional DC power sources for long-term operation of relief valves to maintain the reactor vessel pressure in accordance with the Quad Cities EOPs.

The sensitivity analysis of this scenario indicated that the core damage frequency would be reduced to less than 1.4E-6/yr as a result of taking credit for this mode of RCIC operation. This would constitute a reduction in the current core damage frequency of approximately 35%.

4.5.5 Source Term Analysis

Any sequence of events that causes core damage may result in a release of radioactivity to the environment in excess of design-basis limits. Such radioactivity releases are possible whether or not the containment building remains intact, because no structure is perfectly leak-tight. The amount of radioactivity that may be released from the containment building if core damage occurs is sequence-dependent and strongly influenced by the size and complexity of the flow paths out of the building. Relatively large and unrestricted flow paths from the containment building are expected to exist if the automatic containment isolation function is impaired, or a containment structural component fails due to high pressure and temperature, or piping attached directly to the reactor vessel fails in an unisolable manner. Relatively small and restrictive flow paths from the containment building are expected to exist if the containment integrity is maintained because a plant's technical specifications allow only an extremely small amount of containment leakage. Thus, much smaller amounts of radioactivity would be released to surrounding buildings and the environment if containment integrity was maintained during a core-damage sequence than if containment was breached.

The large amounts of radioactivity that are associated with a severe accident occur because radioactive isotopes that are the by-product of the fission process accumulate in the fuel pellets. During core damage, the fuel pellets would overheat, chemically react with other reactor materials, and possibly melt and move outside of the original core geometry. As the fuel pellets dissociated, the highly radioactive fission products would be released from the fuel. Typical fission product isotopes include the Noble gases Xenon (Xe) and Krypton (Kr), as well as Cesium (Cs), Iodine (I), Tellurium (Te), Strontium (Sr). Cesium and Iodine predominantly react to form the relatively low boiling-point (or volatile) compound Cesium-Iodide (CsI). Tellurium and Strontium typically oxidize and form relatively higher boiling-point (or non-volatile) compounds (e.g., SrO, TeO_2). The amounts of radioactivity released from containment as these various isotopes constitute the so-called source term for an accident sequence.

The purpose of a source term analysis is to quantitatively estimate the masses of the various fission products that are released from the containment structure for the PRT end-states (or sequences) that result in core damage. Performing actual source term calculations for each sequence is an impossibility, however, given the large number of sequences defined by the PRTs. Thus, the scope of the source term analysis was limited to a consideration of the 100 highest-frequency sequences.

The scope of the source term analysis was decreased more as a result of a very detailed review of the progressions of the top 100 sequences. Each of the top 100 core damage sequences was traced through the PRTs to determine the disposition (i.e., success, failure, not asked) of every node in a sequence's complete PRT path. These nodal dispositions were then translated into a list of systems and equipment that would operate during each sequence. In this way, similarity of event progression was verified for each sequence within a PDS. The system and equipment operability lists for the PDSs were also compared. This comparison identified a number of PDSs with very similar combinations of functional failures, accident progressions, and fission product release paths after core damage. These similarities allowed a further reduction in the number of deterministic fission product release calculations needed to characterize the source term behavior of the top 100 sequences. The groupings of similar PDSs identified in this manner are listed in Table 4.5.5-1.

A total of fourteen unique sequence progressions were thus identified for source term analysis from the twenty-four PDSs containing the top 100 sequences. The specific event progression of the highest-frequency sequence in the highest-frequency PDS in each of these fourteen groupings of PDSs was chosen to be used as a basis for estimating each PDS group's source term characteristics. Deterministic fission product release calculations then were performed using a CECo-specific version of the MAAP code to simulate the fourteen unique sequences of events.

Table 4.5.5-2 lists the fourteen sequences analyzed deterministically. The sequences are identified in this table by their placement in the list of the top 100 core damage sequences, as well as by their PDS. The range of containment performance obtained from these detailed analyses includes:

- One sequence in which the combination of core spray injection into the failed RPV and torus cooling prevent torus venting and containment structural failure;
- One sequence in which the combination of core spray and CRD injection into the failed RPV and torus venting prevent containment structural failure;
- One sequence in which the combination of drywell spray and CRD injection into the failed RPV in combination with torus cooling and torus venting prevent containment structural failure;
- Two sequences in which CRD injection into the failed RPV and torus venting prevent containment structural failure;
- Three sequences in which torus venting cannot prevent drywell structural failure between 25 and 48 hours after the sequence start;
- Two sequences in which the torus is the probabilistically-determined failure site and torus failure occurs after 24 hours;
- Two ATWS sequences in which drywell and torus structural failures occur within the first hour of the sequences;
- One SBO sequence in which drywell structural failure is predicted to occur between 12 and 24 hours after the sequence start; and
- One SBO sequence in which drywell structural failure is predicted to occur within the first 12 hours of the sequence.

4.5.5.1 Sequence Source Terms

Key results of the mechanistic analysis for each sequence listed in Table 4.5.5-2 are summarized in Table 4.5.5-3. A capsule summary of each sequence is presented below. The following characteristics are common to all of the analyses reported.

- Normal containment leakage is modeled by a hole size equivalent to the leakage rate allowed by the plant technical specifications.
- The sequences are analyzed for 48 hours to assure that containment behavior and fission product transport behavior are well-developed and that identification of potential accident management insights is not limited by the 24-hour mission time. Thus, the equipment status at 24 hours is assumed to apply through the end of the calculation as appropriate, considering normal equipment limitations (e.g., loss of NPSH for a pump or depletion of a tank inventory). Also, fission product releases are reported after 48 hours of elapsed sequence time.
- No credit is taken for the possibility that fission product aerosols could accrete in the containment failure site and eventually plug the failure.
- No credit is taken for fission product retention in the reactor building.

TABLE 4.5.5-1

SEQUENCE PROGRESSION CHARACTERISTICS FOR PDSs IN QUAD IPE TOP 100 SEQUENCE LIST

CORE POWER	CORE DAMAGE TIME (HR)	RPV PRESS AT VESSEL FLR	CORE DEBRIS COOLING	CONTAINMENT HEAT REMOVAL	CNTMT VENT USED	CONTAINMENT STATUS	PDS
Decay Ht	>6	High	None	None	None	DWF hours after VF	BLAY
Decay Ht	>6	High	None	None	None	DWF hours after VF	BLAT
Decay Ht	<2	High	Core Spray	SPC	None	Intact	LEAB
Decay Ht	<2	High	Core Spray	SPC	None	Intact	TEAB
Decay Ht	>6	High	Core Spray	SPC	None	Intact	LLAB
Decay Ht	>6	High	Core Spray	SPC	None	Intact	TLAB
Decay Ht	<2	High	Core Spray	SPC	None	Intact	MEFB
Decay Ht	<2	High	Core Spray	SPC	None	Intact	MEFG
Decay Ht	<2	High	Core Spray	SPC	None	Intact	TEFB
ATWS (with SLC)	<2	High	Core Spray	SPC	None	Intact	TEFG
Decay Ht	>6	Low	None	None	Wetwell	Intact 24 hours; DWF later	LLBO
Decay Ht	<2	Low	None	None	Wetwell	Intact 24 hours; DWF later	IEBO
Decay Ht	>6	Low	None	None	Wetwell	Intact 24 hours; DWF later	LLCO
ATWS	<2	High	None	None	None	WWF before VF	TEEQ
Decay Ht	<2	High	Cntmt Spray (on CCST)	None	Wetwell	Intact 24 hours; DWF later	LEAC
Decay Ht	>6	High	Cntmt Spray (on CCST)	None	Wetwell	Intact 24 hours; DWF later	LLAC
Decay Ht	<2	High	LPCI	SPC	None	Intact after 24 hrs; DWF later	LEAH
Decay Ht	<2	High	LPCI/CRD	SPC	None	Intact after 24 hrs; DWF later	TEFH

TABLE 4.5.5-1

SEQUENCE PROGRESSION CHARACTERISTICS FOR PDSs IN QUAD IPE TOP 100 SEQUENCE LIST

CORE POWER	CORE DAMAGE TIME (HR)	RPV PRESS AT VESSEL FLR	CORE DEBRIS COOLING	CONTAINMENT HEAT REMOVAL	CNTMT VENT USED	CONTAINMENT STATUS	PDS
Decay Ht	>6	Low	Cntmt Spray	None	None	WWF hours after VF	LLCS
Decay Ht	>6	Low	Cntmt Spray	None	None	WWF hours after VF	MLCS
Decay Ht	<2	High	CRD	None	Wetwell	WWF hours after VF	TEFS
Decay Ht	<2	High	CRD	None	None	Intact	TEFE
Decay Ht	2-6	Low	CRD	None	None	WWF hours after VF	TIGS
Decay Ht	<2	High	None	None	None	DWF hours after VF	BEAY
Decay Ht	>6	Low	None	None	Wetwell	Intact	ALCE
ATWS	<2	High	None	None	None	DWF before VF	TEER
Decay Ht	<2	Low	Cntmt Spray	SPC	Wetwell	Intact	AEGG
Decay Ht	<2	Low	Cntmt Spray	SPC	Wetwell	Intact	LEBG
Decay Ht	>6	Low	None	None	None	WWF hours after VF	BLAS
Decay Ht	>6	Low	None	None	None	WWF hours after VF	TLBS

TABLE 4.5.5-2
SEQUENCES ANALYZED DETERMINISTICALLY TO CHARACTERIZE SOURCE TERMS FOR THE QUAD CITIES IPE TOP 100 SEQUENCES

SEQUENCE NO. FROM THE TOP 100 LIST	PDS	OTHER PDSs REPRESENTED BY THIS SEQUENCE
1	BLAY	BLAT
4	MEFG	LEAB, LLAB, MEFB, TEAB, TLAB, TEFB, TEFG
38	LLCO	LLBO, LEBO
10	TEEQ	None
2	BEAY	None
11	TEFE	None
-	TIGE	TIGS
-	LLAC	LEAC, LEAH, TEFH
17	TEER	None
31	AEGG	LEBG
29	ALCE	None
-	LLCS	MLCS, TLBS
-	LLAS	TEFS, BLAS

TABLE 4.5.5-3
QUAD CITIES NUCLEAR STATION
CONTAINMENT AND SOURCE TERM ANALYSIS DOMINANT SEQUENCES

SEQUENCE DESIGNATOR	BLAY	MEFG	LLCO	TEEQ	BEAY	TEFE	TIGE
Sequence No.	1	4	38	10	2	11	-
Source Term Bin	F	S	G	C	F	B	B
MAAP Run Database ID No.	Q930806	Q930304	Q930701	Q930903	Q930807	Q930128	Q930127
CORE/CONTAINMENT RESPONSE							
Time of Core Uncovery (hr)	9.75	0.79	9.02	0.68	0.77	0.73	2.07
Time of Core Relocation (hr)	11.2	1.58	10.4	1.12	1.55	1.81	3.41
Time of Vessel Failure (hr)	13.4	3.03	12.4	2.3	2.8	3.57	5.11
Time of Containment Failure (hr)	18.7	--	31	0.66	10.5	46.4	--
Time of Venting (hr)	--	--	12.4	--	--	4.57	0.49
Maximum Drywell Pressure (psig)	76.2	46.9	53.5	106	62.6	45.3	59.9
Maximum Drywell Temperature (°F)	777	326	861	798	871	641	620
Fraction of Clad Reacted in Vessel	0.1041	0.1002	0.1003	0.0882	0.0914	0.1064	0.0945
ENVIRONMENTAL RELEASE @ 48 Hr.							
Noble Release (%)	99.7	2.23	100	99.9	86.5	98.8	100
Volatile FP Release (%)	41	7E-03	28.2	2.3	54.2	0.2	0.15
Non-Volatile FP Release (%)	0.015	1E-05	0.28	9E-03	3E-03	2E-04	1E-03
Tellurium-based FP Release (%)	12.8	0	44.8	0.18	16.1	0.04	0.13

TABLE 4.5.5-3 (Continued)
QUAD CITIES NUCLEAR STATION
CONTAINMENT AND SOURCE TERM ANALYSIS DOMINANT SEQUENCES

SEQUENCE DESIGNATOR	LLAC	TEER	AEGG	ALCE	LLCS	LLAS
Sequence No.	-	17	31	29	-	-
Source Term Bin	D	F	A	B	B	B
MAAP Run Database ID No.	Q930702	Q930904	Q930405	Q930406	Q930705	Q930703
CORE/CONTAINMENT RESPONSE						
Time of Core Uncovery (hr)	10.4	0.67	0.02	0.02	9.02	10.4
Time of Core Relocation (hr)	11.9	1.06	0.45	24.7	10.4	11.9
Time of Vessel Failure (hr)	14	2.17	0.93	26.7	12.4	14
Time of Containment Failure (hr)	40	0.65	--	--	25.5	26
Time of Venting (hr)	14	--	23.3	20.1	--	--
Maximum Drywell Pressure (psig)	67.3	109	47.6	51.9	108	108
Maximum Drywell Temperature (°F)	711	805	603	573	917	752
Fraction of Clad Reacted in Vessel	0.1002	0.0821	0.0164	0.0412	0.1003	0.1002
ENVIRONMENTAL RELEASE @ 48 Hr.						
Noble Release (%)	98.2	99.9	61	99.9	100	99.6
Volatile FP Release (%)	4.7	57.9	0.011	0.12	0.26	0.11
Non-Volatile FP Release (%)	9E-05	0.06	1E-04	8E-04	4E-04	6E-05
Tellurium-based FP Release (%)	1.8	14	0.015	0.072	0.78	0.73

Sequence 1 - BLAY

Sequence Description

This accident scenario is initiated by a loss of off-site power and is followed by a failure of the diesel generators. The unit's loss of off-site power causes a reactor and turbine trip. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

HP1 - High Pressure Coolant Injection: Initiation of HPCI either manually or automatically at -59 inches. HPCI is assumed to operate until battery power is exhausted at 4 hours after the start of the sequence.

In addition, the following characteristics of this scenario are noteworthy.

- No other reactor vessel injection is available.
- No containment heat removal is available.
- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence includes HPCI isolation at 4 hr when the battery power is depleted. Following the loss of HPCI, reactor vessel pressure increases to the Target Rock valve safety mode setpoint. Intermittent operation of this safety valve increases the suppression pool water temperature but no suppression pool cooling can be initiated. Similarly, no systems are available to inject into the vessel. Thus, a sustained uncovered core state starts at 9.75 hr.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory boils off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 11.2 hr, continuing core degradation, and eventual reactor vessel failure at 13.4 hr. At that time, about 10% of the core Zircaloy inventory has oxidized.

When the reactor vessel fails, the release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. No containment heat removal is available, so the debris in containment heats up sufficiently to erode the concrete floor. The core debris and fission products remaining in the vessel, and the core debris in containment both radiatively heat the containment gas, while concrete erosion releases hot gases to the containment atmosphere. The containment pressure and drywell gas temperature increase until drywell shell failure is predicted at 18.7 hr.

Fission products are released from the containment building via the drywell failure. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

•	Noble Gases	99.7 %
•	Volatile Fission Products (Represented by CsI and RbI)	41.0 %
•	Non-Volatile Fission Products (Represented by SrO)	0.015 %
•	Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	12.8 %

Sequence 4 - MEFG

Sequence Description

This accident scenario is initiated by a medium LOCA. The energy release to the drywell quickly raises drywell pressure above 2.0 psig, which causes a reactor and turbine trip. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

RC - Reactivity Control: Control rods insert and scram the reactor.

OIADS - Operator inhibits ADS. The operator is assumed to inhibit automatic operation of the ADS, as directed by the emergency procedures.

OCNTS, CNTS - Operator action to initiate containment sprays; Proper operation of containment spray equipment. The operators are assumed to initiate containment (drywell only) sprays per the EOPs.

In addition, the following characteristics of this scenario are noteworthy:

- Feedwater is assumed to continue operating past the initiating event until the hotwell inventory is depleted.

- CRD hydraulic pumps are assumed to operate until their available CCST inventory is depleted.

- One train of Core Spray injection is assumed to initiate as designed and to continue injecting after vessel failure.

- One train of torus cooling is assumed to be initiated per the EOPs.

- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence includes initiation of drywell sprays at two minutes, initiation of torus cooling at ten minutes into the sequence, and feedwater operation for 0.48 hr until the hotwell inventory is depleted. Following the loss of feedwater injection, reactor vessel pressure decreases towards the Core Spray shut-off head, due to the continuing mass, energy flows out the LOCA site. However, the RPV pressure does not decrease quickly enough to allow significant Core Spray injection before a sustained uncovered core state starts at 0.79 hr.

Since sufficient vessel injection does not occur during this sequence prior to significant core relocation, the vessel inventory is depleted through the LOCA and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 1.58 hr, continuing core degradation, and eventual reactor vessel failure at 3.03 hr. At that time, about 10% of the core Zircaloy inventory has oxidized.

The continuous operation of the drywell sprays causes accumulation of a one-foot deep water pool on the drywell floor. At this pool depth, additional water overflows the bottom of the downcomer pipes to the torus. When the reactor vessel fails, the release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. The Core Spray pump, which was injecting into the vessel once the vessel pressure dropped below 373 psig at 2.79 hr, continues to inject into the failed vessel. The water injected into the vessel immediately flows out of the lower plenum failure site, and into the water pool on the drywell floor. The debris in containment is cooled by the continual water addition via the sprays and through the vessel, and the heated water that flows to the suppression pool is cooled by the continuing suppression pool cooling. However, per the EOPs, drywell sprays are secured at 10.5 hr due to high torus water level and this allows the core debris and deposited fission products remaining within the reactor vessel to slowly heat the drywell gas. The containment pressure and drywell gas temperature remain far below drywell failure limits through 48 hr.

Fission products are released from the containment building only via the assumed normal drywell leakage area . At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

• Noble Gases	2.23 %
• Volatile Fission Products (Represented by CsI and RbI)	7E-03 %
• Non-Volatile Fission Products (Represented by SrO)	1E-05 %
• Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	0.0 %

Sequence 38 - LLCO

Sequence Description

This accident scenario is initiated by a loss of off-site power, which causes a reactor and turbine trip. Since the feed pumps trip and the operators are assumed to not manually start high pressure injection, low RPV level is reached, MSIVs close, and the main condenser is isolated from the reactor. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

RC - Reactivity Control: Control rods insert and scram the reactor.

RVO, RVC - Relief Valves Open; Relief Valves Close: All relief valves open and close as dictated by the reactor vessel pressure.

HP1 - Automatic initiation of HPCI and operation in the level control mode. HPCI is assumed to be unused after reactor vessel pressure decreases below its turbine's low inlet pressure setpoint that causes automatic isolation.

OAD2, ADS - Operator action to depressurize per the HCL curve; proper operation of the ADS system in response to an actuation demand. The operator is assumed to depressurize the RPV to remain with the torus HCL to maintain the reactor vessel pressure low enough that low pressure ECCS pumps can inject. However, insufficient NPSH exists for operation of these pumps.

OVNT, LVW - Operator action to initiate containment venting; containment venting through the eight-inch wetwell vent. The operators are assumed to initiate wetwell venting through the eight-inch vent (hard-piped to the stack) per the emergency procedures.

In addition, the following characteristics of this scenario are noteworthy.

- The Feedwater, Safe Shutdown Makeup Pump and CRD Hydraulic systems are assumed unavailable.
- The seven drywell coolers are inoperable as a result of the initiator.
- Containment is successfully isolated.

<u>Sequence Quantification</u>

The pre-core damage phase of this sequence includes HPCI isolation at 4.4 hr due to low RPV pressure. Since no low pressure injection source is available during this sequence, the vessel inventory boils off, causing a re-pressurization of the RPV to the lowest relief valve setpoint. Manual depressurization of the RPV to maintain pressure within the Heat Capacity Limit (HCL) curve in the emergency procedures commences at 8.6 hr. This action hastens the occurrence of a sustained uncovered core state starting at 9.02 hr.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory continues to boil off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 10.4 hr, continuing core degradation, and eventual reactor vessel failure at 12.4 hr. At that time, about 10% of the core Zircaloy inventory has oxidized.

The release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. The operator initiates containment venting by opening the wetwell vent for the first time when Torus Bottom Pressure exceeds 46 psig at 12.4 hr. The operator is assumed to close this vent when Torus Bottom Pressure decreases below 36 psig, and then to cycle the vent as needed thereafter. Since there is no containment heat removal, drywell gas temperature continues to increase and heats the drywell shell sufficiently that drywell failure is predicted to occur at 31.0 hr.

Fission products are released from the containment building via both the wetwell vent and the drywell failure location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

•	Noble Gases	100 %
•	Volatile Fission Products (Represented by CsI and RbI)	28.2 %
•	Non-Volatile Fission Products (Represented by SrO)	0.28 %
•	Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	44.8 %

Sequence 10 - TEEQ

Sequence Description

This accident scenario is initiated by a turbine trip with the main condenser isolated, and a failure of the control rods to insert. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

AT1 - Anticipated Transient Without Scram System: Division 1 of the ATWS circuitry initiates signals to trip the recirc pumps.

RPT1 - Recirculation Pump Trip - Automatic: Recirc pump motor - generator field breakers open upon receipt of an ATWS signal.

FWA - Feedwater available: Reactor vessel level is controlled automatically using the feed pumps for as long as hotwell inventory is available.

RVO - Relief Valves Open: Sufficient numbers of relief and safety valves operate to limit RPV pressure below design limits.

WW/DW - Containment failure in the wetwell.

In addition, the following characteristics of this scenario are noteworthy:

- The main condenser hotwell inventory is initially 76,000 gallons, with a 900 gpm make-up flow.
- The CRD Hydraulic system is assumed unavailable.

- HPCI, Core Spray and RHR pumps are available and assumed to operate automatically. In addition, routine operator actions regarding use of the RHR system are credited. However, torus conditions change so rapidly during this sequence that loss of NPSH occurs for the low pressure pumps prior to any possible use of them.

- Drywell coolers are operable initially but assumed to fail seconds after the sequence start due to safety valve discharge into the drywell.

- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence includes a predicted average ATWS power level of about 56% full power. Maintaining normal water level with this power level depletes the hotwell inventory in about 9 minutes, at which time the feed pump trips. HPCI initiates 25 seconds later. Full HPCI flow is unable to sustain the vessel water level and power, so vessel level decreases to near the Top of Active Fuel; at this water level, HPCI flow is able to keep up with the core power of about 34% full power. The high core power associated with ATWS causes a rapid drywell and suppression pool heat-up. The HPCI pump trips at 39 minutes as a result of high turbine back pressure and a large wetwell failure occurs about one-half minute later. Since vessel injection is unavailable at this point and core power is significantly greater than decay power levels, a sustained uncovered core state starts at about 41 minutes.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory boils off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 1.12 hr, continuing core degradation, and eventual reactor vessel failure at 2.3 hr. At that time, about 9% of the core Zircaloy inventory has oxidized.

Fission products are released from the containment building via both the wetwell failure and the normal drywell leakage location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

• Noble Gases	99.9 %
• Volatile Fission Products (Represented by CsI and RbI)	2.3 %
• Non-Volatile Fission Products (Represented by SrO)	9E-03 %
• Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	0.18 %

Sequence 2 - BEAY

Sequence Description

This accident scenario is initiated by a loss of off-site power and is followed by a failure of the diesel generators. The unit's loss of off-site power causes a reactor and turbine trip. For this accident scenario, the following characteristics are noteworthy:

- No reactor vessel injection is available.
- No containment heat removal is available.
- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence is characterized by RPV pressure increasing to the lowest Electromatic relief valve setpoints. At 4 hr when the battery power is depleted, the Electromatic relief valves close and reactor vessel pressure increases further to the Target Rock valve safety mode setpoint. Intermittent operation of the Electromatic and Target Rock valves increases the suppression pool water temperature but no suppression pool cooling can be initiated. Similarly, no systems are available to inject into the vessel. Thus, a sustained uncovered core state starts at 0.77 hr.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory boils off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 1.55 hr, continuing core degradation, and eventual reactor vessel failure at 2.80 hr. At that time, about 9% of the core Zircaloy inventory has oxidized.

When the reactor vessel fails, the release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. No containment heat removal is available, so the debris in containment heats up sufficiently to erode the concrete floor. The core debris and fission products remaining in the vessel, and the core debris in containment both radiatively heat the containment gas, while concrete erosion releases hot gases to the containment atmosphere. The containment pressure and drywell gas temperature increase until drywell shell failure is predicted at 10.5 hr.

Fission products are released from the containment building via the drywell failure. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

•	Noble Gases	86.5 %
•	Volatile Fission Products (Represented by CsI and RbI)	54.2 %
•	Non-Volatile Fission Products (Represented by SrO)	0.003 %
•	Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	16.1 %

Sequence 11 - TEFE

Sequence Description

This accident scenario is initiated by a feedwater pump trip, which causes a reactor and turbine trip. The turbine bypass valves initially are used to control reactor pressure but, since the feed pumps trip and high pressure injection fails, low RPV level is quickly reached, MSIVs close, and the main condenser is isolated from the reactor. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

RC - Reactivity Control: Control rods insert and scram the reactor.

OVNT, LVW - Operator action to initiate containment venting; containment venting through the eight-inch wetwell vent. The operators are assumed to initiate wetwell venting through the eight-inch vent (hard-piped to the stack) per the emergency procedures.

In addition, the following characteristics of this scenario are noteworthy.

- The Safe Shutdown Makeup Pump is assumed unavailable.
- The seven drywell coolers operate until reactor vessel failure occurs.
- Containment is successfully isolated.

<u>Sequence Quantification</u>

The pre-core damage phase of this sequence includes main condenser isolation due to low RPV level at 10.7 sec. Since the operator action to depressurize the RPV fails during this sequence, the vessel inventory boil-off causes repressurization of the RPV to the lowest relief valve setpoint. Continuing CRD injection is inadequate to makeup the vessel boil-off and an uncovered core state is sustained from 0.73 hr.

Since recovery of sufficient vessel injection to provide adequate core cooling does not occur during this sequence, the vessel inventory continues to boil off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 1.8 hr, continuing core degradation, and eventual reactor vessel failure at 3.57 hr. At that time, about 11% of the core Zircaloy inventory has oxidized.

The release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. The operator initiates containment venting by opening the wetwell vent for the first time when Torus Bottom Pressure exceeds 46 psig at 4.57 hr. The operator is assumed to close this vent when Torus Bottom Pressure decreases below 36 psig, and then to cycle the vent as needed thereafter. Since there is no core debris cooling or containment heat removal, containment pressure and drywell shell temperature increase until drywell failure is predicted to occur at 46.4 hr.

Fission products are released from the containment building via both the wetwell vent and the drywell failure location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

•	Noble Gases	98.8 %
•	Volatile Fission Products (Represented by CsI and RbI)	0.20 %
•	Non-Volatile Fission Products (Represented by SrO)	2E-04 %
•	Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	0.04 %

Sequence - TIGE

Sequence Description

This accident scenario is initiated by a turbine trip with the main condenser isolated, and a failure of the control rods to insert. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

AT1 - Anticipated Transient Without Scram system: Division 1 of the ATWS circuitry initiates signals to trip the recirc pumps.

RPT1 - Recirculation Pump Trip - Automatic: Recirc pump motor-generator field breakers open upon receipt of an ATWS signal.

FWA - Feedwater available: Reactor vessel level is controlled automatically using the feed pumps for as long as hotwell inventory is available.

RVO - Relief Valves Open: Sufficient numbers of relief and safety valves operate to limit RPV pressure below design limits.

OSL1 - Operator action to initiate SLC before 110°F. The operator manually starts one SLC pump per the emergency procedures.

HP1 - Automatic operation of HPCI in the level control mode. HPCI is assumed to be unused after reactor vessel pressure decreases below its turbine's low inlet pressure setpoint that causes automatic isolation.

OAD2, ADS - Operator action to depressurize per the HCL curve; proper operation of the ADS system in response to an actuation demand. The operator is assumed to depressurize the RPV to remain within the torus HCL curve to maintain the RPV pressure low enough that low pressure ECCS pumps can inject. However, insufficient NPSH exists for operation of these pumps.

OVNT, LVW - Operator action to initiate containment venting; containment venting through the eight-inch wetwell vent. The operators are assumed to initiate wetwell venting through the eight-inch vent (hard-piped to the stack) per the EOPs.

In addition, the following characteristics of this scenario are noteworthy.

- The main condenser hotwell inventory is initially 76,000 gallons, with a 900 gpm make-up flow.
- The CRD Hydraulic system is assumed operable.
- Core Spray and RHR pumps are available and assumed to operate automatically. In addition, routine operator actions regarding use of these systems are credited. However, torus conditions change so rapidly during this sequence that loss of NPSH occurs for the low pressure pumps prior to any possible use of them.
- Drywell coolers are operable initially but assumed to fail seconds after the sequence starts due to safety valve discharge into the drywell.
- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence includes a predicted average ATWS power level of about 56% full power. This high core power causes a rapid drywell and suppression pool heat-up. Torus water temperature reaches 110°F in about 2 minutes, at which time one SLC pump is assumed to be initiated by the operators. At about 8 minutes after the sequence starts, the operators are assumed to start opening SRVs to maintain RPV pressure below the torus HCL curve. Maintaining normal water level with the ATWS power level depletes the hotwell inventory in about 9 minutes, at which time the feed pump trips. The high ATWS core power level maintained by HPCI injection causes the wetwell vent to be opened, per the EOPs, when torus bottom pressure reaches 46 psig at 29 minutes. The high power causes containment pressure to continue to increase, after the vent is opened, until the boron addition to the core region via the SLC pump injection is assumed to be sufficient to reduce core power to decay heat levels at about 32 minutes. This reduced power level combined with the opening of the SRVs to comply with the HCL causes HPCI pump trip at 39 minutes as a result of low RPV pressure. Since sufficient vessel injection for adequate core cooling is unavailable at this point, a sustained uncovered core state starts at about 2.07 hr.

Since recovery of sufficient vessel injection for adequate core cooling does not occur during this sequence, the vessel inventory boils off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 3.4 hr, continuing core degradation, and eventual reactor vessel failure at 5.1 hr. At that time, about 9% of the core Zircaloy inventory has oxidized.

When the reactor vessel fails, the release of core debris, steam and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. The core debris in the pedestal and drywell is initially cooled by entering the water pool on the drywell floor. The depth of this pool is limited to about one foot; at this height the water exceeds the elevation of the bottom of the downcomer pipes and overflows to the suppression pool. The core debris energy losses are large enough to saturate the drywell water pool and net water addition via CRD flow through the failed vessel. The steaming in the drywell requires periodic venting to control drywell pressure. Vessel injection by the CRD hydraulic system pumps continues to flow into the drywell until the CCST inventory available to these pumps is depleted at 29.7 hr. The drywell water pool is boiled off by the drywell/pedestal core debris decay heat over the next four hours, and shortly thereafter the core debris in the pedestal sump starts to erode concrete. Drywell pressure is controlled through the end of the sequence by periodic use of the wetwell vent, but drywell gas temperature steadily increases during this timeframe due to radiant heating by the ex-vessel core debris as well as non-condensible gas generation by the continuing core-concrete interaction. Although drywell conditions are deteriorating, containment structural failure is not predicted by the end of the sequence.

Fission products are released from the containment building via the wetwell vent and the normal drywell leakage location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

- Noble Gases 100 %
- Volatile Fission Products (Represented by CsI and RbI) 0.15 %
- Non-Volatile Fission Products (Represented by SrO) 1E-03 %
- Tellurium-based Fission Products (Represented by Te_2 and TeO_2) 0.13 %

Sequence - LLAC

Sequence Description

This accident scenario is initiated by a loss of off-site power which causes a reactor and turbine trip. Since the feed pumps trip and the operators are assumed to not manually start high pressure injection, low RPV level is reached, MSIVs close, and the main condenser is isolated from the reactor. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

RC - Reactivity Control: Control rods insert and scram the reactor.

RVO, RVC - Relief Valves Open, Relief Valves Close: All relief valves open and close as dictated by the reactor vessel pressure.

HP1 - Automatic initiation of HPCI and operation in the level control mode. HPCI is assumed to be unused after reactor vessel pressure decreases below its turbine's low inlet pressure setpoint that causes automatic isolation.

OCST, CST - Operator action to align low pressure pump suction to CCST; proper operation of CCST suction alignment hardware. The operators are assumed to align containment spray suction to the CCST.

OCNTS, CNTS - Operator action to initiate containment sprays; proper operation of containment spray equipment. The operators are assumed to initiate containment (drywell only) sprays per the EOPs.

OVNT, LVW - Operator action to initiate containment venting; containment venting through the eight-inch wetwell vent. The operators are assumed to initiate wetwell venting through the eight-inch vent (hard-piped to the stack) per the EOPs.

In addition, the following characteristics of this scenario are noteworthy:

- The Feedwater, Safe Shutdown Makeup Pump and CRD Hydraulic systems are assumed unavailable.
- The seven drywell coolers are inoperable as a result of the initiator.
- Containment is successfully isolated.

<u>Sequence Quantification</u>

The pre-core damage phase of this sequence includes HPCI isolation at 4.4 hr due to low RPV pressure. Since low pressure injection is unavailable during this sequence, the vessel inventory boils off, causing a repressurization of the RPV to the lowest relief valve setpoint. An uncovered core state is sustained from 10.4 hr.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory continues to boil off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 11.9 hr, continuing core degradation, and eventual reactor vessel failure at 14.0 hr. At that time, about 10% of the core Zircaloy inventory has oxidized.

The release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. The operator immediately initiates drywell sprays with pump suction aligned to the CCST and then he opens the wetwell vent for the first time when Torus Bottom Pressure exceeds 46 psig a few minutes later. The operator is assumed to close this vent when Torus Bottom Pressure decreases below 36 psig, and then to cycle the vent as needed thereafter. Drywell sprays are secured at 14.3 hr when the CCST inventory is depleted. Since there is no further core debris cooling or containment heat removal, drywell gas temperature and pressure eventually increase until drywell failure is predicted to occur at 40.0 hr.

Fission products are released from the containment building via both the wetwell vent and the drywell failure location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

•	Noble Gases	98.2 %
•	Volatile Fission Products (Represented by CsI and RbI)	4.66 %
•	Non-Volatile Fission Products (Represented by SrO)	9E-5 %
•	Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	1.8 %

Sequence 17 - TEER

Sequence Description

This accident scenario is initiated by a turbine trip with the main condenser isolated, and a failure of the control rods to insert. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

AT1 - Anticipated Transient Without Scram system: Division 1 of the ATWS circuitry initiates signals to trip the recirc pumps.

RPT1 - Recirculation Pump Trip - Automatic: Recirc pump motor-generator field breakers open upon receipt of an ATWS signal.

FWA - Feedwater available: Reactor vessel level is controlled automatically using the feed pumps for as long as hotwell inventory is available.

RVO - Relief Valves Open: Sufficient numbers of relief and safety valves operate to limit RPV pressure below design limits.

In addition, the following characteristics of this scenario are noteworthy.

- The main condenser hotwell inventory is initially 76,000 gallons, with a 900 gpm make-up flow.

- The CRD Hydraulic system is assumed unavailable.

- HPCI, Core Spray and RHR pumps are available and assumed to operate automatically. In addition, routine operator actions regarding use of these systems are credited. However, torus conditions change so rapidly during this sequence that loss of NPSH occurs for the low pressure pumps prior to any possible use of them.

- Drywell coolers are operable initially but assumed to fail seconds after the sequence starts due to safety valve discharge into the drywell.

- Containment is successfully isolated.

- Containment failure occurs in the drywell.

Sequence Quantification

The pre-core damage phase of this sequence includes a predicted average ATWS power level of about 56% full power. Maintaining normal water level with this power level depletes the hotwell inventory in about 9 minutes, at which time the feed pump trips. The high core power associated with ATWS causes a rapid drywell and suppression pool heat-up. The HPCI pump trips at 39 minutes as a result of high turbine backpressure, and a large drywell failure occurs about 10 seconds later. Since vessel injection is unavailable at this point and core power is significantly greater than decay power levels, a sustained uncovered core state starts at about 40 minutes.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory boils off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 1.06 hr, continuing core degradation, and eventual reactor vessel failure at 2.2 hr. At that time, about 8% of the core Zircaloy inventory has oxidized.

Fission products are released from the containment building via the drywell failure location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

- Noble Gases 99.9 %
- Volatile Fission Products (Represented by CsI and RbI) 57.9 %
- Non-Volatile Fission Products (Represented by SrO) 0.06 %
- Tellurium-based Fission Products (Represented by Te_2 and TeO_2) 14.0 %

Sequence 31 - AEGG

Sequence Description

This accident scenario is initiated by a large (3.0 ft^2) suction-side recirculation piping break. The large mass and energy flows out the break cause a rapid reactor vessel depressurization, which initiates a reactor and turbine trip. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

RC - Reactivity Control: Control rods insert and scram the reactor.

OCNTS, CNTS - Operator action to initiate containment sprays; proper operation of containment spray equipment. The operators are assumed to initiate drywell sprays per the EOPs.

OHX, RHRHX - Operator action to align RHR heat exchangers for use with drywell sprays; proper operation of RHR heat exchangers and associated equipment.

In addition, the following characteristics of this scenario are noteworthy:

- The Feedwater, HPCI, RCIC and Safe Shutdown Makeup pumps are assumed unavailable.
- The CRD hydraulic system is assumed operable.
- The seven drywell coolers are inoperable as a result of the initiator.
- The operator is assumed to initiate torus cooling per the EOPs.
- The operator is assumed to initiate wetwell venting in accordance with the EOPs.
- Containment is successfully isolated.

<u>Sequence Quantification</u>

The pre-core damage phase of this sequence includes drywell spray initiation shortly after the postulated pipe rupture. The mass flow out of the break causes a sustained uncovered core state to start at 64 seconds.

Since sufficient vessel injection does not occur during this sequence, the bulk of the vessel inventory is rapidly depleted through the LOCA site. The remaining inventory boils off and the core overheats. Suppression pool cooling is assumed to start at ten minutes. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 0.45 hr, continuing core degradation, and eventual reactor vessel failure at 0.93 hr. At that time, about 2% of the Zircaloy inventory has oxidized.

When the reactor vessel fails, the release of core debris, steam and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. The core debris in the pedestal and drywell is initially cooled by entering the water pool on the drywell floor, and its decay heat is subsequently removed by the continual water addition from the drywell sprays and CRD injection to the overlying pool. The depth of this pool is limited to about one foot; at this height the water exceeds the elevation of the bottom of the downcomer pipes and overflows to the suppression pool. The heated water that flows to the torus is cooled by the continuing suppression pool cooling. However, per the EOPs, drywell sprays are secured at 15.9 hr due to high torus water level and this allows the core debris energy to saturate the drywell water pool and net water addition via CRD flow through the failed vessel. The ensuing steaming in the drywell causes the pressure and gas temperature to increase until the wetwell vent is first opened at 23.3 hr. Vessel injection by the CRD hydraulic system pumps continues to flow into the drywell until the CCST inventory available to these pumps is depleted at 25.8 hr. The drywell water pool is boiled off by the drywell/pedestal core debris decay heat over the next five hours, and shortly thereafter the core debris in the pedestal sump starts to erode concrete. Drywell pressure is controlled through the end of the sequence by periodic use of the wetwell vent, but drywell gas temperature steadily increases during this timeframe due to radiant heating by the ex-vessel core debris as well as non-condensible gas generation by the continuing core-concrete interaction. Although drywell conditions are deteriorating, containment structural failure is not predicted by the end of the sequence.

Fission products are released from the containment only via wetwell vent and the assumed normal leakage area. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

- Noble Gases 61%
- Volatile Fission Products (Represented by CsI and RbI) 0.011%
- Non-Volatile Fission Products (Represented by SrO) 1E-04%
- Tellurium-based Fission Products (Represented by Te_2 and TeO_2) 0.015%

Sequence 29 - ALCE

Sequence Description

This accident scenario is initiated by a large (3.0 ft^2) suction-side recirculation piping break. The large mass and energy flows out the break cause a rapid reactor vessel depressurization, which initiates a reactor and turbine trip. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

CS - Automatic initiation of one train of the Core Spray system.

OVNT, LVW - Operator action to initiate containment venting; containment venting through the eight-inch wetwell vent. The operators are assumed to initiate wetwell venting through the eight-inch vent (hard-piped to the stack) per the emergency procedures.

In addition, the following characteristics of this scenario are noteworthy:

- The Feedwater, HPCI, RCIC and Safe Shutdown Makeup pumps are assumed unavailable.

- The CRD hydraulic system is assumed operable.

- The seven drywell coolers are inoperable as a result of the initiator.

- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence includes core spray initiation shortly after the postulated pipe rupture and core spray flow starts at 49.1 seconds. The mass flow out of the break causes a sustained uncovered core state to start at 64.5 seconds, however, core spray flow provides sufficient cooling to prevent core overheating. The continuous core spray cooling lasts until high suppression pool temperature combined with periodic torus vent operation (starting at 20.1 hr) causes a loss of NPSH condition for the core spray pumps at 23.4 hrs.

After core spray pump failure, RPV injection into the lower head by the CRD pumps continues, but is ineffective for core cooling due to the assumption that this flow leaks out the jet pumps' slip joints into the downcomer region. Since other sources of sufficient vessel injection are not available during the remainder of this sequence, the bulk of the vessel inventory is rapidly depleted through the LOCA site. The remaining inventory boils off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 24.7 hr. CRD injection ends as 25.9 hr due to depletion of the CCST inventory. The reactor vessel eventually fails at 26.7 hr. At that time, about 4% of the Zircaloy inventory has oxidized.

When the reactor vessel fails, the release of core debris, steam and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. The core debris in the pedestal and drywell is initially cooled by entering the water pool on the drywell floor, and its decay heat is subsequently removed for several hours as the overlying pool is heated up and boiled off. The depth of this pool was initially at its limit of about one foot; at this height the water exceeds the elevation of the bottom of the downcomer pipes and overflows to the suppression pool. Shortly after the drywell pool is depleted, the core debris in the pedestal sump heats up sufficiently to start to erode concrete. Drywell pressure is controlled through the end of the sequence by periodic use of the wetwell vent, but drywell gas temperature steadily increases during this timeframe due to radiant heating by the ex-vessel core debris as well as non-condensible gas generation by the continuing core-concrete interaction. Although drywell conditions are deteriorating, containment structural failure is not predicted by the end of the sequence.

Fission products are released from the containment only via wetwell vent and the assumed normal leakage area. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

• Noble Gases	99.9%
• Volatile Fission Products (Represented by CsI and RbI)	0.12%
• Non-Volatile Fission Products (Represented by SrO)	8E-04%
• Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	0.072%

Sequence - LLCS

Sequence Description

This accident scenario is initiated by a loss of off-site power, which causes a reactor and turbine trip. Since the feed pumps trip and the operators are assumed to not manually start high pressure injection, low RPV level is reached, MSIVs close, and the main condenser is isolated from the reactor. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

RC - Reactivity Control: Control rods insert and scram the reactor.

RVO, RVC - Relief Valves Open, Relief Valves Close: All relief valves open and close as dictated by the reactor vessel pressure.

HP1 - Automatic initiation of HPCI and operation in the level control mode. HPCI is assumed to be unused after reactor vessel pressure decreases below its turbine's low inlet pressure setpoint that causes automatic isolation.

OAD2, ADS - Operator action to depressurize per the HCL curve; proper operation of the ADS system in response to an actuation demand. The operator is assumed to depressurize the RPV to remain with the torus HCL to maintain the reactor vessel pressure low enough that low pressure ECCS pumps can inject. However, insufficient NPSH exists for operation of these pumps.

In addition, the following characteristics of this scenario are noteworthy.

- The Feedwater, Safe Shutdown Makeup Pump and CRD Hydraulic systems are assumed unavailable.
- The seven drywell coolers are inoperable as a result of the initiator.
- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence includes HPCI isolation at 4.4 hr due to low RPV pressure. Since no low pressure injection source is available during this sequence, the vessel inventory boils off, causing a re-pressurization of the RPV to the lowest relief valve setpoint. Manual depressurization of the RPV to maintain pressure within the Heat Capacity Limit (HCL) curve in the emergency procedures commences at 8.6 hr. This action hastens the occurrence of a sustained uncovered core state starting at 9.02 hr.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory continues to boil off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at

10.4 hr, continuing core degradation, and eventual reactor vessel failure at 12.4 hr. At that time, about 10% of the core Zircaloy inventory has oxidized.

The release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. Since there is no containment heat removal, or core debris cooling, containment pressure increases and torus failure is predicted to occur at 25.5 hr.

Fission products are released from the containment building via both the wetwell vent and the drywell failure location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

•	Noble Gases	100 %
•	Volatile Fission Products (Represented by CsI and RbI)	0.26 %
•	Non-Volatile Fission Products (Represented by SrO)	4E-04 %
•	Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	0.78 %

Sequence - LLAS

Sequence Description

This accident scenario is initiated by a loss of off-site power, which causes a reactor and turbine trip. Since the feed pumps trip and the operators are assumed to not manually start high pressure injection, low RPV level is reached, MSIVs close, and the main condenser is isolated from the reactor. For this accident scenario, the following equipment, systems, and operating instructions were assumed available:

RC - Reactivity Control: Control rods insert and scram the reactor.

RVO, RVC - Relief Valves Open, Relief Valves Close: All relief valves open and close as dictated by the reactor vessel pressure.

HP1 - Automatic initiation of HPCI and operation in the level control mode. HPCI is assumed to be unused after reactor vessel pressure decreases below its turbine's low inlet pressure setpoint that causes automatic isolation.

OCST, CST - Operator action to align low pressure pump suction to the CCST; proper operation of CCST suction alignment hardware. The operators are assumed to align containment spray suction to the CCST.

OCNTS, CNTS - Operator action to initiate containment sprays; proper operation of containment spray equipment. The operators are assumed to initiate containment (drywell only) sprays per the EOPs.

WW/DW - Containment failure in the wetwell.

In addition, the following characteristics of this scenario are noteworthy.

- The Feedwater, Safe Shutdown Makeup Pump and CRD Hydraulic systems are assumed unavailable.

- The seven drywell coolers are inoperable as a result of the initiator.

- Containment is successfully isolated.

Sequence Quantification

The pre-core damage phase of this sequence includes HPCI isolation due to low RPV pressure at 4.4 hr. Since no low pressure injection source is available during this sequence, the vessel inventory boils off, causing repressurization of the RPV to the lowest relief valve setpoint. An uncovered core state is sustained from 10.4 hr.

Since recovery of vessel injection does not occur during this sequence, the vessel inventory continues to boil off and the core overheats. The core damage phase of this sequence is characterized by core debris relocation within the original core geometry at 11.9 hr, continuing core degradation, and eventual reactor vessel failure at 14.0 hr. At that time, about 10% of the core Zircaloy inventory has oxidized.

The release of core debris, steam, and water to the pedestal results in an increased energy load on containment, and increases the rates of change of pressure and gas temperature. In response to the worsening drywell conditions the operator initiates drywell sprays with pump suction aligned to the CCST several minutes after vessel failure. Drywell sprays are secured at 14.3 hr when the CCST inventory is depleted. Since there is no further core debris cooling or containment heat removal, containment pressure increases until torus failure is predicted to occur at 26.0 hr.

Fission products are released from the containment building via both the wetwell failure location and the assumed normal drywell leakage location. At 48 hours after the sequence begins, the following fission product mass fractions are calculated to be released from the containment:

• Noble Gases	99.6 %
• Volatile Fission Products (Represented by CsI and RbI)	0.11 %
• Non-Volatile Fission Products (Represented by SrO)	6E-05 %
• Tellurium-based Fission Products (Represented by Te_2 and TeO_2)	0.73 %

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4.5.6 Source Term Sensitivity Analyses

Due to the complexity of a nuclear plant, the models developed to perform a Level 2 analysis contain simplified representations of operator actions, plant and equipment behavior, and severe accident phenomena. These simplifications, while necessary, all introduce uncertainty into the base-line characterization of a plant. To describe the uncertainty associated with the source term calculations, a number of additional MAAP calculations are performed. These sensitivity calculations are based on the MAAP runs described in Section 4.5.5, but each differs from its base case by having one key sequence attribute changed in a bounding way. For example, the AEGG sequence described above assumed operation of the torus vent per the EOPs, and this resulted in a large Noble gas release at 23.3 hr when the vent was opened. A variation on this sequence would be to delay or forego venting altogether, to determine how much additional time might be available for recovery of containment heat removal. In this way, a set of such source term sensitivity analyses can identify those plant and/or sequence attributes that have the largest effect on the likelihood or timing of containment failure and the magnitude of the source term without explicitly calculating individual and/or combined uncertainties.

To obtain the most insight from a sensitivity analysis, sequence attributes that have the largest impact on source term results must be varied. The following information sources were used to identify potential sensitivity calculations:

- Table A.5 in NUREG-1335, which lists a number of key parameters for a sensitivity study identified by the NRC,
- EPRI report EPRI-TR-100167 (Draft - To Be Published 1992), "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," and
- IPE analyst insights pertaining to equipment operation, operator actions, and plant modeling assumptions.

Tables 4.5.6-1 and 4.5.6-2 provide lists of the sensitivity calculations recommended by NUREG-1335 and the EPRI report, respectively, and indicate how these recommendations are dispositioned in the IPE. Table 4.5.6-3 provides a list of the

TABLE 4.5.6-1
PARAMETERS RECOMMENDED FOR SENSITIVITY STUDY BY NUREG-1335

ISSUE	TREATMENT IN IPE	DISCUSSION
Performance of CHR[1] systems during core melt accidents	Realistic performance of CHR systems quantified in PRTs[2]	Use of DWC[3] past LOCA initiator, SV[4] discharge, or RPV[5] failure not considered due to difficulty in determining degraded heat transfer performance.
Hydrogen production and combustion	Combustion treated in PES[6]. Production treated in CECo-MAAP runs Q930806A, Q930806B, and Q930806C.	PES concludes hydrogen combustion is not likely to cause early DW failure, however, long-term DW failure possible if inappropriate recovery action is taken. See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.
Induced failure of the RCS	Not considered a BWR issue.	NC[7] flow via SGs[8] allow hot core gases to heat surge line faster than if no NC flow. BWRs have no comparable NC flow path.
Core relocation Characteristics	Treated in CECo-MAAP runs Q930806A, Q930806B, and Q930806C.	See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.
Mode of RPV failure	Encompassed by the source term analysis performed.	MAAP assumes the RPV fails at a lower head penetration when the core debris temperature is above the steel melting point; the hole then ablates by debris flow through it. An alternative view is that core debris in the lower head is cooled until the water above is boiled away; the debris then heats up and causes the entire lower head to fall off. This alternative typically delays RPV failure by about an hour but cannot be modeled using MAAP. However, fission product releases from containment are expected to be lower for the later RPV failure mode because fission products released from unrelocated core debris have a longer time interval to be swept to the suppression pool and retained there.
Fuel/coolant interactions	Treated in steam explosion PES.	PES concludes steam explosions are no threat to RPV or containment and promote debris dispersal and cooling.
DCH[9]	Treated in DCH PES.	PES concludes DCH would cause negligible containment pressurization and would not cause early DW failure.
Potential for early CF[10] due to pressure load	Treated in CECo-MAAP run Q930806B.	See Table 4.5.6-3 for CECo-MAAP run definition. See Table 4.5.6-4 for CECo-MAAP run results.
Long-term disposition of core debris	Treated in MCCI[11], LMT[12], and containment over-pressurization PES.	The MCCI and LMT PES's each discuss a possible long-term debris configuration and evaluate possible outcomes (see table entries below). Since the actual ex-vessel debris distribution will be highly uncertain, the over-pressure PES provides a CF curve that reduces DW shell strength as its temperature increases. All IPE CECo-MAAP runs used this temperature-dependent failure curve.
Potential for early CF due to direct contact by core debris	Treated in LMT PES and CECo-MAAP runs Q930701A, Q930701B, Q930701C, and Q930701D.	PES concludes that core debris in contact with the DW liner is unlikely to melt through if there is water on the DW floor, but more likely if the DW floor is dry. See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.
Long-term core-concrete interactions	Treated in MCCI PES and CECo-MAAP runs Q930304A and Q930806D.	PES concludes CF by pressure and/or temperature would occur before basemat penetration. CECo-MAAP runs consider debris coolability uncertainty. See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.

NOTES:

1. CHR = Containment heat removal
2. PRTS = Plant response trees
3. DWC = Drywell coolers
4. SV = Safety Valve
5. RPV = Reactor pressure vessel
6. PES = Phenomenological Evaluation Summary
7. NC = Natural Circulation
8. SG = Steam Generator
9. DCH = Direct containment heating
10. CF = Containment failure
11. MCCI = Molten core-concrete interaction
12. LMT = Liner melt-through

TABLE 4.5.6-2
MAAP MODEL PARAMETERS RECOMMENDED FOR SENSITIVITY STUDY BY EPRI-TR-100167 (DRAFT)

VARIABLE	TREATMENT IN IPE	DISCUSSION
FMAXCP	Encompassed by the source term analysis performed.	This variable sets when the core debris mass remaining in the original core nodes is "dumped" into the lower head. EPRI report recommends changing this variable's value from 0.1 to 0.8 for one sequence in which long-term revaporization dominates fission product releases. Level 2 CECo-MAAP runs and sensitivity cases show a variety of fission product release behavior, without explicitly changing this variable. For example, fission product releases for CECo-MAAP run Q930701 are dominated by long-term revaporization while CECo-MAAP run Q930701A has significant early and late fission product releases. See Section 4.5.5 for CECo-MAAP run Q930701 (LLCO) definition and Table 4.5.6-3 for CECo-MAAP run Q930701A definition. See Tables 4.5.5-3 and 4.5.6-4 for CECo-MAAP run results.
FCHF	Treated in CECo-MAAP runs Q930304A and Q930806D.	This variable is the coefficient used in the MAAP critical heat flux formula for debris coolability calculations. EPRI report says to change this variable value from 0.1 to 0.02, and this was done for the CECo-MAAP runs indicated. See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.
DKPLUG	Treated in all Level 2 CECo-MAAP runs.	This variable is the coefficient for the Morewitz model for plugging restricted flow paths by accretion of aerosol particles passing through them. EPRI report says to set this variable to a large value if the assumed containment failure flow path is assumed to be greater than 1 cm in height. All Level 2 IPE CECo-MAAP runs assumed large enough aerosol flow paths that plugging would not be effective.
FCRBLK	Treated in CECo-MAAP runs Q930806A, Q930806B, and Q930806C.	This variable selects the core "blockage" model assumed for a MAAP run. A value of 1 corresponds to the IDCOR assumption of channel-wide blockage when any node becomes molten, which terminates Zr oxidation in the affected channel; a value of 0 (default) causes only molten nodes to block and allows Zr oxidation within an affected channel at higher elevations; a value of -1 allows unimpeded Zr oxidation. EPRI report says to change this variable from 0 to 1 for a station blackout sequence. A run corresponding to each permissible value was performed because this parameter's value was changed during all other Level 2 CECo-MAAP runs from 0 to 1 when at least 4 core nodes had more than 120% of their original UO_2 mass. See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.
ACVENT, ADWLEK	Treated in CECo-MAAP runs Q930701A, Q930701B, Q930701C, Q930701D, and Q930704A, Q930704B, Q930704C, Q930704D.	These variables represent the areas of the flow paths from the wetwell and drywell, respectively. EPRI report recommends considering both wetwell and drywell failure locations and various sizes. Failure location was considered in PRTs and included as appropriate in CECo-MAAP runs. Sensitivities of source term magnitudes to failure size/location were further considered by performing the CECo-MAAP runs indicated. See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.
PCFAIL	Treated in all Level 2 CECo-MAAP runs .	This variable represents containment failure pressure. EPRI report recommends including effects of temperature on shell strength and this was done for all Level 2 IPE CECo-MAAP runs. EPRI report also says to model effects of a lower failure pressure in one dry sequence. This was effectively done by the early drywell failure of CECo-MAAP run Q930806B. See Table 4.5.6-3 for CECo-MAAP run definitions. See Table 4.5.6-4 for CECo-MAAP run results.
ADWF	Treated in molten core-concrete interaction Phenomenological Evaluation Summary (PES),	This variable represents drywell floor area. EPRI report says to investigate effect of reduced core debris spreading by using 1/4 of default value for one dry sequence. PES concludes that use of 1/4 of default value is justified only for a wet sequence. Analysis in PES also shows that reduced spreading in a dry sequence is unlikely to result in basemat failure prior to other potential shell failure mechanisms.

TABLE 4.5.6-3
DEFINITION OF LEVEL 2 SENSITIVITY CASES

SENSITIVITY CASE CECo-MAAP RUN DATABASE ID NO.	BASE CASE SEQUENCE DESIGNATOR	DIFFERENCE FROM BASE CASE SEQUENCE	ISSUE(S) ADDRESSED BY SENSITIVITY CASE
Q930701A	LLCO	1-inch diameter hole opened in drywell shell 2 minutes after reactor vessel failure	Potential for early drywell failure due to direct contact by core debris
Q930701B	LLCO	20 square inch hole opened in drywell shell 2 minutes after reactor vessel failure	Potential for early drywell failure due to direct contact by core debris
Q930701C	LLCO	1-inch diameter hole opened in drywell shell 30 minutes after reactor vessel failure	Potential for early drywell failure due to direct contact by core debris
Q930701D	LLCO	20 square inch hole opened in drywell shell 30 minutes after reactor vessel failure	Potential for early drywell failure due to direct contact by core debris
Q930304A	MEFG	Parameter FCHF changed from default value of 0.10 to 0.02	Effectiveness of core debris cooling and its impact on core-concrete interactions
Q930806A	BLAY	Parameter FCRBLK not changed from default of 0 to +1 when 4 core nodes have accumulated > 120% of initial UO_2 mass	Hydrogen production, core relocation characteristics
Q930806B	BLAY	Parameter FCRBLK initially set to -1 and not changed to +1 when 4 core nodes have accumulated > 120% of initial UO_2 mass	Hydrogen production, core relocation characteristics, and potential for early containment failure due to pressure load
Q930806C	BLAY	Parameter FCRBLK initially set to +1	Hydrogen production, core relocation characteristics
Q930806D	BLAY	Parameter FCHF changed from default value of 0.10 to 0.02	Effectiveness of core debris cooling and its impact on long-term core-concrete interactions
Q930704A	LLAX	No wetwell venting and only wetwell failure allowed	Effect of containment failure location and size
Q930704B	LLAX	No wetwell venting and only large wetwell failure allowed	Effect of containment failure location and size
Q930704C	LLAX	No wetwell venting and only drywell failure allowed	Effect of containment failure location and size
Q930704D	LLAX	No wetwell venting and only large drywell failure allowed	Effect of containment failure location and size

sensitivity cases analyzed using the CECo-specific version of MAAP in the Quad Cities IPE. Table 4.5.6-4 provides a summary of numerical results for the Level 2 sensitivity cases. Table 4.5.6-5 provides an overview of the key results for these cases. A brief discussion of these results and quantitative estimates of the possible range of key event times and fission product releases are provided in the following subsections.

4.5.6.1 RPV Failure Timing

The influence of the core melt progression model on timing of RPV failure has been investigated by the sensitivity analyses and is discussed below.

MAAP allows the following three options in its core melt progression model.

- Local blockage, local node cut-off (FCRBLK = 0): steam flow is prohibited through but not above molten core nodes. Hydrogen cannot be produced in molten nodes with this option, but can be produced in any other unmolten core node. EPRI recommends use of this option as the default.

- Channel blockage (FCRBLK = 1): steam flow is prohibited through and above any molten core node. Relative to the default model option, this option results in an uncoolable geometry and a faster core melt; it also produces less hydrogen as the core degrades because Zircaloy oxidation cannot occur in core regions where there is no steam flow.

- No local blockage, no local node cut-off (FCRBLK = -1): steam flow is allowed through and above all core nodes at all times. Relative to the default model option, this option results in a coolable geometry and a faster core melt; it also produces much more hydrogen as the core degrades because Zircaloy oxidation occurs wherever there is steam flow in the core.

All base case Level 2 CECo-MAAP runs use the default model option until four core nodes have accumulated more than an additional 20% of their initial fuel mass. At this time, the channel blockage model is assumed, which forces an uncoolable core geometry and thereby prevents the possibility of unrealistically optimistic predictions of core recovery in vessel. The impact of this modeling approach was investigated by running cases Q930806A,B,C in which each of the core melt progression model options was chosen at the beginning of the sequence and not changed throughout. The effect of the core melt progression model assumption is a range of 2.1 hr (12.5 hr to 14.6 hr) in vessel failure time. A larger spread in vessel failure times would be expected for a sequence which reached core damage later due to the lower decay power, while a smaller spread in vessel failure times would be expected for a sequence which reached core damage earlier.

TABLE 4.5.6-4
QUAD CITIES NUCLEAR STATION
CONTAINMENT AND SOURCE TERM SENSITIVITY ANALYSIS

KEY SEQUENCE CHARACTERISTIC	BASE CASE	WW FAILURE	LARGE WW FAILURE	DW FAILURE	LARGE DW FAILURE
Sequence MAAP Run Database ID No.	Q930704	Q930704A	Q930704B	Q930704C	Q930704D
Sequence Designator	LLAX	--	--	--	--
CORE/CONTAINMENT RESPONSE					
Time of Core Uncovery (hr)	10.4	10.4	10.4	10.4	10.4
Time of Core Relocation (hr)	11.9	11.9	11.9	11.9	11.9
Time of Vessel Failure (hr)	14	14	14	14	14
Time of Containment Failure (hr)	37.6	26.1	26.1	19.3	19.3
Time of Venting (hr)	14	---	---	---	---
Maximum Drywell Pressure (psig)	69.8	107.9	107.9	78.1	78.1
Maximum Drywell Temperature (°F)	709	812	740	744	707
Fraction of Clad Reacted in Vessel	0.1001	0.1001	0.1001	0.1001	0.1001
ENVIRONMENTAL RELEASE @ 24 hr/48 hr					
Noble Release (%)	73.4/99.4	0.07/95.3	0.07/99.8	15.1/99.6	99.3/99.8
Volatile FP Release (%)	0.012/13.1	0.01/0.19	0.01/1.2	0.63/39.4	9.10/28.6
Non-Volatile FP Release (%)	8E-5/1E-4	4E-5/6E-5	4E-5/3E-4	0.010/0.013	0.32/0.35
Tellurium-based FP Release (%)	7E-3/3.8	3E-3/0.48	3E-3/1.8	0.66/9.8	11.9/16.2

TABLE 4.5.6-4 (Continued)
QUAD CITIES NUCLEAR STATION
CONTAINMENT AND SOURCE TERM SENSITIVITY ANALYSIS

KEY SEQUENCE CHARACTERISTIC	BASE CASE	SMALL LMT EARLY	LARGE LMT EARLY	SMALL LMT DELAYED	LARGE LMT DELAYED
Sequence MAAP Run Database ID No.	Q930701	Q930701A	Q930701B	Q930701C	Q930701D
Sequence Designator	LLCO	--	--	--	--
CORE/CONTAINMENT RESPONSE					
Time of Core Uncovery (hr)	9.02	9.02	9.02	9.02	9.02
Time of Core Relocation (hr)	10.4	10.4	10.4	10.4	10.4
Time of Vessel Failure (hr)	12.4	12.4	12.4	12.4	12.4
Time of Containment Failure (hr)	31	12.4/30.5	12.4	12.9/30.4	12.9
Time of Venting (hr)	12.4	12.4	12.4	12.4	12.4
Maximum Drywell Pressure (psig)	53.5	53.5	53.5	53.5	53.5
Maximum Drywell Temperature (°F)	861	819	665	816	673
Fraction of Clad Reacted in Vessel	0.1003	0.1003	0.1003	0.1003	0.1003
ENVIRONMENTAL RELEASE @ 24 hr/48 hr					
Noble Release (%)	75.9/100	71.5/100	95.2/95.3	70.6/100	94.3/94.4
Volatile FP Release (%)	0.029/28.2	2.35/24.0	19.8/29.6	2.06/24.9	15.6/25.5
Non-Volatile FP Release (%)	9E-05/0.28	8E-03/0.24	0.23/0.23	7E-03/0.25	0.20/0.21
Tellurium-based FP Release (%)	0.080/44.8	0.71/38.5	5.74/10.3	0.67/40.8	5.47/7.79

TABLE 4.5.6-4 (Continued)
QUAD CITIES NUCLEAR STATION
CONTAINMENT AND SOURCE TERM SENSITIVITY ANALYSIS

KEY SEQUENCE CHARACTERISTIC	BASE CASE	FCHF = 0.02			
Sequence MAAP Run Database ID No.	Q930304	Q930304A			
Sequence Designator	MEFG				
CORE/CONTAINMENT RESPONSE					
Time of Core Uncovery (hr)	0.79	0.79			
Time of Core Relocation (hr)	1.58	1.58			
Time of Vessel Failure (hr)	3.03	3.03			
Time of Containment Failure (hr)	--	--			
Time of Venting (hr)	--	--			
Maximum Drywell Pressure (psig)	46.9				
Maximum Drywell Temperature (°F)	32.6				
Fraction of Clad Reacted in Vessel	0.1002				
ENVIRONMENTAL RELEASE @ 24 hr/48 hr					
Noble Release (%)	1.09/2.23	.081/1.56			
Volatile FP Release (%)	6E-03/7E-03	2E-3/3E-3			
Non-Volatile FP Release (%)	2E-04/1E-05	1E-4/3E-4			
Tellurium-based FP Release (%)	0.0/0.0	0.0/0.0			

TABLE 4.5.6-4 (Continued)
QUAD CITIES NUCLEAR STATION
CONTAINMENT AND SOURCE TERM SENSITIVITY ANALYSIS

KEY SEQUENCE CHARACTERISTIC	BASE CASE	FCRBLK = 0	FCRBLK = -1	FCRBLK = +1	FCRBLK = 0.02
Sequence MAAP Run Database ID No.	Q930806	Q930806A	Q930806B	Q930806C	Q930806D
Sequence Designator	BLAY	--	--	--	--
CORE/CONTAINMENT RESPONSE					
Time of Core Uncovery (hr)	9.75	9.75	9.75	9.75	9.75
Time of Core Relocation (hr)	11.2	11.2	11.2	11.2	11.2
Time of Vessel Failure (hr)	13.4	14.6	12.5	13.4	13.4
Time of Containment Failure (hr)	18.7	16.3	12.5	19.8	18.4
Time of Venting (hr)	--	--	--	--	--
Maximum Drywell Pressure (psig)	76.2	91.6	109	74	86.4
Maximum Drywell Temperature (°F)	777	705	628	749	883
Fraction of Clad Reacted in Vessel	0.1041	0.205	0.4476	0.0782	0.1041
ENVIRONMENTAL RELEASE @ 24 hr/48 hr					
Noble Release (%)	17.3/99.7	29.6/99.9	98.6/99.7	12.3/99.5	26.8/99.0
Volatile FP Release (%)	0.57/41.0	1.0/18.9	0.57/0.88	0.33/28.0	2.7/63.2
Non-Volatile FP Release (%)	0.011/0.015	0.010/0.014	7E-3/7E-3	8E-3/0.013	7E-3/0.13
Tellurium-based FP Release (%)	0.81/12.8	0.84/9.0	0.29/0.62	0.59/14.5	2.8/47.9

TABLE 4.5.6-5
KEY RESULTS OF LEVEL 2 SENSITIVITY CASES

SENSITIVITY CASE CECo-MAAP RUN DATABASE ID NO.	BASE CASE SEQUENCE DESIGNATOR	DIFFERENCE FROM BASE CASE SEQUENCE	KEY DIFFERENCES FROM BASE CASE RESULTS
Q930701A	LLCO	1-inch diameter hole opened in drywell shell 2 minutes after reactor vessel failure	No effect on timing of key events, but significant early volatile/Te release fractions.
Q930701B	LLCO	20 square inch hole opened in drywell shell 2 minutes after reactor vessel failure	No effect on timing of key events, but very large early volatile/Te release fractions.
Q930701C	LLCO	1-inch diameter hole opened in drywell shell 30 minutes after reactor vessel failure	No effect on timing of key events, but significant early volatile/Te release fractions.
Q930701D	LLCO	20 square inch hole opened in drywell shell 30 minutes after reactor vessel failure	No effect on timing of key events, but very large early volatile/Te release fractions.
Q930304A	MEFG	Parameter FCHF changed from default value of 0.10 to 0.02	No impact on containment status or fission product release; much greater pedestal concrete erosion depth
Q930806A	BLAY	Parameter FCRBLK not changed from default of 0 to +1 when 4 core nodes have accumulated > 120% of initial UO_2 mass	RPV failure 1.2 hr later; 450 lb more H_2; drywell failure 2.4 hr earlier; slightly smaller volatile fission product release.
Q930806B	BLAY	Parameter FCRBLK initially set to -1 and not changed to +1 when 4 core nodes have accumulated > 120% of initial UO_2 mass	RPV failure 0.9 hr earlier; 1700 lb more H_2; drywell failure at RPV failure; much smaller non-Noble fission product release.
Q930806C	BLAY	Parameter FCRBLK initially set to +1	Same RPV failure time; 100 lb less H_2; drywell failure 1.1 hr later; comparable fission product releases.
Q930806D	BLAY	Parameter FCHF changed from default value of 0.10 to 0.02	Slightly earlier drywell failure; much larger non-volatile fission product release; slightly greater pedestal concrete erosion depth
Q930704A	LLAX	No wetwell venting and only wetwell failure allowed	Much smaller early Noble gas release; much smaller late non-Noble fission product release.
Q930704B	LLAX	No wetwell venting and only large wetwell failure allowed	Much smaller early Noble gas release; much smaller late volatile fission product release.
Q930704C	LLAX	No wetwell venting and only drywell failure allowed	Much larger early non-Noble fission product release; somewhat larger volatile/Te and much larger non-volatile late releases.
Q930704D	LLAX	No wetwell venting and only large drywell failure allowed	Much larger early non-Noble fission product release; somewhat larger volatile/Te and much larger non-volatile late releases.

4.5.6.2 Containment Failure Timing

The influence of the core melt progression model and ex-vessel core debris coolability on containment failure timing have been investigated by the Level 2 sensitivity analyses and are discussed below.

Core Melt Progression Model - As discussed above, MAAP allows three options in its core melt progression model. These options result in a variety of in-core hydrogen production estimates and, therefore, affect containment pressurization behavior. As described above, all base case Level 2 runs use the default model option until four core nodes have accumulated more than an additional 20% of their initial fuel mass; at this time, the channel blockage model is assumed. The impact of this modeling approach was investigated by cases Q930806A,B,C in which each of the core melt progression model options was chosen at the beginning of the sequence and not changed throughout. The effect of the core melt progression model assumption is a range of 1798 lb (404 lb to 2202 lb) for in-vessel hydrogen production, with a corresponding range of 7.3 hr (12.5 hr to 19.8 hr) for containment failure time. A larger spread in containment failure times would be expected for a sequence which reached core damage earlier due to the lower containment pressure and temperature that would exist at vessel failure, while a smaller spread in containment failure times would be expected for a sequence which reached core damage later.

Ex-Vessel Core Debris Coolability - MAAP models ex-vessel core debris cooling using a Critical Heat Flux (CHF) formula in which an experimentally-determined coefficient (called FCHF in MAAP) sets the magnitude of the heat flux out of the core debris. The recommended default value of FCHF is 0.1, which corresponds to saturated pool boiling CHF. All Level 2 CECo-MAAP runs were performed using this default value. The impact of this modeling approach was investigated by running cases Q930304A and Q930806D in which the value of FCHF was set to 0.02 to simulate a stable, long-term film-boiling condition on the core debris in containment. CECo-MAAP run Q930304A is a "wet" case in that Core Spray pumps inject into a failed reactor vessel and this water pours onto the core debris in containment; run Q930806D is a "dry" case in that no water is provided to cool the core debris in containment. The impact of less effective ex-vessel core debris cooling for the wet case is the occurrence of concrete erosion by core debris in the pedestal sumps through the end of the sequence and a higher containment pressure than the base case; no containment failure is predicted for either the "wet" case sensitivity run or its base case. For the "dry" case, the effect of less effective ex-vessel core debris cooling is a 0.3 hr earlier containment failure. An earlier containment failure time would be expected with less effective core debris cooling by water because of increased radiant heating of the containment during the relatively short interval that the water pool in the drywell is boiled off by the core debris.

4.5.6.3 Fission Product Releases

The influence of containment failure timing, size, and location on fission product release behavior have been investigated by the sensitivity analyses and are discussed below.

Containment Failure Timing - The time interval between reactor failure and containment failure is a primary influence on fission product release magnitudes because it defines the time period of maximum effectiveness for naturally-occurring and operator-initiated fission product retention mechanisms. All non-ATWS Level 2 base case runs predicted significant time intervals between vessel failure and containment failure (5 to 13 hr for no-venting cases; 18 to 43 hr for venting cases). The impact on fission product releases for shorter intervals between these occurrences was investigated by running CECo-MAAP cases for liner melt-through (Q930701A, Q930701C), and with the core melt progression model parameter FCRBLK set equal to -1 (Q930806B).

The liner melt-through cases involved opening a small hole (one-inch diameter based on the Liner Melt-Through Phenomenological Evaluation Summary) at 2 minutes and at 30 minutes after vessel failure (Note: the LMT PES estimated a likely time for occurrence as tens of minutes after vessel failure). Fission product releases were very similar for both liner melt-through sensitivity cases. Compared to the base case results, in which containment failure was predicted about 18.6 hr after vessel failure, the earlier containment failure causes much higher fission product releases at 24 hr into the sequence (about 100 times higher volatile, non-volatile and Tellurium-based fission product releases), but has negligible effect on fission product magnitudes 48 hours after the start of the sequence.

The variation of core melt progression model case involved setting the parameter FCRBLK to -1 (see above for a discussion of the MAAP core melt progression model) to maximize in-vessel hydrogen production. Containment pressure and temperature were sufficiently elevated at reactor vessel failure that the subsequent energy release caused immediate containment failure. Compared to the base case results, in which containment failure was predicted about 5 hr after vessel failure, the early containment failure causes much higher Noble gas fission product releases at 24 hr into the sequence (about 6 times higher), but generally lower volatile, non-volatile and Tellurium-based fission product releases. The latter fission product magnitudes 48 hours after the start of the sequence are much lower than predicted for the base case due to less volatile revaporization and a lower driving pressure difference between the drywell and the environment.

Containment Failure Size - The size of the containment failure can have a large influence on fission product release magnitudes because it defines the driving force for the initial and later fission product releases. For example, a containment failure large enough to quickly depressurize containment reduces the time available for naturally-occurring fission product retention mechanisms to operate. Subsequently in the sequence, since no significant pressure can build up in containment, fission products in the containment atmosphere are released at a lower rate. All Level 2 base case runs used a drywell failure area that was small enough to slowly depressurize the drywell at drywell shell temperatures below 700°F, and large enough to rapidly depressurize the drywell at drywell shell temperatures above 700°F (per the Containment Over-Pressure Phenomenological Evaluation Summary). The impact on fission product releases of containment failure size was investigated by running cases with liner melt-through (Q930701A, B, C, D), and with small and large torus and drywell failures (Q930704A, B, C, D).

The liner melt-through cases involved opening a small hole (one-inch diameter based on the Liner Melt-Through Phenomenological Evaluation Summary) and a large hole (20 sq. in.) at 2 or 30 minutes after vessel failure (Note: the LMT PES estimated a likely time for occurrence as tens of minutes after vessel failure). Compared to the small liner melt-through case results, the large liner melt-through causes much higher fission product releases at 24 hr into the sequence (about 8 times higher volatile and Tellurium-based fission product releases, about 30 times higher non-volatile fission product release); 48 hours after the start of the sequence volatile and non-volatile fission product releases are about the same for both failure sizes, but Tellurium-based fission product releases for the large liner melt-through cases are 4 to 5 times lower than those for the smaller liner melt-through cases.

The torus/drywell failure size sensitivity cases considered four variations on sequence LLAX. All four of these variants involved failing to operate the torus vent. Then, in two cases containment failure was forced to occur in the torus, while the other two cases forced drywell failure. For each failure location the failure size was either small enough to cause a slow containment depressurization (~ hours) or large enough to cause a rapid depressurization (~ minutes).

Comparing the results of the wetwell failure cases, at 24 hours the fission product release magnitudes are comparable because containment has not yet been opened. At 48 hr after the sequence start, the Noble gas releases are comparable, while the other releases are about 5 times higher for the larger failure size. Comparing the results of the drywell failure cases, the Noble gas releases are about 6 times higher at 24 hr and the other fission product release magnitudes are about 15 to 30 times higher for the large failure size. At 48 hr after the sequence start, the Noble gas, volatile, and Tellurium-based releases are all comparable, while the non-volatile releases remain about 30 times higher for the large failure size.

Containment Failure Location - The torus/drywell failure sensitivity cases considered four variations on sequence LLAX. All four of these variants involved failing to operate the torus vent. Then, in two cases containment failure was forced to occur in the torus, while the other two cases forced drywell failure. For each failure location the failure size was either small enough to cause a slow containment depressurization (~ hours) or large enough to cause a rapid depressurization (~ minutes).

Comparing the results of the small wetwell and drywell failure cases, at 24 hours the fission product release magnitudes are about 100 to 200 times higher for the drywell failure. At 48 hr after the sequence start, the Noble gas releases are comparable, while the volatile and non-volatile releases are about 200 times higher and the Tellurium-based releases about 20 times higher for the drywell failure. Comparing the results of the large wetwell and drywell failure cases, the releases are about 3 to 4 orders of magnitude higher at 24 hr for the drywell failure case. At 48 hr after the sequence start, the Noble gas releases are comparable, the volatile, and Tellurium-based releases are about a factor of 10 higher, and the non-volatile releases remain about 3 orders of magnitude higher for the drywell failure case.

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4.6 IPE Results

This section provides a discussion and explanation of the Quad Cities IPE accident sequence results. First, the IPE screening criteria that were used to identify sequences to be reported to the NRC are described. Then, traditional results based on the mission time of 24 hours are reported: overall core damage frequency, with a subsequent breakdown of core damage frequency by initiating event, individual IPE screening criteria, Plant Damage State, containment status, and equipment/operator failures. Then, unique features of the Quad Cities plant design are discussed, and an evaluation of the decay heat removal capability of the Quad Cities plants is presented. Finally, an innovative aspect of the Quad Cities IPE/AM program is evaluated: the so-called Accident Management (AM) sequence endstates.

In the Quad Cities IPE/AM Program, two types of AM sequences have been defined. "Success with Accident Management" or "SAM" endstates have been defined for sequences with no core damage within 24 hours, but requiring accident management actions after 24 hours to assure continued long-term core cooling. "Containment success with Accident Management" or "CAM" endstates have been defined for sequences with core damage and no containment failure within 24 hours, but requiring accident management actions after 24 hours to assure continued long-term containment integrity. The method used for classifying sequences in the Quad Cities IPE/AM project is shown in Table 4.6-1.

4.6.1 IPE Screening Criteria

The following screening criteria were followed to determine those important sequences to be reported to the NRC that might lead to core damage or unusually poor containment performance:

A. Any sequence that contributes 1E-07 or more per reactor year to core damage.

B. All sequences within the upper 95 percent of the total core damage frequency.

C. All sequences within the upper 95 percent of the total containment failure probability.

D. Sequences that contribute to a containment bypass frequency in excess of 1E-08 per reactor year.

E. Any sequence that CECo determines from previous applicable PRAs or by engineering judgment to be of interest irrespective of core damage frequency or estimated containment performance.

The total number of unique sequences reported was determined by the criteria listed above, but did not exceed the 100 most significant sequences. Sequences meeting more than one criterion were also identified. In addition to sequences reported under the above

TABLE 4.6-1
SEQUENCE CLASSIFICATION METHOD

SEQUENCE CLASSIFIED AS:	SEQUENCE TIME		APPROXIMATE SOURCE TERM MAGNITUDE	
	0 to 24 HR	> 24 HR	AT 24 HR	AT 48 HR
SUCCESS	CD NO Vent NO CF N/A	CD NO Vent NO CF N/A	Noble Gas 0 Volatile 0	Noble Gas 0 Volatile 0
SUCCESS WITH ACCIDENT MANAGEMENT (SAM)	CD NO Vent NO CF N/A	CD YES Vent NO CF N/A	Noble Gas 0 Volatile 0	Noble Gas * Volatile *
CORE DAMAGE, CONTAINMENT VENTED AND INTACT	CD YES Vent YES CF NO	CD YES Vent YES CF NO	Noble Gas ~100% Volatile ~0.01%	Noble Gas ~100% Volatile ~0.1%
CORE DAMAGE, CONTAINMENT VENTED AND INTACT	CD YES Vent YES CF NO	CD YES Vent YES CF YES	Noble Gas ~25% Volatile ~0.01%	Noble Gas ~100% Volatile ~10%
CORE DAMAGE, CONTAINMENT NOT VENTED AND INTACT (POSSIBLE CAM)	CD YES Vent NO CF NO	CD YES Vent NO CF YES	Noble Gas ~1% Volatile ~0.01%	Noble Gas ~100% Volatile ~10%
CORE DAMAGE, CONTAINMENT VENTED AND FAILED	CD YES Vent YES CF YES	CD YES Vent YES CF YES	Noble Gas ~100% Volatile ~10%	Noble Gas ~100% Volatile ~10%
CORE DAMAGE, CONTAINMENT NOT VENTED AND FAILED	CD YES Vent NO CF YES	CD YES Vent NO CF YES	Noble Gas ~100% Volatile ~10%	Noble Gas ~100% Volatile ~10%

CD = Core Damage
Vent = Wetwell vent operated in accordance with procedures within 24 hours
CF = Containment Failure
N/A = Not Applicable
* = Not Estimated
CAM = Containment success with Accident Management
Note: A sequence is designated as "core damage" if core damage is predicted to occur within 24 hours, in accordance with the traditional approach. Similarly, a sequence is designated as "containment intact" if containment failure is not predicted to occur within 24 hours, in accordance with the traditional approach.

screening criteria, any sequence that dropped below the core damage frequency criteria because the frequency was reduced by more than an order of magnitude by credit taken for human recovery actions not in the Quad Cities emergency procedures were identified. Additionally, sequences which have been designated as "SAM" and respective insights of major importance are also reported.

These screening criteria are consistent with those recommended in NUREG-1335, Section 2.1.6

4.6.2 Summary of Results

This section provides a discussion and explanation of the Quad Cities IPE accident sequence results previously tabulated in Section 4.5.3. The core damage frequency (CDF) and the initiating event frequency are shown in Table 4.6.2-1 by initiating event. As can be seen from this table, the top two initiators contribute 68.25% to the total CDF, the top three initiators contribute 79.05% , and 98.3% of the CDF comes from the top seven initiating events. The remaining ten initiating events contribute 1.7% of the total core damage frequency.

The top contributor to core damage frequency is Dual Unit Loss of Offsite Power at 56.67% followed by Single Unit Loss of Offsite Power at 11.58%, General Transient at 10.8%, ATWS at 7.68%, Medium LOCA at 5.16%, Loss of 125VDC Bus 1B-1 at 3.43%, and Loss of Service Water at 2.99%.

The remaining discussion in this section is based upon the screening criteria presented in Section 4.6.1.

- Screening criterion A required that "any sequence that contributes 1E-07 or more per reactor year to core damage" be identified. The first 4 sequences previously shown in Table 4.5.3-1 fall into this criterion.

- Screening criterion B required that "all sequences within the upper 95 percent of the total core damage frequency" be identified but not to exceed a total of 100 sequences. The first 100 sequences previously shown in Table 4.5.3-1 add up to 94.76% of the total CDF.

- Screening criterion C required that "all sequences within the upper 95 percent of the total containment failure probability" be identified but not to exceed a total of 100 core damage sequences. The total containment failure probability (CFP) is 1.44E-06 (which represents 66.52% of the total core damage frequency). Those sequences that are identified by the letter "C," "E," "O," "Q," "R," "S," "T," "X" or "Y" as the fourth character and "GG" in the third and fourth characters in the five-character endstate designator[1] lead to containment failure. There are 54 such sequences in the top 100 core damage sequences and these 54 sequences represent 95.67% of the total containment failure probability.

1 The endstate designator definitions are located in Section 4.1.3.

TABLE 4.6.2-1
CORE DAMAGE FREQUENCY BY INITIATING EVENT

INITIATING EVENT	INITIATING EVENT FREQUENCY (/YR)	CORE DAMAGE FREQUENCY (/YR)	PERCENT CONTRIBUTION
Dual Unit LOSP[1]	1.61E-02	1.229E-06	56.67
Single Unit LOSP	3.20E-02	2.510E-07	11.58
General Transient	3.87	2.341E-07	10.8
ATWS[2]	1.16E-04	1.665E-07	7.68
Medium LOCA[3]	8.00E-04	1.118E-07	5.16
L1B1[4]	1.01E-03	7.442E-08	3.43
LOSW[5]	9.10E-03	6.486E-08	2.99
Large LOCA	3.00E-04	2.103E-08	0.97
LB13[6]	4.65E-03	5.190E-09	0.24
LOIA[7]	1.43E-02	4.029E-09	0.19
Small LOCA	3.00E-03	2.556E-09	0.12
IORV[8]	1.06E-01	1.234E-09	0.06
LB14[9]	5.88E-04	8.150E-10	0.04
LB18[10]	4.18E-04	6.786E-10	0.03
L182[11]	1.34E-03	4.085E-10	0.02
LB11[12]	2.75E-04	3.302E-10	0.02
LB12[13]	2.05E-04	3.066E-10	0.01
ISLOCA[14]	1.20E-07	2.631E-10	0.01
TOTAL		2.169E-06	100

Notes for Table 4.6.2-1:

1.	LOSP	=	Loss of Offsite Power
2.	ATWS	=	Anticipated Transient Without Scram
3.	LOCA	=	Loss of Coolant Accident
4.	L1B1	=	Loss of 125VDC Bus 1B-1
5.	LOSW	=	Loss of Service Water
6.	LB13	=	Loss of Bus 13
7.	LOIA	=	Loss of Instrument Air
8.	IORV	=	Inadvertent Open Relief Valve
9.	LB14	=	Loss of Bus 14
10.	LB18	=	Loss of Bus 18
11.	L182	=	Loss of MCC 18-2
12.	LB11	=	Loss of Bus 11
13.	LB12	=	Loss of Bus 12
14.	ISLOCA	=	Interfacing System LOCA

- Screening criterion D required that "sequences that contribute to a containment bypass frequency in excess of 1E-08 per reactor year" be identified. There are no interfacing system LOCA sequences which cause a containment bypass greater than 1E-08 per year.

- Screening criterion E required that any sequences deemed to be of interest be identified. There are no sequences of interest that fall under this criterion.

The accident sequences as shown in Table 4.6.2-2 meet one or more of the screening criteria. Sequences #1 through #4 meet screening criterion A. All top 100 sequences meet screening criterion B. There are 54 sequences which meet screening criterion C. There are two sequences that meets criteria A, B, and C only. There are two sequence that meet criteria A and B only. There are 44 sequences that meet criterion B only. There are 53 sequences which meet criteria B and C only. There are no containment bypass (interfacing systems LOCA) sequences above 1E-08 or within the top 100 sequences.

The containment failure sequences for screening criterion C are presented in Table 4.6.2-3. The sequences with an "O," or "C" as the fourth character in the endstate designator[1] are "semi-controlled" releases in which the containment may be vented and later fails due to high temperature; there are 13 such sequences in the top 100 core damage sequences, contributing 2.04E-08 to the total containment failure probability. The sequences with a "E" (except TEFE) and "GG" are "controlled releases" in which the containment is vented and does not fail structurally during the 24 hour mission time; there are 10 such sequences in the top 100 core damage sequences, contributing 1.06E-07 to the total containment failure probability. The sequences with a "Q," "R," "S," "T" or "Y" are containment high temperature/pressure failures ("uncontrolled releases") either in the drywell ("R," "T" or "Y") or wetwell ("Q" or "S"); within the top 100 core damage sequences there are 31 "Q/R/S/T/Y" sequences contributing a total of 1.25E-06. The top 100 core damage sequences, therefore, contribute a total of 95.67% to the total containment failure probability.

Containment isolation failures were not included in the accident sequence modeling. During normal operation, the containment is normally isolated and kept inerted with nitrogen and at a slightly positive pressure. Therefore, containment isolation failures were not modeled.

There were no accident sequences that dropped below the core damage frequency criteria because the frequency had been reduced by more than an order of magnitude by credit taken for human recovery actions not defined in the Quad Cities emergency procedures.

The core damage frequency by "plant damage state" is shown in Table 4.6.2-4 for plant damage states contributing more than 0.5% to CDF. As can be seen from this table, failures of the high-pressure makeup function tend to dominate the overall core damage frequency. In fact, 86% of the total CDF involves plant damage states with a loss of the high-pressure makeup function. This is primarily attributed to the long-term consequences of station blackout and hardware failures of high pressure injection following a LOOP.

1 The endstates represented here are only from the top 100 sequences. Other endstates are included in these classifications (See Table 4.6.2-6).

Table 4.6.2-5 identifies the key contributors, both hardware failures and operator errors for each of the unsuccessful events in the top 14 sequences. Because screening criterion A only identifies 4 sequences, this table lists sequences down to 1% of the CDF. These 14 sequences contribute approximately 82 % to the CDF. Sequences #15 and above have a CDF contribution of less than 1% per sequence and a frequency of less than 3.9E-07 .

Conclusions From Accident Sequence Quantification

The core damage frequency of 2.17E-06 for Quad Cities is dominated (39.77% of CDF) by sequence #1, a dual-unit loss of offsite power and failure to restore AC power, which leads to SBO and late (6-24 hours) core damage. The dual-unit loss of offsite power initiator contributes 56.67% of the CDF and the single-unit loss of offsite power contributes another 11.58% for a total of 68.25% contribution to the CDF from loss of offsite power. The General Transient initiator contributes 10.88%, the ATWS initiator contributes 7.68%, the Medium LOCA initiator contributes 5.16%, the Loss of 125VDC Bus 1B-1 contributes 3.43%, and the Loss of Service Water initiator contributes 2.99%.. These top seven initiators contribute 98.31% of the CDF and comprise 28 of the top 28 accident sequences. The top 14 sequences have individual contributions greater than 1% of the CDF. Sequences numbered #15 and higher all contribute less than 1%, indicating a very flat sequence distribution.

These results show a significant contribution to CDF from support systems, specifically AC power. Also, there is a significant contribution to CDF from failures of high-pressure makeup systems. There is a minor contribution from operator actions such as failure to initiate depressurization or torus cooling, or failing to align RHR or CS pump suction to the CCST.

Given core damage, there is about a 67% probability that containment will also fail. Most of the CDF occurs early, less than two hours into the event. The contribution of containment rapid, high-pressure failure is small. Only 18 of the top 100 sequences are ATWS sequences. Containment failure subsequent to venting occurs in 17 of the top 100 sequences. Containment failure without venting occurs in 13 of the top 100 sequences and venting without containment failure occurs in six of the top 100 sequences . In 43 of the top 100 sequences, the containment is intact without venting.

TABLE 4.6.2-2
SEQUENCES MEETING ONE OR MORE SCREENING CRITERION

SEQUENCE NUMBER	SCREENING CRITERIA			
	A - >1E-07 CDF	B - 95% CDF	C - 95% CFP	D - >1E-08 Bypass
1-2	Yes	Yes	Yes	
3-4	Yes	Yes		
5		Yes		
6		Yes	Yes	
7-9		Yes		
10-11		Yes	Yes	
12		Yes		
13-15		Yes	Yes	
16		Yes		
17		Yes	Yes	
18-19		Yes		
20		Yes	Yes	
21		Yes		
22-23		Yes	Yes	
24		Yes		
25		Yes	Yes	
26-27		Yes		
28-31		Yes	Yes	
32-33		Yes		
34		Yes	Yes	
35		Yes		
36		Yes	Yes	
37		Yes		
38		Yes	Yes	
39-40		Yes		
41		Yes	Yes	
42		Yes		
43		Yes	Yes	
44		Yes		
45		Yes	Yes	
46		Yes		
47-53		Yes	Yes	
54-56		Yes		
57-59		Yes	Yes	
60-62		Yes		
63-66		Yes	Yes	
67		Yes		
68		Yes	Yes	
69-71		Yes		
72		Yes	Yes	
73		Yes		
74-77		Yes	Yes	
78-79		Yes		
80-81		Yes	Yes	
82-83		Yes		
84		Yes	Yes	
85-88		Yes		
89-92		Yes	Yes	
93-94		Yes		
95		Yes	Yes	
96		Yes		
97		Yes	Yes	
98		Yes		
99-100		Yes	Yes	

TABLE 4.6.2-3
CONTAINMENT FAILURE SEQUENCES[1]

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	DAMAGE STATE Containment Flr (Release) Type (4)	EVENT (5)	VALUE (6)	DESCRIPTION (7)
1	8.62E-07	39.8	BLAYF Uncontrolled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
2	1.56E-07	7.2	BEAYF Uncontrolled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				ROP2	1.00E+00	FAILURE TO REC OSP; SBO, SHORT TIME AVAILABLE
6	7.04E-08	3.25	BLAYF Uncontrolled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				ROP2	5.09E-02	FAILURE TO REC OSP
10	5.22E-08	2.41	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
11	4.08E-08	1.88	TEFEB Controlled	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
13	2.72E-08	1.25	TEFEB Controlled	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
14	2.43E-08	1.12	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
15	2.04E-08	0.94	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
17	1.56E-08	0.72	TEERF Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)

[1] Notes located at the end of the table

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	DAMAGE STATE Containment Flr (Release) Type (4)	EVENT (5)	VALUE (6)	DESCRIPTION (7)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
20	1.31E-08	0.6	TEFEB Controlled	GTR	3.87E+00	GENERAL TRANSIENT IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
22	8.73E-09	0.4	TEFEB Controlled	GTR	3.87E+00	GENERAL TRANSIENT IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
23	7.27E-09	0.34	TEERF Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
25	6.92E-09	0.32	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
28	6.09E-09	0.28	TEERF Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
29	6.04E-09	0.28	ALCEB Controlled	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
30	5.80E-09	0.27	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
31	5.71E-09	0.26	AEGGA Controlled	LLOCA	3.00E-04	LLOCA IE
				CS	6.66E-03	CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE
				LV	2.89E-03	LV FAILS; ALL SUPPORTS AVAILABLE
34	5.21E-09	0.24	LLBOG Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
36	4.31E-09	0.2	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
38	3.29E-09	0.15	LLCOG Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	DAMAGE STATE Containment Flr (Release) Type (4)	EVENT (5)	VALUE (6)	DESCRIPTION (7)
41	2.64E-09	0.12	LLBOG Semi-Controlled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
43	2.07E-09	0.1	TEERF Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
45	2.04E-09	0.09	ALCEB Controlled	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	1.00E+00	EVENT FAILS
				SBCS	1.00E+00	EVENT FAILS
47	1.83E-09	0.08	BLASB Uncontrolled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO OCCURS IN UNIT 1
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
48	1.82E-09	0.08	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				SLC	2.35E-04	SLC FAILS; 1/2 PUMPS; ALL SUPPORTS AVAILABLE
49	1.79E-09	0.08	LLCOG Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP B SUCCESS
50	1.73E-09	0.08	TEERF Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				MC	6.16E-01	MAIN COND FAILS (GIVEN FW FAILS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	4.50E-02	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
51	1.70E-09	0.08	TEFSB Uncontrolled	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)
52	1.66E-09	0.08	LLCOG Semi-Controlled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
53	1.62E-09	0.07	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				RPT1	1.99E-04	AUTO RPT FAILS; ALL SUPPORTS AVAILABLE
57	1.34E-09	0.06	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	DAMAGE STATE Containment Flr (Release) Type (4)	EVENT (5)	VALUE (6)	DESCRIPTION (7)
				MC	3.18E-01	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				AT1	1.28E-02	ATWS1 ACTUATION FAILS
				AT2	1.28E-02	ATWS2 ACTUATION FAILS (GIVEN AT1 SUCCESS)
58	1.31E-09	0.06	BLAYF Uncontrolled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				DGB	1.35E-01	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)
				1ES	1.51E-03	LOSS OF ESS BUS (901-49), 17 & 18 UNAVAIL
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
59	1.29E-09	0.06	TEERF Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (28)
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)
63	1.18E-09	0.05	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	8.70E-03	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (1)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
				OAL	1.40E-01	OPTR FAILS TO CONTROL RV LEVEL AFTER ATWS (11)
64	1.14E-09	0.05	LLBOG Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	1.00E+00	EVENT FAILS
65	1.13E-09	0.05	TEFSB Uncontrolled	L1B1	1.01E-03	LOSS OF 125VDC BUS 1B-1 IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				ADS	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)
66	1.12E-09	0.05	AEGGA Controlled	LLOCA	3.00E-04	LLOCA IE
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				LV	1.00E+00	EVENT FAILS
68	9.87E-10	0.05	TLBSB Uncontrolled	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				OHX	1.00E-05	OPTR FAILS TO ALIGN COOLING TO RHR (2)
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
				OCST	1.00E+00	OPTR FAILS TO ALIGN TO CCST SOURCE (25)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
72	9.47E-10	0.04	ALCEB Controlled	LLOCA	3.00E-04	LLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				DG1	1.37E-01	LOP FROM DG1 TO BUS 14-1 (24 HRS)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
74	9.09E-10	0.04	LLCOG Semi-Controlled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP B SUCCESS

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	DAMAGE STATE Containment Flr (Release) Type (4)	EVENT (5)	VALUE (6)	DESCRIPTION (7)
75	8.59E-10	0.04	BLASB Uncontrolled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO IN UNIT 1, NO SBO IN UNIT 2
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.99E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 1T2 AVAILABLE
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
76	8.26E-10	0.04	BEAYF Uncontrolled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 1, SBO IN UNIT 2
				HP1	8.32E-02	HP FAILS; 1R1, 1T2 AVAILABLE
				RCIC	1.00E+00	EVENT FAILS
				ROP2	1.00E+00	FAILURE TO REC OSP; SBO, SHORT TIME AVAILABLE
77	8.00E-10	0.04	LEACD Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	9.60E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (19)
				OAD1	5.20E-02	OPTR FAILS TO INITIATE ADS (12)
				ROP1	1.00E+00	EVENT FAILS
80	6.98E-10	0.03	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				FWA	1.28E-01	FW FAILS (FRACTION OF IEs THAT ARE LOFW)
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	8.70E-03	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (1)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
81	6.83E-10	0.03	LLCOG Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	6.14E-03	CCST SUCTION VALVES FAIL, RHR PUMP A SUCCESS
84	6.65E-10	0.03	TEEQC Uncontrolled	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				AT1	1.28E-02	ATWS1 ACTUATION FAILS
				OIADS	3.00E-03	OPTR FAILS TO INHIBIT ADS (26)
89	6.38E-10	0.03	AEGGA Controlled	LLOCA	3.00E-04	LLOCA IE
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				CS	5.24E-02	CS FAILS; 13-1, 18, 1M1 AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				LV	5.01E-01	LV FAILS; 18, 19, 1M1 AVAILABLE
90	6.11E-10	0.03	IEBOG Semi-Controlled	IORV	1.06E-01	IORV + OTHER IEs x RVC
				1M1	7.34E-05	LOSS OF 125VDC TB MAIN BUS 1A
				1R1	8.38E-05	LOSS OF 125VDC TB RESERVE BUS 1B-1
				FW	1.00E+00	EVENT FAILS
				HP1	1.00E+00	EVENT FAILS
				RCIC	1.00E+00	EVENT FAILS
				SSMP1	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
91	6.01E-10	0.03	TIGSB Uncontrolled	GTR	3.87E+00	GENERAL TRANSIENT IE
				131	1.84E-04	LOSS OF BUS 13-1, 13 AVAIL
				141	1.26E-02	LOSS OF BUS 14-1 AFTER 13-1, 14 AVAIL
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				LPB	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				CST	1.00E+00	EVENT FAILS
				CS	1.00E+00	EVENT FAILS
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
92	6.00E-10	0.03	MLCSB Uncontrolled	MLOCA	8.00E-04	MLOCA IE

NUMBER (1)	FREQUENCY (2)	PERCENT (3)	DAMAGE STATE Containment Flr (Release) Type (4)	EVENT (5)	VALUE (6)	DESCRIPTION (7)
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
				OVNT	5.10E-02	OPTR FAILS TO VENT CONT (18)
95	5.79E-10	0.03	LLBOG Semi-Controlled	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				18	1.13E-02	LOSS OF BUS 18, 13-1 UNAVAIL
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	1.00E+00	EVENT FAILS
97	5.75E-10	0.03	LLBOG Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CST	5.75E-03	CCST SUCTION VALVES FAIL, CS SUCCESS
99	5.52E-10	0.03	LLBOG Semi-Controlled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
100	5.48E-10	0.03	BLATF Uncontrolled	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				141	1.17E-02	LOSS OF BUS 14-1, 14 & DG1 UNAVAIL, X-TIE AVAIL
				SBO?	1.00E+00	SBO OCCURS IN UNIT 1
				ROP2	5.09E-02	FAILURE TO REC OSP
				SSMP1	1.88E-02	SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLP: 241, 2ES)
				LVW	1.00E+00	EVENT FAILS
				LVD	1.00E+00	EVENT FAILS
				WW/DW	2.30E-01	FRAC OF CONT FLRS IN DW (VS. WW)

Notes for Table 4.6.2-3:

1. "Number" refers to accident sequence ranking in the top 100 sequences.

2. "Frequency" is the frequency per year that this sequence is expected to occur.

3. "Percent" is the percent of total core damage represented by this single sequence.

4. "Damage State" is the plant damage state to which this sequence belongs. "Containment Failure (Release) Type" denotes whether the containment failure mode leads to an uncontrolled release (as a containment failure with no venting) or a semi-controlled release (as a vented condition followed by a high temperature failure).

5. "Event" is the list of PRT and support system event tree top events which have failed in this sequence.

6. "Value" is frequency (for initiators) or probability (for failures) associated with each event.

7. "Description" defines the "Event" label.

TABLE 4.6.2-4
CORE DAMAGE FREQUENCY BY PLANT DAMAGE STATE

STATE	DESCRIPTION	FREQUENCY	PERCENT CONTRIBUTION
BLAY	Station Blackout with late core damage (6-24 hours), operator fails to recover offsite power and failure to supply high-pressure coolant makeup	9.37E-07	43.20
LEAB	Loss of Offsite Power (single or dual unit) and early core damage (0-2 hours) with failure to supply high-pressure coolant makeup	2.95E-07	13.62
TEFB	Transient event with early core damage (0-2 hours) and failure to depressurize or maintain sufficient high-pressure coolant makeup	2.04E-07	9.39
BEAY	Station Blackout with early core damage (0-2 hours), operator fails to recover offsite power and failure to supply high-pressure coolant makeup	1.59E-07	7.34
TEEQ	ATWS with early core damage (0-2 hours) and failure to trip recirc pumps or failure to inject SLC	1.25E-07	5.78
MEFG	Medium LOCA with early core damage (0-2 hours) and failure to depressurize or maintain sufficient high-pressure coolant makeup	1.07E-07	4.91
TEFE	Transient event with early core damage (0-2 hours) and failure to depressurize or maintain sufficient high-pressure coolant makeup	9.16E-08	4.22
TEAB	Transient event with early core damage (0-2 hours) with failure to supply high-pressure coolant makeup	5.69E-08	2.62
TEER	ATWS with early core damage (0-2 hours) and failure to trip recirc pumps or failure to inject SLC	3.74E-08	1.72
LLAB	Loss of Offsite Power (single or dual unit) with late core damage (6-24 hours) with failure to supply high-pressure coolant makeup	2.38E-08	1.10
LLCO	Loss of Offsite Power (single or dual unit) with late core damage (6-24 hours) and torus cooling fails	1.82E-08	0.84
LLBO	Loss of Offsite Power (single or dual unit) and late core damage (6-24 hours) with loss of all low-pressure reactor coolant inventory makeup	1.61E-08	0.74
ALCE	Large LOCA with late core damage (6-24 hours) and torus cooling failure	1.08E-08	0.50
	TOTAL	2.08E-06	95.68 of total CDF

TABLE 4.6.2-5
KEY CONTRIBUTORS TO DOMINANT ACCIDENT SEQUENCES

SEQUENCE	EVENT	NODE	DESCRIPTION OF KEY CONTRIBUTORS
1	DLOOP	DG1	DG1 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 14-1
		DG2	DG2 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 24-1
		DGB	DG1/2 fails to run for 6 hours due to common cause mechanisms after DG1 and DG2
		SBO?	Station blackout occurs in Units 1 and 2
		ROP2	Failure to recover offsite power within 6 hours to prevent core damage
2	DLOOP	DG1	DG1 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 14-1
		DG2	DG2 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 24-1
		DGB	DG1/2 fails to run for 6 hours due to common cause mechanisms after DG1 and DG2
		SBO?	Station blackout occurs in Units 1 and 2
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		ROP2	Failure to recover offsite power within 6 hours to prevent core damage
3	LOOP	1TB	By procedure, TBCCW not restarted after LOOP
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
4	MLOCA	HP1	HPCI turbine fails to run for 5 hours
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
5	GTR	OFW1	Operator fails to restart a feedwater pump
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP-3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
6	DLOOP	DG1	DG1 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 14-1
		DG2	DG2 fails to start or run for 6 hours, or output circuit breaker faults cause loss of power at Bus 24-1
		DGB	DG1/2 fails to run for 6 hours due to common cause mechanisms after DG1 and DG2
		SBO?	Station blackout occurs in Units 1 and 2
		HP1	HPCI turbine fails to run for 5 hours
		ROP2	Failure to recover offsite power to prevent core damage
7	DLOOP	1TB	By procedure, TBCCW not restarted after DLOOP
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS

8	GTR	PCSA	Power Conversion System unavailable
		OFW1	Operator fails to restart a feedwater pump
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
9	LOSW	SW	Service water system fails due to initiating event
		1IA	Instrument Air fails due to initiating event
		PCSA	Power Conversion System unavailable
		FW	Feedwater system fails due to initiating event
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		CRD	Control rod drive system injection fails
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
10	ATWS	RCFM	Fraction of RPS failures that are mechanical
		OIADS	Operator fails to inhibit Automatic Depressurization System
11	L1B1	1M1	125VDC main bus 1A fails for 24 hours due to initiating event
		1R1	125VDC reserve bus 1B-1 fails for 24 hours
		FW	Feedwater fails due to loss of control power
		HP1	HPCI fails due to loss of control power
		LPA	RHR A fails due to loss of control power
		LPB	RHR B fails due to loss of control power
		RCIC	RCIC fails due to loss of control power
		SSMP1	SSMP fails due to loss of control power
		ADS	ADS fails due to loss of control power
		CS	Core Spray fails due to loss of control power
12	GTR	FW	Feedwater system fails with all supports available
		HP1	HPCI turbine fails to run for 5 hours
		RCIC	RCIC turbine fails to run for 24 hours
		OSMP3	Operator fails to provide RPV injection from the SSMP (aligned to the CCST) by failing to recognize ECCS signal or by omission of a procedure step
		OAD1	Operator misreads RPV level or omits procedure step to initiate ADS
13	L1B1	1M1	125VDC main bus 1A fails for 24 hours due to initiating event
		1R1	125VDC reserve bus fails for 24 hours
		PCSA	Power Conversion System unavailable
		FW	Feedwater fails due to loss of control power
		HP1	HPCI fails due to loss of control power
		LPA	RHR A fails due to loss of control power
		LPB	RHR B fails due to loss of control power
		RCIC	RCIC fails due to loss of control power
		SSMP1	SSMP fails due to loss of control power
		ADS	ADS fails due to loss of control power
		CS	Core Spray fails due to loss of control power
14	ATWS	MC	Main Condenser unavailable (given FW success) after ATWS
		RCFM	RPS mechanical failure
		OIADS	Operator fails to inhibit Automatic Depressurization System

Summary of Screening Criteria:

A. sequences with core damage frequency greater than 1E-07
B. sequences in upper 95% of total core damage frequency
C. sequences within upper 95% of total containment failure probability
D. sequences with containment bypass frequency greater than 1E-08

Release Categorization

Table 4.6.2-6 breaks down the total core damage frequency into several release frequencies. The release frequencies are combinations of endstate frequencies in which the core has been damaged and the containment is not intact. A "controlled release" is defined as resulting from those core-damage endstates in which the containment is vented through one of the installed vent paths. A "semi-controlled release" is defined as resulting from those core-damage endstates in which the containment fails after having been previously vented. An "uncontrolled release" is defined as resulting from those core-damage endstates in which the containment fails either without having been vented or during ATWS sequences in which venting has no impact.

In Table 4.6.2-6, releases are identified which exceed the limits specified in 10CFR100 (releases exceeding 10CFR100 - RE100). As can be seen from this table, essentially all the sequences fall into this category. There are three types of accident sequences, however, that greatly exceed the contribution to release of the other sequences. These are the ATWS sequences leading to rapid containment failure, the long-term, high-temperature containment failures, and the long-term station blackout sequences. These three sequence types result in larger releases of volatile fission products than expected from other RE100 events. Although ATWS sequences are relatively small contributors to core damage, they deserve specific attention due to the associated source term and because the IPE/AM insights indicate that this source term could potentially be reduced by appropriate use of drywell sprays.

TABLE 4.6.2-6
SUMMARY OF RESULTS FROM RISK PERSPECTIVE

Risk Indicator	IPE Model
Core Damage Frequency	2.17E-06
Controlled Release Frequency (Fourth character of endstate designator: E, F, I, J, K, L or M; Third and fourth character of endstate designator combination: GG)	1.14E-07
RE100 Frequency[1]	1.14E-07
Semi-Controlled Release Frequency (Fourth character of endstate designator: C, D, O, P, or X)	4.08E-08
RE100 Frequency	4.08E-08
Uncontrolled Release Frequency (Fourth character of endstate designator: N, Q, R, S, T, U, V or Y)	1.29E-06
RE100 Frequency	1.29E-06

Notes for Table 4.6.2-6:

1. RE100 is defined here as a release exceeding 10CFR100 exposure limits to either the whole body or the thyroid.

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4.6.3 Unique Features

During the performance of the Quad Cities IPE, a number of good features were identified as insights by the IPE analysts. Such features are those aspects of plant design or operation which may contribute to reducing initiating event frequency, or to enhancing the capability of the station to withstand challenges to the fission product barriers. Section 4.7.1 provides a summary of the insight process. This section highlights those features which are believed to be "unique" or important from the standpoint of preventing core damage accidents or mitigating the consequences of such accidents.

As noted earlier in Section 2.3, the IPE model reflects the Quad Cities as-built condition as it existed in July 1991, with one exception: the hardened containment vent installation was included in the plant model because the modification was imminent at the start of this analysis. Other changes in plant design or operation since that time which may affect the beneficial aspects of the "unique" features will be evaluated as part of the periodic review and update of the Quad Cities PRA - the "Living PRA" process.

Table 4.6.3-1 summarizes those aspects of Quad Cities that are considered to be "unique" features. A brief description of each "unique" feature is provided in the following paragraphs.

Shared Service Water System: The Service Water (SW) System at Quad Cities is shared between Units 1 and 2. This sharing significantly reduces the probability of a loss of SW leading to initiation of an accident condition.

During an accident on one unit, the SW System can provide the heat removal needs of both units with as few as 2 out of 5 SW pumps if the unaffected unit is brought to hot shutdown and non-essential heat loads shed from the SW System. As a result, the frequency for loss of SW capability was found to be very low.

Shared Contaminated Condensate Storage Tanks: The two contaminated condensate storage tanks at Quad Cities are normally cross-connected. The normal use of the CCST is makeup to the hotwell through the condensate transfer pumps and as the CRD, SSMP, and HPCI suctions. In an emergency, RHR and RCIC may be lined up to take a suction from the CCST.

Safe Shutdown Makeup Pump System: The Quad Cities station has installed an electric-motor-driven pump system as a backup to the RCIC system for certain fire scenarios. This pump can also be used during other plant upsets to inject into the reactor vessel. Due to the fire-related nature of the design, the system has redundant electrical supplies from each unit, can feed either unit, can be controlled fully from the control room or locally, and can draw from the fire protection system for suction on loss of the normal suction from the CCSTs. Even without power to the pump, fire protection water can be injected into the reactor vessel through the SSMP system piping, when reactor vessel pressure is sufficiently low.

TABLE 4.6.3-1
UNIQUE FEATURES OF QUAD CITIES STATION

UNIQUE FEATURE	CONTRIBUTION TO RISK PROFILE
Shared Service Water System	• Reduced frequency of Loss of SW Initiating Event • Reduced frequency of core damage
Shared Condensate Storage Tanks	• Reduced frequency of core damage
Safe Shutdown Makeup Pump System	• Reduced frequency of core damage

4.6.4 Decay Heat Removal Evaluation

This section provides an evaluation of the decay heat removal critical safety function at Quad Cities based upon the results from the IPE as required by Generic Letter 88-20. The purpose of the evaluation is to identify potential decay heat removal vulnerabilities for events initiated from power operation and to examine whether or not risks attributed to the loss of decay heat removal can be lowered in a cost-effective manner.

Decay heat removal during the first 24 hours following a plant trip is accomplished by the following key systems at Quad Cities:

- During most transient-type events (including small LOCA), decay heat is removed via the main condenser or suppression pool. High pressure injection systems, feedwater, HPCI, RCIC or SSMP, maintain RPV level while steam is bled to the main condenser via the turbine bypass valves (bleed and feed). If the main condenser is unavailable, as in the small LOCA case, the same systems will maintain RPV level, however, the steam produced by decay heat will be discharged through the relief valves to the torus. If high pressure injection sources fail, then the reactor is manually depressurized and the low pressure systems (LPCI mode of RHR and Core Spray) in conjunction with SPC are used. After the RPV pressure is below 100 psig, the shutdown cooling system can also be used to remove decay heat.

- During medium or large LOCA events (excluding ISLOCA) and inadvertent open relief valve (IORV) events, decay heat is removed directly by the low pressure systems and SPC. This includes the low pressure coolant injection (LPCI) mode of RHR with the RHR service water (RHRSW) system, the core spray (CS) pumps and the associated operator actions.

Given that successful decay heat removal depends upon the above systems and operations, the following is a discussion of these systems and their respective features. The decay heat removal functional failures that contribute to the core damage frequency (CDF) are discussed below:

- If the MSIVs remain open following a turbine trip, the reactor pressure will increase until the turbine-bypass valves open and steam is relieved to the main condenser. Reactor water level is maintained by the high pressure injection sources, FW, HPCI, RCIC or SSMP. If the bypass valves or the main condenser is unavailable, then the relief valves will automatically open and relieve steam directly to the suppression pool. During those transient events which do not result in loss of either turbine bypass to the main condenser or feedwater, decay heat can be removed by the condenser.

- HPCI or RCIC in conjunction with the relief valves can function in the feed and bleed mode even with a station blackout, provided that DC power is available. However, after recovery of AC onsite power, SPC with the low pressure systems would be required to prevent suppression pool overheating and loss of net positive suction head (NPSH) to the low pressure pumps. The SSMP is important during a single unit station blackout because it can be powered from either unit and can be aligned to inject to either unit. The SSMP system can also be aligned to allow the diesel driven fire system to inject directly into the RPV.

- If the high pressure injection systems fail, the reactor vessel may be depressurized to allow the low pressure injection systems to inject. This is accomplished by using the turbine bypass valves to depressurize to the main condenser (OAD2) or the relief valves of the Automatic Depressurization System (ADS) to depressurize to the torus (OAD1). Manual depressurization is not initiated in either case until the low pressure systems have been started and are recirculating through their respective minimum flow lines.

- The low pressure injection systems, LPCI mode of RHR and CS, can inject when reactor pressure is less than 325 psig. LPCI employs the four RHR pumps, each capable of providing full injection flow. CS consists of two pumps, each capable of providing full injection flow. These pumps provide a highly redundant means of supplying makeup to the reactor vessel from diverse sources through redundant and diverse injection points.

- If the decay heat is directed to the suppression pool through either the relief valves or the break (except for interfacing system LOCAs), the suppression pool must be cooled to maintain a minimum net positive suction head (NPSH) for the low pressure pumps. The heat is removed by SPC, using the RHR pumps, heat exchangers, and the RHRSW system. If heat is not removed by SPC, suction of the low pressure pumps can be aligned to the CCST or external sources of water can be used to obviate the dependence on NPSH from the suppression pool.

- The Control Rod Drive Hydraulic system (CRD) that also can be used to ensure long term core success. However, because of the relatively small flow rate of the CRD system (106 - 150 gpm), it is not adequate to remove all decay heat until several hours after the reactor is shutdown. Heatup of the torus can precipitate a depressurization of the RPV as well as lead to failure of low pressure pumps aligned to the torus. Because this heatup takes several hours to occur, the CRD system, even with its relatively low flow rate, is effective in preventing core damage following failure of other injection systems caused by torus heating.

In the top 100 core damage sequences, those initiated by a dual unit loss of offsite power (LOSP), a single unit LOSP, a general transient, or an ATWS are significant contributors. Less than 20% of the CDF was precipitated by failures of components or operator actions associated with the RHR system (LPA, LPB, OHX, RHRHX, OSPC, SPC, OSDC, SDC). Failures by the operators to correctly initiate long term heat removal (OHX, OSPC, OSDC) accounted for only about 1% of the CDF. It must be noted that the single unit and dual unit LOSP initiators lead to degraded support states which contribute to failures of RHR since, for many cases, only one train of RHR is available.

Quad Cities was found to have a low overall CDF with less than 20% of the CDF related to decay heat removal. Because of the low contribution of failure of decay heat removal, in conjunction with the fact that there were no specific vulnerabilities identified related to this function, additional effort to reduce the CDF attributable to decay heat removal failure is not required. It is considered that the evaluation of decay heat removal performed during this IPE provides adequate understanding of this vital function and Quad Cities' means for dealing with decay heat removal during accident conditions to resolve this generic issue.

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4.6.5 Evaluation of AM Endstates

SAM Endstates - The dominant Plant Response Tree sequences with SAM endstates from the IPE analysis have been reviewed and are shown in Table 4.6.5-1. Table 4.6.5-1 (located at the end of this subsection) individually lists the top 65 SAM sequences and provides the following information:

- Sequence number (if the SAM damage states were combined with core damage states in a new top 100 damage sequence listing).
- Accident sequence frequency
- Percent contribution to total core damage frequency
- Plant damage state (bin)
- Initiating event name and frequency
- Failed support state event tree nodes, probabilities, and descriptions
- Failed plant response tree nodes, probabilities, and descriptions

The core damage frequency for these 65 SAM sequences is 1.49E-06. This is 68.6% of the base core damage frequency of 2.169E-06. The total SAM damage state frequency (all SAM sequences) is 1.82E-06, or 84% of the base core damage frequency of 2.169E-06.

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
2	2.92E-07	7.31	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
4	1.48E-07	3.71	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
6	1.17E-07	2.92	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
8	9.99E-08	2.5	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
12	5.87E-08	1.47	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
14	5.74E-08	1.44	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				1IA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
17	5.06E-08	1.27	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				ROP1	1.00E+00	EVENT FAILS
18	4.92E-08	1.23	SAM	LB13	4.65E-03	LOSS OF BUS 13 IE
				13	1.00E+00	EVENT FAILS
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				1IA	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
19	4.34E-08	1.09	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				14	1.40E-03	LOSS OF BUS 14 (345KV UNAVAIL)
				1IA	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
20	4.28E-08	1.07	SAM	LOSW	9.10E-03	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)
				SW	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				PCSA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
22	3.09E-08	0.77	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
24	2.80E-08	0.7	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
26	2.56E-08	0.64	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
						CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
28	2.38E-08	0.6	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
29	2.14E-08	0.53	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				14	1.40E-03	LOSS OF BUS 14 (345KV UNAVAIL)
				LPA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
32	1.61E-08	0.4	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	9.60E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (19)
				ROP1	1.00E+00	EVENT FAILS
33	1.61E-08	0.4	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
34	1.57E-08	0.39	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				LPB	1.00E+00	EVENT FAILS
				RHRHX	9.16E-03	RHR HX FAILS/RHR A PUMP AVLBL; 13, 18, 1M1 AVAILABLE
				SSMP1	1.72E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
36	1.55E-08	0.39	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)
				DGB	1.12E-01	LOSS OF DG1/2 AFTER DG2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
37	1.51E-08	0.38	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
38	1.44E-08	0.36	SAM	MLOCA	8.00E-04	MLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SBCS	1.00E+00	EVENT FAILS
39	1.43E-08	0.36	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
43	1.21E-08	0.3	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
44	1.21E-08	0.3	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
45	1.14E-08	0.29	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
46	1.10E-08	0.27	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.02E-02	RHR B FAILS/A FLR; ALL SUP AVAIL (1LP-TR-2/1LP-TR1A)
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
47	1.06E-08	0.27	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
49	1.02E-08	0.26	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
50	9.65E-09	0.24	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
51	9.51E-09	0.24	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
52	9.44E-09	0.24	SAM	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				OSL1	8.70E-03	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (1)
				OSL2	5.60E-02	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)
53	9.39E-09	0.24	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				LPB	2.97E-03	RHR B FAILS/A SUCCESS; ALL SUPPORTS AVAILABLE
				RHRHX	9.16E-03	RHR HX FAILS/RHR A PUMP AVLBL; 13, 18, 1M1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
54	8.77E-09	0.22	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
56	8.15E-09	0.2	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
57	8.15E-09	0.2	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
				CS	5.24E-02	CS FAILS; 14-1, 19, 1R1 AVAILABLE
58	7.71E-09	0.19	SAM	LB13	4.65E-03	LOSS OF BUS 13 IE
				13	1.00E+00	EVENT FAILS
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				DG1	1.37E-01	LOP FROM DG1 TO BUS 14-1 (24 HRS)
				1IA	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
59	7.61E-09	0.19	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	9.60E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (19)
				ROP1	1.00E+00	EVENT FAILS
61	7.23E-09	0.18	SAM	LB13	4.65E-03	LOSS OF BUS 13 IE
				13	1.00E+00	EVENT FAILS
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				DGB	1.37E-01	LOP FROM DG1/2 (24HR)
				1IA	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				LPA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
66	6.37E-09	0.16	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				SW	1.00E-01	FAILURE OF SW (LOOP), 16 UNAVAIL
				1IA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
67	6.13E-09	0.15	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	9.60E-02	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (19)
				ROP1	1.00E+00	EVENT FAILS
68	6.10E-09	0.15	SAM	LB13	4.65E-03	LOSS OF BUS 13 IE
				13	1.00E+00	EVENT FAILS
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				1IA	1.00E+00	EVENT FAILS
				FW	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	1.00E+00	EVENT FAILS
				RCIC	1.01E-01	RCIC FAILS; ALL SUPPORTS AVAILABLE
				OSMP3	1.50E-01	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (13)
				CRD	1.00E+00	EVENT FAILS
71	5.83E-09	0.15	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				1IA	1.96E-02	IA FAILS (LOOP, DLOOP)
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
73	5.72E-09	0.14	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
76	5.63E-09	0.14	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				1IA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
77	5.54E-09	0.14	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
78	5.52E-09	0.14	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.02E-02	RHR B FAILS/A FLR; ALL SUP AVAIL (1LP-TR-2/1LP-TR1A)
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
79	5.43E-09	0.14	SAM	LB13	4.65E-03	LOSS OF BUS 13 IE
				13	1.00E+00	EVENT FAILS
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				SW	9.94E-02	FAILURE OF SW (TRANS), 13/14 OR 1B-1/16 UNAVAIL
				1IA	1.00E+00	EVENT FAILS
				PCSA	4.05E-01	POWER CONVERSION SYSTEM UNAVAILABLE
				FW	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				CRD	1.00E+00	EVENT FAILS
80	5.33E-09	0.13	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DG2	9.68E-02	LOSS OF DG2 AFTER DG1 (6 HRS)
				SBO?	1.00E+00	SBO IN UNIT 2, NO SBO IN UNIT 1
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				LPB	1.00E+00	EVENT FAILS
				SSMP1	1.72E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
81	5.32E-09	0.13	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)
				DGB	1.12E-01	LOSS OF DG1/2 AFTER DG2, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
84	5.16E-09	0.13	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				DG1	9.04E-02	LOP FROM DG1 TO BUS 14-1 (6 HRS)
				DGB	9.68E-02	LOSS OF DG1/2 AFTER DG1, (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
86	5.03E-09	0.13	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				1IA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
87	5.00E-09	0.13	SAM	ISLOCA	1.20E-07	INTERFACING SYSTEM LOCA IE
				CRV	4.31E-02	RHR SYSTEM RELIEF VALVES FAIL TO RECLOSE
				OIB	1.00E+00	OPTR FAILS TO ISOLATE THE BREAK (10)
88	4.91E-09	0.12	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				SW	4.59E-02	FAILURE OF SW (LOOP)
				1IA	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				LPB	3.10E-03	RHR B FAILS; 14-1, 1R1, 19 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
89	4.86E-09	0.12	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DG2	7.83E-02	LOP FROM DG2 TO BUS 24-1 (6 HRS)
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
90	4.86E-09	0.12	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
91	4.86E-09	0.12	SAM	MLOCA	8.00E-04	MLOCA IE
				14	2.80E-03	LOSS OF BUS 14 (345KV AVAIL)
				LPA	3.10E-03	RHR A FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	1.00E+00	EVENT FAILS
				SBCS	1.00E+00	EVENT FAILS
92	4.82E-09	0.12	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				14	1.40E-03	LOSS OF BUS 14 (345KV UNAVAIL)

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				SW	1.00E-01	FAILURE OF SW (LOOP), 16 UNAVAIL
				1IA	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
93	4.80E-09	0.12	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				OSMP1	1.40E-03	OPTR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)
				ROP1	1.00E+00	EVENT FAILS
94	4.79E-09	0.12	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	3.39E-04	RHR HX FAILS/RHR A/B PUMPS AVLBL; ALL SUPPORTS AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
95	4.76E-09	0.12	SAM	ATWS	1.16E-04	ATWS INITIATOR
				RCFM	3.33E-01	FRAC RPS FAILURES THAT ARE MECHANICAL
				SLC	2.35E-04	SLC FAILS; 1/2 PUMPS; ALL SUPPORTS AVAILABLE
96	4.73E-09	0.12	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				1TB	1.00E+00	EVENT FAILS
				LPB	2.97E-03	RHR B FAILS/A SUCCESS; ALL SUPPORTS AVAILABLE
				RHRHX	9.16E-03	RHR HX FAILS/RHR A PUMP AVLBL; 13, 18, 1M1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
97	4.68E-09	0.12	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				13	1.48E-03	LOSS OF BUS 13 (345KV UNAVAIL)
				14	1.17E-01	LOSS OF BUS 14 AFTER 13, 14-1 AVAIL
				1IA	1.00E+00	EVENT FAILS
				HP1	8.32E-02	HP FAILS; ALL SUPPORTS AVAILABLE
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
98	4.60E-09	0.12	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				DG1	9.68E-02	LOSS OF DG1 AFTER DG1/2, (6 HRS)
				14	1.40E-03	LOSS OF BUS 14 (345KV UNAVAIL)
				1IA	1.00E+00	EVENT FAILS
				LPA	1.00E+00	EVENT FAILS
				RHRHX	1.00E+00	EVENT FAILS
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
99	4.49E-09	0.11	SAM	LOOP	3.20E-02	LOSS OF OFFSITE POWER IN ONE UNIT
				14	1.40E-03	LOSS OF BUS 14 (345KV UNAVAIL)

TABLE 4.6.5-1
SUMMARY OF SAM ENDSTATES

Number (1)	Frequency (2)	Percent (3)	Damage State (4)	Event (5)	Value (6)	Description (7)
				1TB	1.00E+00	EVENT FAILS
				1IA	1.00E+00	EVENT FAILS
				RHRHX	9.16E-03	RHR HX FAILS/RHR A/B PUMPS AVLBL; 13, 18, 1M1 AVAILABLE
				SSMP1	1.60E-02	SSMP\CST FAILS; ALL SUPPORTS AVAILABLE
				ROP1	1.00E+00	EVENT FAILS
100	4.46E-09	0.11	SAM	DLOOP	1.61E-02	LOSS OF OFFSITE POWER IN BOTH UNITS
				DGB	9.03E-02	LOP FROM DG1/2 (6 HRS)
				SW	2.87E-02	FAILURE OF SW (DLOOP), 13 & 23 UNAVAIL
				LPA	1.00E+00	EVENT FAILS
				RHRHX	9.09E-03	RHR HX FAILS/RHR B PUMP AVLBL; 14, 19, 1R1 AVAILABLE
				SSMP1	1.71E-02	SSMP\CST FAILS (DLP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE
				ROP1	1.00E+00	EVENT FAILS

Notes:

1. "Number" refers to accident sequence ranking in the top 100 sequences.
2. "Frequency" is the frequency per year that this sequence is expected to occur.
3. "Percent" is the percent of total core damage represented by this single sequence.
4. "Damage State" is the plant damage state to which this sequence belongs. The fifth character presents the release associated with this type of sequence and is manually assigned at the end of the analysis in presentations of dominant sequences.
5. "Event" is the list of PRT and support system event tree top events which have failed in this sequence.
6. "Value" is frequency (for initiators) or probability (for failures) associated with each event.
7. "Description" defines the "Event" label.

CAM Endstates - The sequences having endstate designators with "B", "G" (except "GG"), or "H" as the fourth character are all sequences in which the containment remains intact throughout the initial 24 hours of the event. CAM sequences would be that subset of these sequences in which the containment conditions were trending toward failure.

Within the top 100 core damage sequences, there are 46 sequences having "B," "H," or "G" (except "GG") as their fourth character. These sequence types were investigated to determine whether the containment was pressurizing, heating up, or whether conditions were stable. Of the 46 containment success sequences, none have been found to trend towards failure in the second 24 hour period. No CAM sequences, therefore, were identified in the Quad Cities IPE analysis.

4.6.6 Evaluation of Large Early Release Frequency (LERF)

The large, early release frequency (referred to as LERF) as defined in the NEI-sponsored PSA Applications Guide (EPRI TR-105396) published in August 1995. ***Large*** means that the event results in "the rapid, unscrubbed release of airborne aerosol fission products to the environment." ***Early*** means that the event occurs "before the effective implementation of the off-site emergency response and protective actions." The PSA Applications Guide also includes a general guideline that an early release "Occurs Before or Within 4 Hours of Vessel Breach."

Based on this general guideline for early release and the MAAP analyses summarized in Table 4.5.5-3, only the TEEQ and TEER damage states among the top 100 dominant sequences contribute to early releases. Both are ATWS sequences with containment failure before vessel failure. For the TEEQ damage state, the containment failure occurs in the wetwell gas space; consequently, scrubbing action by the suppression pool is credited for the TEEQ damage state. For the TEER damage state, however, the containment failure occurs in the drywell and no scrubbing is credited. The environmental release fractions given in Table 4.5.5-3 for the TEEQ and TEER damage states illustrate the benefit of scrubbing. For these ATWS damage states, the main benefit of scrubbing is a large reduction in the release fractions for non-volatile fission products and Tellurium-based fission products.

Therefore, the only damage state among the top 100 dominant sequences that gives an unscrubbed, early release (and thus a contribution to LERF) is TEER. Other damage states included in the analysis also contribute to LERF, but those damage states together contribute less than 0.1% of the total CDF and therefore give an insignificant contribution to LERF. Based on the TEER data given in Table 4.6.2-4, therefore, the Modified IPE gives a LERF of 3.74E-8/yr, less than 2% of the total CDF value. This LERF value corresponds to approximately once in 27,000,000 years.

The TEEQ damage state falls outside the LERF definition because credit is given for scrubbing due to the location of the containment failure and because a scrubbed release does not meet the definition of ***Large*** given by the PSA Applications Guide. Nevertheless, as shown in Table 4.5.5-3, the volatile fission product release fraction (as represented by CsI and RbI) is significant, although some benefit of scrubbing is seen. The only damage state among the top 100 dominant sequences that gives a scrubbed, early release is TEEQ. Other damage states included in the analysis (but not contributing to the top 100 dominant sequences) also give a scrubbed, early release, but those damage states together contribute less than 0.1% of the total CDF. Therefore, based on the TEEQ damage state data given in Table 4.6.2-4, the Modified IPE gives a scrubbed, early release frequency of 1.25E-7/yr, approximately 5.78% of the total CDF value. This LERF value corresponds to approximately once in 8,000,000 years.

Combined, the TEEQ and TEER damage states have a frequency of 1.63E-07/yr for an early release, approximately 7.5% of the total CDF value. This LERF value corresponds to approximately once in 6,100,000 years.

> Note: The LERF definition discussed above was not available at the time of the orginal IPE Submittal Report. Several other types of release frequencies were reported. Those other types of release frequencies are not being used by ComEd for PSA applictions and, as a consequence, were not calculated as part of the Modified IPE.

4.7.2 Evaluation Against NUMARC Severe Accident Issue Closure Guidelines

The results of the Quad Cities IPE have been evaluated against the NUMARC Severe Accident Closure Guidelines. The guidelines were used to assess the proposed enhancements developed via insights related to severe accidents.

The first step in using the Severe Accident Closure Guidelines was to group the core damage sequences; the groupings used were those of Table B-1 of that document.

The grouping was carried out for all core damage sequences down to the quantification frequency cutoff of 1E-12 for a given sequence. The following groups contain some contribution to the total core damage frequency:

IA Accident sequences involving loss of coolant inventory makeup in which the reactor pressure remains high.

IB Accident sequences involving a loss of all AC power and loss of coolant inventory makeup (i.e., station blackout).

ID Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced.

II Accident sequences involving loss of containment heat removal leading to containment failure and subsequent loss of coolant inventory makeup.

IIIB Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized and inadequate coolant inventory makeup is available.

IIIC Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is at low pressure and inadequate coolant inventory makeup is available.

IV Accident sequences involving an ATWS leading to containment failure due to high pressure and subsequent loss of inventory makeup.

V Unisolated LOCA outside containment leading to loss of effective coolant inventory makeup.

The sequence numbers of the top 100 sequences included in each group are listed in Table 4.7.2-1 with the resulting mean group core damage frequency and percent contribution to the total core damage frequency. The group core damage frequency and contribution is based upon all sequences.

TABLE 4.7.2-1
NUMARC SEVERE ACCIDENT CLOSURE GUIDELINES
SEQUENCE GROUPING INFORMATION

Sequence Group	Sequence Numbers [1]	Total Group Core Damage Frequency	Percent Contribution to Total CDF
IA	3, 5, 7, 8, 9, 11, 12, 13, 16, 18, 19, 20, 21, 22, 24, 26, 27, 32, 33, 35, 37, 39, 40, 42, 44, 46, 51, 54, 55, 56, 62, 65, 67, 69, 70, 71, 73, 77, 78, 79, 82, 83, 85, 86, 87, 88, 94, 96	7.0E-07	32%
IB	1, 2, 6, 47, 58, 75, 76, 100	1.1E-06	51%
ID	34, 41, 64, 68, 90, 91, 95, 97, 98, 99	3.4E-08	2%
II	38, 49, 52, 74, 81, 92	2.4E-08	1%
IIIB	4, 60, 61, 93	1.1E-07	5%
IIIC	29, 31, 45, 66, 72, 89	2.3E-08	1%
IV	10, 14, 15, 17, 23, 25, 28, 30, 36, 43, 48, 50, 53, 57, 59, 63, 80, 84	1.6E-07	8%
V	(None in the top 100 sequences)	2.5E-10	<<1%

[1] Refers to the sequence position in the ranking of core damage sequences in descending magnitude of core damage frequency.

The core damage frequency and percent contribution to the total core damage frequency for each group were then evaluated against Tables 1 and 2 of the Severe Accident Closure Guidelines. Table 2 was used for the containment bypass sequences (group V only), and Table 1 was used for all other groups. With the exception of the Class IB sequences, the comparison shows all the accident sequence groups fall below the Severe Accident Closure Guidelines frequency cutoffs.

The Class IB sequences (Station Blackout) exceed the percentage cutoff value specified on Table 1 and just exceed the lowest frequency limit requiring action. Recently, additional diesel generators have been added at the station; one new diesel generator has been added at each unit. These diesel generators will significantly reduce the likelihood of station blackout and will be included in an update to the Quad Cities PRA model.

5.0 ACCIDENT MANAGEMENT

Commonwealth Edison (CECo) has integrated the development of an Accident Management (AM) Program with the performance of the IPE. This process was used in the belief that an effective approach to AM is a structured evaluation process that is tightly coupled with the knowledge base developed during the IPE program. Potential and possibly subtle strategies and insights are best identified and documented while related information is actively under evaluation by the IPE analysts.

The CECo AM program development and considerations for future direction are described in this section. The AM insights derived from the Quad Cities IPE will be evaluated in conjunction with industry and owners group guidance during implementation of the Nuclear Energy Institute (NEI) Formal Industry Position on Severe Accident Management (SAM).

It should be noted that CECo, in common with other nuclear plant operators, has an accident management program already in place. Such a program has existed almost from the inception of the industry. Within CECo, the current program consists of two parts. The first of these are the plant General Abnormal Procedures (QGAs), the trained personnel who implement them, and the plant staff who support that implementation. For events which may proceed beyond the design bases, this first phase is supported by one which centers around the Generating Stations Emergency Plan (GSEP), the trained respondents, and the facilities and tools to implement that plan.

CECo believes that the management of severe accidents with potential or actual core damage, where the situation is beyond the realm of the QGAs, should primarily be the responsibility of the emergency response organization outside the Control Room. The CECo AM program is being developed with this philosophy.

5.1 Introduction to CECo's Accident Management Program

The AM program is part of CECo's overall effort to identify, evaluate, and resolve severe accident issues with emphasis on 1) the prevention of such accidents, and 2) the need to respond effectively in the unlikely event of an accident. The objective of developing the input to the AM program in parallel with the performance of the IPE is to prevent the potential loss of valuable information that might occur if observations are not elicited and documented at the same time system performance, system interactions, operator actions, and accident progression are under study by the IPE analyst. Improved understanding of the plant capability to respond to accidents and the operator response to accident symptoms is one of the most important benefits to be obtained from the Quad Cities IPE, and the decision to develop and evaluate AM insights as part of the IPE for Quad Cities has maximized this benefit.

The CECo process used to define the AM Program for Quad Cities is comparable to those developed by EPRI (Reference 5.1) and the NRC (Reference 5.2). Both the EPRI and the NRC methodologies for developing AM programs are based on an evaluation of the IPE results to identify accident management activities. The methodology used by CECo for Quad Cities is a forward looking process based on evaluation of each phase of the IPE work.

The CECo approach encompasses the key aspects of the EPRI and NRC methodologies and employs a simultaneous "top-down" and "bottom-up" method as illustrated in Figure 5.1-1. The top-down evaluation has logically defined the elements of an intuitive AM program framework and identified where the various aspects of the IPE effort could support enhancement of these elements. The bottom-up approach examined the technical analysis at each of the major steps of the IPE for observations that could fall into one or more of the five AM framework elements.

The CECo AM framework elements are similar to those proposed by the NRC. The five elements of the CECo AM program are:

- Organization and Decision Making,
- Accident Management Guidance (Strategies),
- Calculational Tools,
- Training, and
- Plant Status Information

A major feature of the CECo framework is the broadening of the element definition to include more information and/or organizational ties within the framework. Differences include the expansion of the plant instrumentation area to include vital plant information needs for AM, the expansion of AM guidance to include the interface with the site emergency plan, and the consideration of predictive and decision making tools within the calculational tool element as necessary to meet BWR Owners Group Severe Accident Management guideline implementation requirements.

The individual insights identified by the bottom-up approach were then evaluated on their technical merit. A qualitative assessment of their potential benefit, as well as potential impact, was also performed. Since individual insights related to the same plant feature can be contradictory or supplementary due to the nature of the insight identification process, all insights related to a given plant feature were then evaluated on an aggregate basis. A qualitative assessment was then performed for each set of combined insights. In some cases, a quantitative assessment of the benefits, in terms of improvements in the Quad Cities severe accident risk indicators (core damage frequency, or source term characteristics), was performed using the IPE model.

This process is compatible with the AM guidelines developed by the industry. However, the approach is believed to have two distinct advantages. First, the search for insights is conducted while the analysts have the information fresh in their minds. Second, the search covers all aspects of the analysis, not just the dominant accident sequences. Thus, the approach supports an AM program that takes maximum advantage of the performance of the IPE for Quad Cities.

It was recognized that the insights derived from a PRA might not fully address the human side of an AM program. To consider this aspect, a task analysis of key GSEP positions was conducted by a behavioral scientist. This task analysis concentrated on three of the AM framework elements: organization, training, and plant status information. CECo will utilize the Emergency Procedure Guideline (EPG) action placement criteria developed by the BWR Owners Group Accident Management Working Group (AMWG) as a screening process to identify the optimum location of such task performance.

The AM program will also consider industry activities such as the development of "Accident Management Technical Basis" by the Electric Power Research Institute (EPRI), and NEI Formal Industry Position guidance.

5.2 The Accident Management Insights Process

Insights are those observations regarding station configuration or practices suggested by the IPE which may affect the risk profile of the plant. Insights can suggest changes to enhance the capability of the plant and its operators to respond to an initiating event to either prevent core damage or to mitigate the consequences of core damage. The IPE Insights (see Section 4.7.1) address the capability of the existing plant to respond to an initiating event. The Accident Management Insights deal with enhancements to the capability of the plant emergency response organization to respond to an accident situation, given that it has occurred. The dividing line between IPE insights and AM insights is not sharp; a distinction is made only to attempt to provide two broad categories of insights which go beyond the normal (i.e., traditional) IPE thought processes.

In order to identify possible AM activities by each IPE analyst, it was necessary to develop structured guidance. The development of the guidance began with the definition of a logical and intuitive process for features which can impact the severe accident risk profile (Reference 5.3). The process identified the framework for insights which could be expected, including: AM strategies, calculational tools to carry out the strategies, the emergency response organization, AM training, and information (including plant status information) required to carry out AM activities. These broad features were then correlated to the IPE functional work products to define the types of AM insights which could be obtained from each task of the IPE analyses, as presented in Figure 5.2-1.

This detailed correlation of possible plant insights vs. IPE work products was used to define a set of questions for each IPE task which would focus and stimulate the IPE analysts to identify applicable AM insights as the IPE tasks were being performed. A sample set of questions for one of the elements of the AM Insights Development Matrix (Figure 5.2-1) is shown in Figure 5.2-2.

Each of the individual insights, identified by the IPE analysts, was evaluated by a "Tiger Team", composed of individuals from CECo and the IPE Partnership. The Team included personnel knowledgeable in Quad Cities plant systems and operation, severe accident phenomena, emergency procedures, and the emergency response organization. The role of the review team was to evaluate the insights from a broad perspective and to make technical assessments of their potential benefit and impact. (See Figure 5.2-3 for an example of the documentation and evaluation of one of the insights from the program.) The results of the evaluation team's activities form the basis for CECo contributions to industry and owners group AM activities.

Observations from the evaluation of the AM insights are summarized below. The AM insights judged to have generic procedure implications were provided to the BWR Owners Group for further evaluation. However, due to the 1991 BWR Owners Group generic review cut-off date, these insights were not explicitly evaluated. The Quad Cities-specific implementation of the BWR Owners Group generic SAM guidance will include disposition of the Quad Cities insights. See Paragraph 5.3.6.

5.3.1 Organization and Decision Making

CECo has evaluated the organizational needs for accident management using three approaches:

- The first involved examination of the results of AM insight development and assessment effort described above.
- The second approach involved the use of a top down, or logical intuitive, AM examination by experienced personnel.
- The third, noted earlier, involved a task analyses, performed by a behavioral scientist, of the key GSEP positions.

<u>Current Organization</u>

CECo's current organization and decision making structure is built around the plant staff for events which lie within the design bases of the plant, and the emergency plan staff for AM activities for events which pass beyond that envelope. The GSEP command, control, and communications channels are well established within each of these areas and at the interfaces between areas. Moreover, those same lines between CECo and outside agencies are equally well defined and established.

It has been determined that the existing GSEP organizational structure and delineation of decision-making responsibility for AM activities are both clear and well-understood.

<u>AM Recommendations on Current Organization</u>

As noted earlier, it is CECo's philosophy that AM should primarily be the responsibility of the GSEP organization, not limited to the Control Room staff. Our evaluation suggests that AM activities are best addressed by the established Technical Support Center (TSC) organization. AM recommendations include, for example, specifying the role of the TSC personnel, upon initial manning, to conduct a review of the accident progression up to that time and to verify the status of all QGA-required mitigative actions. The EPG action placement criteria developed by the BWR Owners Group to optimize the location of the EPG action (CR vs. TSC) will be utilized in this determination. CECo further endorses the full complement of NEI Formal Industry Position recommendations to ensure the TSC personnel have the calculational and diagnostic tools available to determine the progression of the accident and to monitor the status of mitigative plant features. Consistent with the NEI Formal Industry Position on SAM, implementation of SAM will be achieved within the constraints of existing personnel and plant hardware.

Additionally, to maintain technical awareness and expertise in severe accident state-of-the-art, a corporate resource is suggested. This resource would supplement the GSEP organization as needed, similar to the support provided by any corporate organization during an emergency condition. The corporate group would follow severe accident research, evaluate new information, interact with the technical community on severe accident issues, and maintain the technical aspects of the program at a level consistent with the state-of-the-art in severe accident technology.

adverse impact of the loss of the suppression pool as a heat sink or as a fission product removal mechanism during wetwell venting. However, consistent with the NEI Formal Industry Position on SAM, no plant modifications will be made such as the RPV support skirt venting.

An AM insight was also formulated based upon severe accident experiments that investigated the efficacy of decay heat removal given debris slumping into the lower plenum by a water spray on the outside surface of the lower plenum. The presence of a support skirt for the Quad Cities design was experimentally investigated and it was determined that the use of a spray avoided the need for vent holes in the support skirt. The measured through-wall heat flux for an externally sprayed RPV lower head with skirt was large enough to remove the fraction of decay heat which would be directed through the thick walled lower plenum. The timely cooling of the RPV wall could be established immediately upon spray initiation and would not be dependent upon the time interval required to flood the containment or a modified pedestal. The ex-vessel spray system would be provided with an external source of water located outside the containment. In addition to providing the source of cooling water for ex-vessel spray and decay heat removal such a flow path would provide a means of establishing a flooded containment condition that did not require injection through the RPV as is the case for the existing plant configuration.

CECo has evaluated the benefits of implementing the above described external vessel cooling strategies utilizing the CECo Societal Risk Analysis Model. The conclusion reached is that the level of avoided risk does not justify implementation of such modifications at this time. As a result, CECo is not actively pursuing implementation of external vessel cooling for either pedestal flooding or external vessel spray strategies.

<u>Alternate Sources for Containment Spray for Source Term Reduction</u>

Two classes of accidents assessed in the Quad Cities IPE indicate the potential for large fission product release fractions. These classes are an unrecovered station blackout and an ATWS event. Accident management strategies which employ the drywell sprays would be an effective means of controlling the fission product release fractions. For a station blackout at a given unit, alternate sources of containment spraying would include the cross-connection to the other unit's RHR system or to the plant's fire protection system. The operation of the drywell sprays even in an intermittent manner will effectively remove the fission products from the containment gas space. Additionally, containment spray operation will remove energy from the gas space such that both the drywell temperature and pressure would be reduced and thereby delay containment failure. A delay of containment failure provides additional time for the recovery of AC power. The operation of the containment sprays also provide cooling which would retard the revaporization of fission products deposited on the containment surfaces. The spray water would flood the core debris accumulated on the drywell and pedestal floor and would help to quench it and thereby terminate core concrete interactions. The water layer overlying the debris bed would also scrub fission products that could be evolved from the debris bed during its quenching and stabilization.

5.3.3 Calculational Tools

CECo currently has a number of post-accident instruments to aid the GSEP organization for the current scope of accident management. It was recognized during the IPE/AM insight process that development of computational aids or tools would be desirable to aid the emergency response organizations in being able to monitor and forecast accident progression. In particular, the ability to predict changes in fission product releases and to predict the impact of various AM guidance alternatives is important in selecting optimal AM actions. CECo will implement the recently developed BWR Owners Group generic Severe Accident Guidelines (SAG) and recommended changes to the EPGs consistent with the NEI Formal Industry Position on SAM.

5.3.4 Training

CECo, like other utilities, conducts training relative to accident management through the QGA training provided to licensed personnel, through the annual training given to GSEP respondents, and through the conduct of GSEP exercises.

The industry IPE effort is yielding a number of insights which may lead directly to enhancements in the QGAs. Training on these potential QGA changes is expected to be integrated into the normal license training process.

For accident management, the positions expected to require additional training generally involve those respondents who are active in the accident recovery and mitigation decision making process, primarily located in the TSC organization. The degree of training ranges from general familiarity with AM to that associated with obtaining a degree of expertise in severe accident behavior and the calculations necessary to monitor and understand accident progression.

It is recognized that the area of training for the SAM guidance under consideration will be a first time effort. The ability to effectively assimilate and use this new SAM guidance will evolve over time to address industry and regulatory initiatives, in addition to individual utility activities. CECo will follow NEI guidance in the regard.

5.3.5 AM Information

The IPE/AM insight process identified areas where development of supplemental information regarding severe accidents might be useful to AM personnel. Examples include:

- The preferential use of Core Spray (over core injection) for debris cooling.
- Heat removal capabilities and limitations for alternate residual heat removal systems.
- The critical QGA steps required for successful core cooling, containment heat removal, and fission product mitigation.
- Minimum equipment needs for core cooling and containment heat removal, based on IPE success criteria.

5.3.6 Insight Implementation

The preparation of IPEs for nuclear plants prior to December 1991, in conjunction with industry research, provided additional insights that were incorporated into the recently developed BWR Owners Group generic Severe Accident Guidelines (SAG) and recommended changes to the EPGs. After completion of the initial Quad Cities IPE, the IPE/AM insights identified during the Quad Cities insight process were provided to the GE BWR Owners Group for consideration in the development of the Owners Group severe AM guidance. However, due to the 1991 BWR Owners Group generic review cut-off date, these insights were not explicitly evaluated. The Quad Cities-specific implementation of the BWR Owners Group generic guidance will include the disposition of the Quad Cities insights.

6.0 SUMMARY AND CONCLUSIONS

6.1 Summary of Modified IPE Results

The core damage frequency for the Quad Cities Modified IPE model was calculated to be 2.17E-06. The core damage frequency for each of the initiating events for the Modified IPE model is shown in Table 6.1-1.

The sequences that individually contribute more than 0.5% to the core damage frequency for the Quad Cities Modified IPE model are shown in Table 6.1-2. These top 20 sequences contribute approximately 86% to the total core damage frequency.

Table 6.1.3, which is based upon the top 100 sequences, provides the containment performance probability given that core damage has occurred. This table indicates a 33% probability of intact containment (no venting or failure) and a 67% probability of containment failure associated with releases of volatile fission products.

TABLE 6.1-1
SUMMARY OF IPE RESULTS FOR INITIATING EVENTS

INITIATING EVENT	CORE DAMAGE FREQUENCY (per year)	PERCENT CONTRIBUTION
Dual Unit LOOP	1.23E-06	56.67%
Single Unit LOOP	2.51E-07	11.58%
General Transient	2.34E-07	10.80%
ATWS	1.66E-07	7.68%
Medium LOCA	1.12E-07	5.16%
Loss of 125 VDC Bus 1B-1	7.44E-08	3.43%
Loss of Service Water	6.49E-08	2.99%
Large LOCA	2.10E-08	0.97%
Loss of Bus 13	5.19E-09	0.24%
Loss of Instrument Air	4.03E-09	0.19%
Small LOCA	2.56E-09	0.12%
IORV	1.23E-09	0.06%
Loss of Bus 14	8.15E-10	0.04%
Loss of Bus 18	6.79E-10	0.03%
Loss of MCC 18-2	4.08E-10	0.02%
Loss of Bus 11	3.30E-10	0.02%
Loss of Bus 12	3.06E-10	0.01%
ISLOCA	2.64E-10	0.01%
TOTAL	2.17E-06	100%

TABLE 6.1-2
SUMMARY OF IPE RESULTS

SEQUENCE DESCRIPTION		MODIFIED IPE MODEL		
EVENT	EVENT DESCRIPTION	SEQ. #.	FREQUENCY	PERCENT
DLOOP	LOSS OF OFFSITE POWER IN BOTH UNITS	1	8.62E-07	39.8
DG1	LOP FROM DG1 TO BUS 14-1 (6 HRS)			
DG2	LOSS OF DG2 AFTER DG1 (6 HRS)			
DGB	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)			
SBO?	SBO IN UNIT 1, SBO IN UNIT 2			
ROP2	FAILURE TO REC OSP			
DLOOP	LOSS OF OFFSITE POWER IN BOTH UNITS	2	1.56E-07	7.2
DG1	LOP FROM DG1 TO BUS 14-1 (6 HRS)			
DG2	LOSS OF DG2 AFTER DG1 (6 HRS)			
DGB	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)			
SBO?	SBO IN UNIT 1, SBO IN UNIT 2			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
RCIC	RCIC FAILS; ALL SUPPORTS AVAILABLE			
ROP2	FAILURE TO REC OSP;SBO,SHORT TIME AVAILABLE			
LOOP	LOSS OF OFFSITE POWER IN ONE UNIT	3	1.46E-07	6.72
1TB	EVENT FAILS			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
RCIC	RCIC FAILS; ALL SUPPORTS AVAILABLE			
OSMP3	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)			
OAD1	OPTR FAILS TO INITIATE ADS (12)			
MLOCA	MLOCA IE	4	1.03E-07	4.74
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
OAD1	OPTR FAILS TO INITIATE ADS (10)			
GTR	GENERAL TRANSIENT IE	5	8.75E-08	4.03
OFW1	OPTR FAILS TO RESTART A FW PUMP (2)			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
RCIC	RCIC FAILS; ALL SUPPORTS AVAILABLE			
OSMP3	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)			
OAD1	OPTR FAILS TO INITIATE ADS (12)			
DLOOP	LOSS OF OFFSITE POWER IN BOTH UNITS	6	7.04E-08	3.25
DG1	LOP FROM DG1 TO BUS 14-1 (6 HRS)			
DG2	LOSS OF DG2 AFTER DG1 (6 HRS)			
DGB	LOSS OF DG1/2 AFTER DG1 AND DG2, (6 HRS)			
SBO?	SBO IN UNIT 1, SBO IN UNIT 2			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
ROP2	FAILURE TO REC OSP			
DLOOP	LOSS OF OFFSITE POWER IN BOTH UNITS	7	6.84E-08	3.15
1TB	EVENT FAILS			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
RCIC	RCIC FAILS; ALL SUPPORTS AVAILABLE			
OSMP3	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)			
OAD1	OPTR FAILS TO INITIATE ADS (12)			
GTR	GENERAL TRANSIENT IE	8	5.79E-08	2.67
PCSA	POWER CONVERSION SYSTEM UNAVAILABLE			
OFW1	OPTR FAILS TO RESTART A FW PUMP (2)			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
RCIC	RCIC FAILS; ALL SUPPORTS AVAILABLE			
OSMP3	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)			
OAD1	OPTR FAILS TO INITIATE ADS (12)			

TABLE 6.1-2 (Continued)
SUMMARY OF IPE RESULTS

SEQUENCE DESCRIPTION		MODIFIED IPE MODEL		
EVENT	EVENT DESCRIPTION	SEQ. #.	FREQUENCY	PERCENT
LOSW	LOSS OF SERVICE WATER IE (INCL LOIA CONT.)	9	5.35E-08	2.47
SW	EVENT FAILS			
1IA	EVENT FAILS			
PCSA	EVENT FAILS			
FW	EVENT FAILS			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
RCIC	RCIC FAILS; ALL SUPPORTS AVAILABLE			
OSMP3	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)			
CRD	EVENT FAILS			
OAD1	OPTR FAILS TO INITIATE ADS (12)			
ATWS	ATWS INITIATOR	10	5.22E-08	2.41
RCFM	FRAC RPS FAILURES THAT ARE MECHANICAL			
OIADS	OPTR FAILS TO INHIBIT ADS (26)			
L1B1	LOSS OF 125VDC BUS 1B-1 IE	11	4.08E-08	1.88
1M1	LOSS OF 125VDC TB MAIN BUS 1A			
1R1	EVENT FAILS			
FW	EVENT FAILS			
HP1	EVENT FAILS			
LPA	EVENT FAILS			
LPB	EVENT FAILS			
RCIC	EVENT FAILS			
SSMP1	EVENT FAILS			
ADS	EVENT FAILS			
CS	EVENT FAILS			
GTR	GENERAL TRANSIENT IE	12	2.85E-08	1.31
FW	FW FAILS; ALL SUPPORTS AVAILABLE			
HP1	HP FAILS; ALL SUPPORTS AVAILABLE			
RCIC	RCIC FAILS; ALL SUPPORTS AVAILABLE			
OSMP3	OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)			
OAD1	OPTR FAILS TO INITIATE ADS (12)			
L1B1	LOSS OF 125VDC BUS 1B-1 IE	13	2.72E-08	1.25
1M1	LOSS OF 125VDC TB MAIN BUS 1A			
1R1	EVENT FAILS			
PCSA	POWER CONVERSION SYSTEM UNAVAILABLE			
FW	EVENT FAILS			
HP1	EVENT FAILS			
LPA	EVENT FAILS			
LPB	EVENT FAILS			
RCIC	EVENT FAILS			
SSMP1	EVENT FAILS			
ADS	EVENT FAILS			
CS	EVENT FAILS			
ATWS	ATWS INITIATOR	14	2.43E-08	1.12
MC	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS			
RCFM	FRAC RPS FAILURES THAT ARE MECHANICAL			
OIADS	OPTR FAILS TO INHIBIT ADS (26)			
ATWS	ATWS INITIATOR	15	2.04E-08	0.94
MC	MAIN COND FAILS (GIVEN FW SUCCESS) AFTER ATWS			
RCFM	FRAC RPS FAILURES THAT ARE MECHANICAL			
OSL1	OPTR FAILS TO INITIATE SLC (1/2 PUMP REQ) (17)			
OSL2	OPTR FAILS TO INITIATE SLC (2/2 PUMPS REQ) (16)			

TABLE 6.1-2 (Continued)
SUMMARY OF IPE RESULTS

SEQUENCE DESCRIPTION		MODIFIED IPE MODEL		
EVENT	EVENT DESCRIPTION	SEQ. #.	FREQUENCY	PERCENT
GTR PCSA FW HP1 RCIC OSMP3 OAD1	GENERAL TRANSIENT IE POWER CONVERSION SYSTEM UNAVAILABLE FW FAILS; ALL SUPPORTS AVAILABLE HP FAILS; ALL SUPPORTS AVAILABLE RCIC FAILS; ALL SUPPORTS AVAILABLE OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9) OPTR FAILS TO INITIATE ADS (12)	16	1.89E-08	0.87
ATWS RCFM OIADS WW/DW	ATWS INITIATOR FRAC RPS FAILURES THAT ARE MECHANICAL OPTR FAILS TO INHIBIT ADS (26) FRAC OF CONT FLRS IN DW (VS. WW)	17	1.56E-08	0.72
LOOP DG1 1TB HP1 RCIC OSMP3 OAD1	LOSS OF OFFSITE POWER IN ONE UNIT LOP FROM DG1 TO BUS 14-1 (6 HRS) EVENT FAILS HP FAILS; ALL SUPPORTS AVAILABLE RCIC FAILS; ALL SUPPORTS AVAILABLE OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9) OPTR FAILS TO INITIATE ADS (12)	18	1.43E-08	0.66
LOOP DGB 1TB HP1 LPA RCIC OSMP3 OAD1	LOSS OF OFFSITE POWER IN ONE UNIT LOP FROM DG1/2 (6 HRS) EVENT FAILS HP FAILS; ALL SUPPORTS AVAILABLE EVENT FAILS RCIC FAILS; ALL SUPPORTS AVAILABLE OPTR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9) OPTR FAILS TO INITIATE ADS (12)	19	1.35E-08	0.62
GTR 1M1 1R1 FW HP1 LPA LPB RCIC SSMP1 ADS CS	GENERAL TRANSIENT IE LOSS OF 125VDC TB MAIN BUS 1A LOSS OF 125VDC TB RESERVE BUS 1B-1 EVENT FAILS EVENT FAILS EVENT FAILS EVENT FAILS EVENT FAILS EVENT FAILS EVENT FAILS EVENT FAILS	20	1.31E-08	0.6

TABLE 6.1-3
CONTAINMENT STATUS - SUMMARY

Release Category[1]	Description	Frequency[2]	P(ES/CD)[3]
S	Less than 50% noble gases, less than 0.1% CsI, less than 1% Te_2/TeO_2, and less than 0.1% SrO; containment success (leakage only)	6.73E-07	0.327
A	Greater than 50% noble gases, less than 0.1% CsI, less than 1% Te_2/TeO_2, and less than 0.1% SrO; includes sequences not expected to fail containment in 24 hours but requiring actions in the 24-48 hour time period to avoid containment failure.	8.03E-09	0.004
B	Greater than 50% noble gases, less than 1% CsI, less than 1% Te_2/TeO_2, and less than 0.1% SrO	1.07E-07	0.052
C	Greater than 50% noble gases, less than 10% CsI, less than 1% Te_2/TeO_2, and less than 0.1% SrO	1.27E-07	0.059
D	Greater than 50% noble gases, less than 10% CsI, greater than 1% Te_2/TeO_2, and less than 0.1% SrO	2.52E-09	0.001
F	Greater than 50% noble gases, greater than 10% CsI, greater than 1% Te_2/TeO_2, and less than 0.1% SrO	1.13E-06	0.546
G	Greater than 50% noble gases, greater than 10% CsI, greater than 1% Te_2/TeO_2, and greater than 0.1% SrO	1.96E-08	0.010

1 Fifth endstate designator for core damage events.

2 The frequencies in this table are based upon the top 100 accident sequences.

3 Conditional probability of endstate given core damage.

6.2 Conclusions

The Modified IPE for Quad Cities Station, as presented in this report, models the baseline response of the station to a wide range of initiating events based on the plant design, operating practices and emergency response procedures that were existing at the middle of 1991. Important operator actions from the original IPE models were updated to reflect operating procedures in place as of December 1995. Further, the Modified IPE contains updated failure data for five key plant systems (HPCI, RCIC, Emergency Diesel Generators, Safe Shutdown Makeup Pump, and RHR) for the three year period of 1993 through 1995. Maintenance unavailability data for these same five systems was collected for the most recent two year interval (1994 and 1995). These time spans were judged to provide the best indication of recent performance for these key systems.

The IPE models and results are based on realistic predictions of the plant and operator response to accident initiators. The predictions are based on best estimate models of the plant response, system and operator action success criteria, and a comprehensive modeling of Quad Cities emergency procedures and abnormal operating procedures. The study includes calculations of realistic timing, a plant-specific evaluation of plant and common cause data, and a human reliability analysis that incorporates the steps in the emergency procedures. Evaluation of severe-accident phenomena is based on Quad Cities-specific features and capacities. It is believed that the CECo IPE process is the first probabilistic safety analysis process which has employed this degree of realism throughout all phases of the study and the first attempt to comprehensively model the emergency and abnormal operating procedures. Several other new approaches have been used in the Quad Cities IPE:

1. explicit and simultaneous performance of a probabilistic safety assessment in conjunction with defining an Accident Management Program, and

2. complete integration of the traditional Level 1 and Level 2 portions of the study by use of Plant Response Trees (PRTs).

The study shows that the capability of the Quad Cities Station design is very good. Quad Cities' systems are effective, and they have redundancy and diversity, as well as excess capacity. There is no critical dependence of the plant on instrument air or HVAC. The dual-unit site is beneficial, providing electric power cross-ties to the other unit, and providing shared service water. Torus water, DC power, and contaminated condensate storage tank inventory are also available from the other unit.

The BWR Owners Group-based emergency procedures are effective in responding to severe accidents. Although the study found a number of small potential emergency procedure improvements, the Modified IPE HRA indicated that no emergency procedure vulnerabilities were found. This result is supported by the sensitivity analysis discussed in

Section 4.5.4 in which key human error probabilities were reduced by an order of magnitude resulting in only a very small decrease in core damage frequency.

The IPE investigation has resulted in an accurate portrayal of current risk, as well as the development of a model which can be used with a high degree of confidence to address any number of future issues regarding core damage accidents, including the development of a comprehensive Accident Management Program for Quad Cities Station.

The core damage frequency was determined to be 2.17E-06 per year. Quad Cities Station is somewhat sensitive to one particular initiating event, Loss of Offsite Power in Both Units. Of the total core damage frequency, over 91% is spread over five initiating events; the Loss of Offsite Power in Both Units contributes 57% toward this total. The next four types of events are Single Unit Loss of Offsite Power (12%), General Transient (11%), ATWS (8%) and Medium LOCA (5%).

The frequency of the most likely sequence, a Loss of Offsite Power in Both Units with subsequent failure of all onsite AC power (station blackout) in both units, is 8.62E-07 per year; this constitutes about 40% to the total core damage frequency. The next most likely sequence is also a DLOOP sequence. This sequence has failure of high pressure injection and contributes about 7% to the core damage frequency (1.6E-07 per year). The third most likely sequence, a single unit LOOP with failure of high pressure injection and failure of operator action to depressurize and to allow low pressure injection, contributes about 7% to overall CDF. Sequence 4 is a Medium LOCA with failure of HPCI and failure of operator action to depressurize and allow low pressure injection. This sequence contributes about 5% to CDF. Sequence 5 is a general transient with failure of high pressure injection and operator action to depressurize and contributes about 4% to CDF. The next two most likely sequence are DLOOP sequences that each contribute about 3% to CDF.

A perusal of the results using the Fussell-Vesely importance measures indicates that the most significant hardware contributors toward total core damage frequency are failures of the diesel generators. The quantitative importance of emergency AC power sources is influenced significantly by the dependency of the plant on electrically-driven systems for long-term decay heat removal. The most significant operator-related contributions result from the failure to depressurize the reactor vessel when required and failure to initiate the SSMP. Of the three cases modeled for operator action to initiate the SSMP (i.e., initiation with suction from the CCST, initiation with suction from the fire system, and initiation with suction from the CCST and an ECCS signal present), the most important is initiation from the CCST with an ECCS signal present. Based on the low overall core damage frequency, these actions do not represent a plant vulnerability. Nonetheless, IPE and AM insights have been identified to reduce the impact of these failures and to provide accident management guidance to emergency response organizations to ensure these important actions are achieved.

A review of the modified IPE results against NUMARC Severe Accident Issue Closure Guidelines (NUMARC 91-04, January, 1992), reveals that with the exception of the Class IB sequences, all the accident sequence groups fall below the Severe Accident Closure Guidelines frequency cutoffs.

The Class IB sequences (Station Blackout) exceed the percentage cutoff value specified by the guidelines and just exceed the lowest frequency limit requiring action. Recently, additional diesel generators have been added at the station; one new diesel generator has been added at each unit. These diesel generators will significantly reduce the likelihood of station blackout and will be included in an update to the Quad Cities PRA model.

A class of accident sequences have been identified in this IPE study which are termed SAMs (Success with Accident Management). The SAM accident sequences do not result in core damage within the first 24 hours of the initiating event, but require some additional operator action after the first 24 hours to achieve a long term safe, stable state. Consistent with traditional PRA philosophy, these have not been classified as core damage sequences in the IPE results. The SAM sequences have a predicted frequency of occurrence of 1.82E-06 per year. The SAM accident sequences identified by the Quad Cities IPE are primarily loss of offsite power sequences (in one or both units) in which decay heat removal ultimately fails. In these sequences actions are required after the initial 24 hours to ensure that long-term core cooling can be maintained.

Only one sequence in the top 100 sequences of the modified IPE resulted in a large early release frequency (LERF), as defined in the NEI-sponsored PSA Applications Guide (EPRI TR-105396). This sequence is an ATWS sequence with containment failure in the drywell. Sequences that result in LERF contribute less than 2% to overall CDF with a frequency of 3.74E-08. The frequency of uncontrolled release caused by high pressure and/or high temperature was calculated to be 1.29E-06 per year. This frequency consists mostly of ATWS and loss of offsite power events where containment venting is either unavailable or ineffective. Source terms in these sequences are much larger than those due to other types of accident sequences. Sequences in which the containment is vented during the event and, though vented, fails later due to high temperature contribute 8E-08 per year. Another group of sequences involve venting the containment with the containment remaining intact; these contribute 1.1E-07 per year. In yet other sequences, low pressure injection or drywell sprays are used in combination with suppression pool cooling to prevent containment failure and limit source terms to containment leakage. The interfacing systems LOCA sequence frequency of 2.6E-10 per year at Quad Cities makes ISLOCA a negligible contributor to source term and plant risk. For an inerted containment, the likelihood of plant operation with a failure to isolate is extremely remote.

The IPE analyses also show no vulnerability to bypass and failure to isolate sequences. The contribution to the total CDF from interfacing system LOCAs is 2.6E-10 per year. Failure to isolate sequences are precluded because plants with inerted containments will not be operating at full power unless the containment is isolated.

The capability of the Quad Cities Station design is very good. The likelihood that its operating/emergency response staff could bring the plant to a safe, stable state without significant fission product releases (i.e., no core damage and/or no containment failure) is very high. It is concluded that there are no "vulnerabilities" for Quad Cities Station which require immediate attention to improve the plant risk profile or to comply with the NUMARC Closure Guidelines.

A number of good features of Quad Cities Station contribute to the capability of the plant design and operating staff to respond to accidents. These features include the Safe Shutdown Makeup Pump system, a SW system which is shared between Units 1 and 2, and a robust ECCS capability with extra capacity for long term ECCS injection via pump suction realignment to the CCST. A sensitivity study was run to determine the impact of the SSMP on core damage. This study shows that without the SSMP, CDF would be about 3.6 times higher than the current model.

The Quad Cities IPE demonstrated that MAAP is a very useful tool for plant analysis. A CECo-specific version of the MAAP code was found to be of value for system success criteria and for event timing, as well as for calculation of radioactive releases.

As a result of the Integrated IPE/AM Program, CECo has developed a unique understanding of the behavior of the plant under accident conditions and of the total plant capabilities to respond to accidents. The enhanced knowledge of Quad Cities Station developed by CECo will be invaluable in the continuing development and evaluation of Accident Management and IPE insights. This evaluation will be part of the periodic review and update of the Quad Cities PRA - the "Living PRA" process.

QUAD CITIES NUCLEAR POWER STATION
UNITS 1 AND 2

INDIVIDUAL PLANT EXAMINATION
SUBMITTAL REPORT

VOLUME 2

REVISION 1

AUGUST 1996

Submitted By

COMMONWEALTH EDISON COMPANY

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FIGURE 4.1.2-1

SUPPORT STATE EVENT TREE FOR TRANSIENTS, LOCAS, AND SPECIAL INITIATORS

TS1	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET	(2 PAGES)
TS2AA	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AA	(1 PAGE)
TS2AB	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AB	(1 PAGE)
TS2AC	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AC	(1 PAGE)
TS2AD	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AD	(1 PAGE)
TS2AE	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AE	(1 PAGE)
TS2AF	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AF	(1 PAGE)
TS2AG	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AG	(1 PAGE)
TS2AH	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AH	(1 PAGE)
TS2AI	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AI	(1 PAGE)
TS2AJ	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2AJ	(1 PAGE)
TS2BA	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2BA	(1 PAGE)
TS2BB	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2BB	(1 PAGE)
TS2BC	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2BC	(1 PAGE)
TS2BD	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2BD	(1 PAGE)
TS2BE	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2BE	(1 PAGE)
TS2BF	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2BF	(1 PAGE)
TS2CA	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2CA	(1 PAGE)
TS2CC	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2CC	(1 PAGE)
TS2DA	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2DA	(1 PAGE)
TS2DB	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2DB	(1 PAGE)
TS2DC	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2DC	(1 PAGE)
TS2DD	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2DD	(1 PAGE)
TS2DE	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2DE	(1 PAGE)
TS2EA	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2EA	(1 PAGE)
TS2EE	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2EE	(1 PAGE)
TS2EF	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2EF	(1 PAGE)
TS2EG	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2EG	(1 PAGE)
TS2EH	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2EH	(1 PAGE)
TS2EI	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2EI	(1 PAGE)
TS2EJ	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2EJ	(1 PAGE)
TS2G	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 2G	(1 PAGE)
TS3A	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 3A	(1 PAGE)
TS3B	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 3B	(1 PAGE)
TS3C	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 3C	(1 PAGE)
TS3D	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 3D	(1 PAGE)
TS3E	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 3E	(1 PAGE)
TS3F	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 3F	(1 PAGE)
TS3G	TRANSIENT, LOCA, AND SPECIAL INITIATOR SSET 3G	(1 PAGE)

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL

SUBTREE TS1

Entry Conditions: Any initiating event other than Loss of Offsite Power

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AA

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Buses 13, 14, 13-1, 14-1 available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AB

Entry Conditions:	Unit 1 Main and Reserve 125VDC available Buses 13, 14, and 14-1 available Bus 13-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AC

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Buses 13, 14, and 13-1 available
Bus 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AD

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Bus 14, 13-1, and 14-1 available
Bus 13 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AE

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Bus 14 and 14-1 available
Buses 13 and 13-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AF

Entry Conditions:	Unit 1 Main and Reserve 125VDC available Bus 13, 13-1 and 14-1 available Bus 14 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AG

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Buses 13 and 13-1 available
Buses 14 and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AH

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Buses 13 and 14-1 available
Buses 14 and 13-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AI

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Buses 14 and 13-1 available
Buses 13 and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2AJ

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Buses 13, 14, 13-1, and 14-1 available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2BA

Entry Conditions:	Unit 1 Main and Reserve 125VDC available Buses 13-1 and 14-1 available Buses 13 and 14 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2BB

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Bus 14-1 available
Buses 13-1, 13, and 14 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2BC

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Bus 13-1 available
Buses 14-1, 13, and 14 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2BD

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Buses 13 and 14 available
Buses 13-1 and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2BE

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Bus 13 available
Buses 14, 13-1, and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2BF

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Bus 14 available
Buses 13, 13-1, and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2CA

Entry Conditions: Unit 1 Main and Reserve 125VDC available
Buses 13, 14, 13-1, and 14-1 not available
-or-
Unit 1 Main 125VDC available, Reserve 125VDC not available
Buses 13, 14, 13-1, and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2CC

Entry Conditions: Unit 1 Reserve 125VDC available, Main 125VDC not available
Bus 14-1 available
Buses 13, 14, and 13-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2DA

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Buses 14 and 14-1 available
Buses 13 and 13-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2DB

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Bus 14 available
Buses 13, 13-1, and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2DC

Entry Conditions: Unit 1 Reserve 125VDC available, Main 125VDC not available
Buses 13, 13-1, and 14-1 available
Bus 14 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2DD

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Buses 13 and 13-1 available
Buses 14 and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2DE

Entry Conditions: Unit 1 Reserve 125VDC available, Main 125VDC not available
Buses 13 and 14-1 available
Buses 14 and 13-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2EA

Entry Conditions: Unit 1 Main 125VDC available, Reserve 125VDC not available
Buses 14, 13-1, and 14-1 available
Bus 13 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2EE

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Bus 13 available
Buses 14, 13-1, and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2EF

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Buses 13, 14, and 13-1 available
Bus 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2EG

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Buses 13, 14, and 14-1 available
Bus 13-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2EH

Entry Conditions:	Unit 1 Main or Reserve 125VDC or both not available Buses 13 and 14 available Buses 13-1 and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2EI

Entry Conditions:	Unit 1 Main or Reserve 125VDC or both not available
	Buses 14 and 13-1 available
	Buses 13 and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2EJ

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Bus 13-1 available
Buses 13, 14, and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS2G

Entry Conditions: Unit 1 Main or Reserve 125VDC or both not available
Buses 13, 14, 13-1, and 14-1 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS3A

Entry Conditions:	Service Water and Buses 15, 17, and 18 available -or- Service Water and Buses 15 and 17 available Bus 18 not available -or- Service Water and Buses 15 and 18 available Bus 17 not available -or- Service Water and Buses 17 and 18 available Bus 15 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS3B

Entry Conditions: Service Water and Bus 15 available
Buses 17 and 18 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS3C

Entry Conditions: Service Water and Bus 17 available
Buses 15 and 18 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS3D

Entry Conditions:

Service Water and Bus 18 available
Buses 15 and 17 not available
-or-
Buses 15 and 17 available
Service Water and bus 18 not available
-or-
Buses 15 and 18 available
Service Water and bus 17 not available
-or-
Buses 17 and 18 available
Service Water and bus 15 not available
-or-
Buses 15, 17, and 18 available
Service Water not available
-or-
Bus 18 available
Service Water and buses 15 and 17 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS3E

Entry Conditions:	Service Water available Buses 15, 17, and 18 not available -or- No supports available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS3F

Entry Conditions: Bus 15 available
Service Water and buses 17 and 18 not available

FIGURE 4.1.2-1
TRANSIENT, LOCA, AND SPECIAL INITIATOR SUPPORT MODEL
(Continued)

SUBTREE TS3G

Entry Conditions: Bus 17 available
Service Water and buses 15 and 18 not available

Notes for Figure 4.1.2-1:

In some situations, an event's success requires the success of a following event. In this case, the model shows branching even though only one condition is possible. In these cases, failure leads to a null end state, end state 2, indicating that failure is not possible. Such situations as this are described below.

1. Because DG1 auxiliaries are powered from Bus 19, DG1 success requires Bus 14-1 to supply power to Bus 19. Since DG1 can only supply Bus 14-1, local failures of Bus 14-1 are impossible given DG1 success. Therefore, the following paths are set to the null end state.

   Tree: TS1 Paths: 11, 14, 34, 43, 84, 87, 98

2. Similar to Item 1 above, since DG1/2 auxiliaries are powered from Bus 18 or 28, DG1/2 success requires Bus 13-1 to supply power to Bus 18. Since in the Transient and LOOP trees we are only interested in DG1/2 supplying Bus 13-1, local failures of Bus 13-1 are impossible given DG1/2 success. Therefore, the following paths are set to the null end state.

   Tree: TS1 Paths: 27, 35, 40, 64, 71

3. When Bus 13-1 is lost, Bus 18 can receive power from Bus 19. Success of Bus 18 after loss of Bus 13-1 requires Bus 19. Therefore, the following paths are set to the null end state.

Subtrees:		
	TS2AB Paths:	3,10,17,24,31,38,45,52
	TS2AE	3,10,17,25,32,39,46
	TS2AH	3,10,17,24
	TS2BB	3

4. When Bus 13 is lost and Bus 14 is available, Bus 15 can be fed from Bus 16 or Bus 17. In the following paths, Bus 17 failure is impossible, due to Bus 15 success with Bus 16 failure. Therefore, the following paths are set to the null end state.

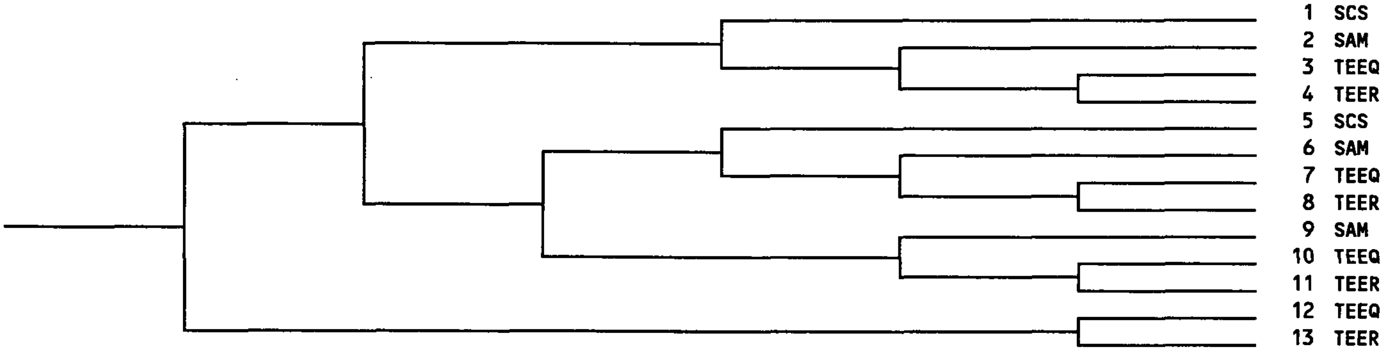
Subtrees:		
	TS2AD Paths:	25
	TS2AE	22
	TS2AI	19
	TS2BF	7

ATWS PLANT RESPONSE TREE

TREE ATW-1	ATWS - INITIAL PHASE	(4 PAGES)
TREE ATW-2	FW AND MC SUCCESS, RPT SUCCESS, ARI FAILURE	(1 PAGE)
TREE ATW-3	FW SUCCESS, MC FAILURE, RPT SUCCESS, ARI FAILURE	(2 PAGES)
TREE ATW-4	FW FAILURE, MC SUCCESS, RPT SUCCESS, ARI FAILURE	(2 PAGES)
TREE ATW-5	FW AND MC FAILURE, RPT SUCCESS, ARI FAILURE	(2 PAGES)
TREE ATW-6	ARI SUCCESS	(2 PAGES)
TREE ATW-7	OIADS, SBLC AND HPCI SUCCESS, RHR HEAT REMOVAL FAILURE	(1 PAGE)
TREE ATW-8	OIADS AND SBLC SUCCESS, HPCI FAILURE, RHR HEAT REMOVAL SUCCESS	(1 PAGE)
TREE ATW-9	OIADS AND SBLC SUCCESS, HPCI AND RHR COOLING FAILURE	(2 PAGES)
TREE ATW-10	OIADS AND SBLC SUCCESS, HPCI AND LPCI FAILURE	(2 PAGES)
TREE ATW-11	ARI SUCCESS, SPC SUCCESS, HPCI AND RCIC FAILURE	(3 PAGES)
TREE ATW-12	ARI SUCCESS, HPCI AND SPC SUCCESS, INVC FAILURE	(2 PAGES)
TREE ATW-13	ARI SUCCESS, HPCI/INVC -OR- RCIC SUCCESS, SPC FAILURE	(5 PAGES)
TREE ATW-14	ARI AND HPCI SUCCESS, RHR COOLING AND INVC FAILURE	(1 PAGE)
TREE ATW-15	ARI SUCCESS, HPCI, RCIC, AND RHR COOLING FAILURE	(7 PAGES)
TREE ATW-16	ARI SUCCESS, HPCI/INVC -OR- RCIC SUCCESS, RHR FAILURE	(4 PAGES)
TREE ATW-17	ARI SUCCESS, HPCI/INVC -OR- RCIC SUCCESS, RHR PUMP FAILURE	(2 PAGES)
TREE ATW-18	ARI SUCCESS, HPCI, RCIC, AND RHR PUMP FAILURE	(4 PAGES)
TREE ATW-19	ARI SUCCESS, HPCI/INVC AND SPC SUCCESS	(2 PAGES)

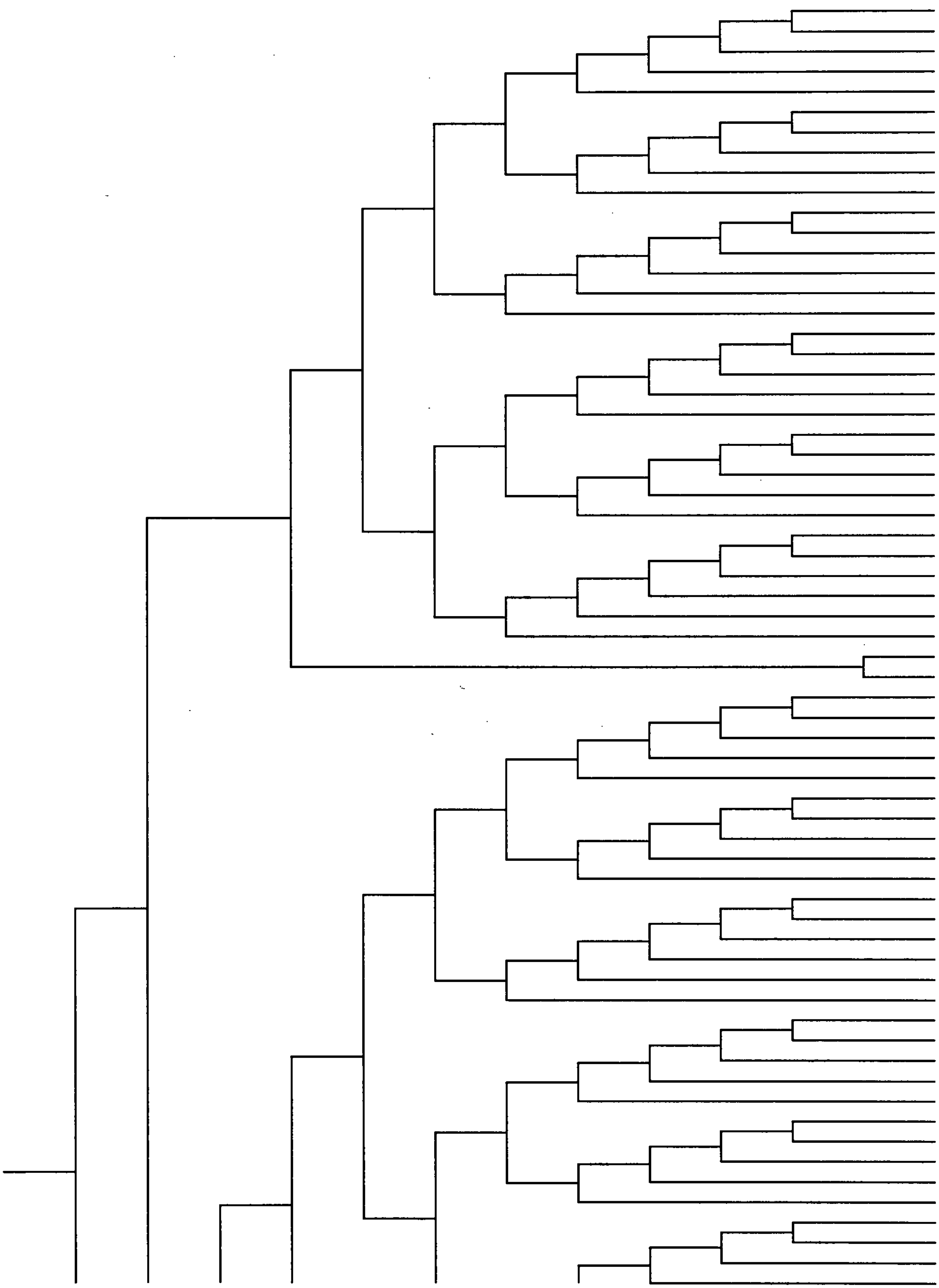
08/23/96 11:14:43 CADET 1.00
ATW2
ATWS - FW AND MC SUCCESS, RPT SUCCESS, ARI FAILURE

RVO	OIADS	OSL1	OSL2	SLC	OAL	WW/DW



08/23/96 11:14:43 CADET 1.00
ATW3
ATWS - FW SUCCESS, MC FAILURE, RPT SUCCESS, ARI FAILURE

RVO	OIADS	OSL1	OSL2	SLC	HP1	LPA	LPB	OHX	RHRHX	OSPC	SPC	WW/DW



1 SUCCESS
2 *ATW7
3 *ATW7
4 *ATW7
5 *ATW7
6 SUCCESS
7 *ATW7
8 *ATW7
9 *ATW7
10 *ATW7
11 SUCCESS
12 *ATW7
13 *ATW7
14 *ATW7
15 *ATW7
16 *ATW7
17 *ATW8
18 *ATW9
19 *ATW9
20 *ATW9
21 *ATW9
22 *ATW8
23 *ATW9
24 *ATW9
25 *ATW9
26 *ATW9
27 *ATW8
28 *ATW9
29 *ATW9
30 *ATW9
31 *ATW9
32 *ATW10
33 TEEQ
34 TEER
35 SUCCESS
36 *ATW7
37 *ATW7
38 *ATW7
39 *ATW7
40 SUCCESS
41 *ATW7
42 *ATW7
43 *ATW7
44 *ATW7
45 SUCCESS
46 *ATW7
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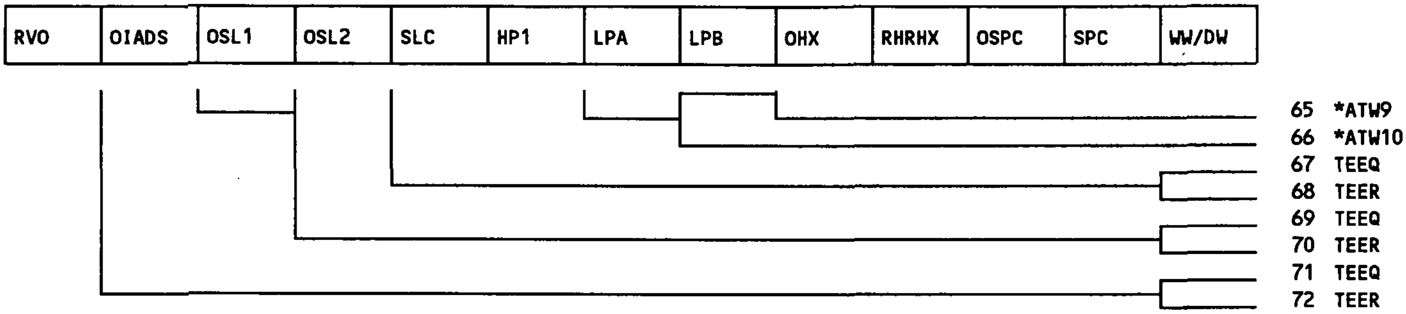
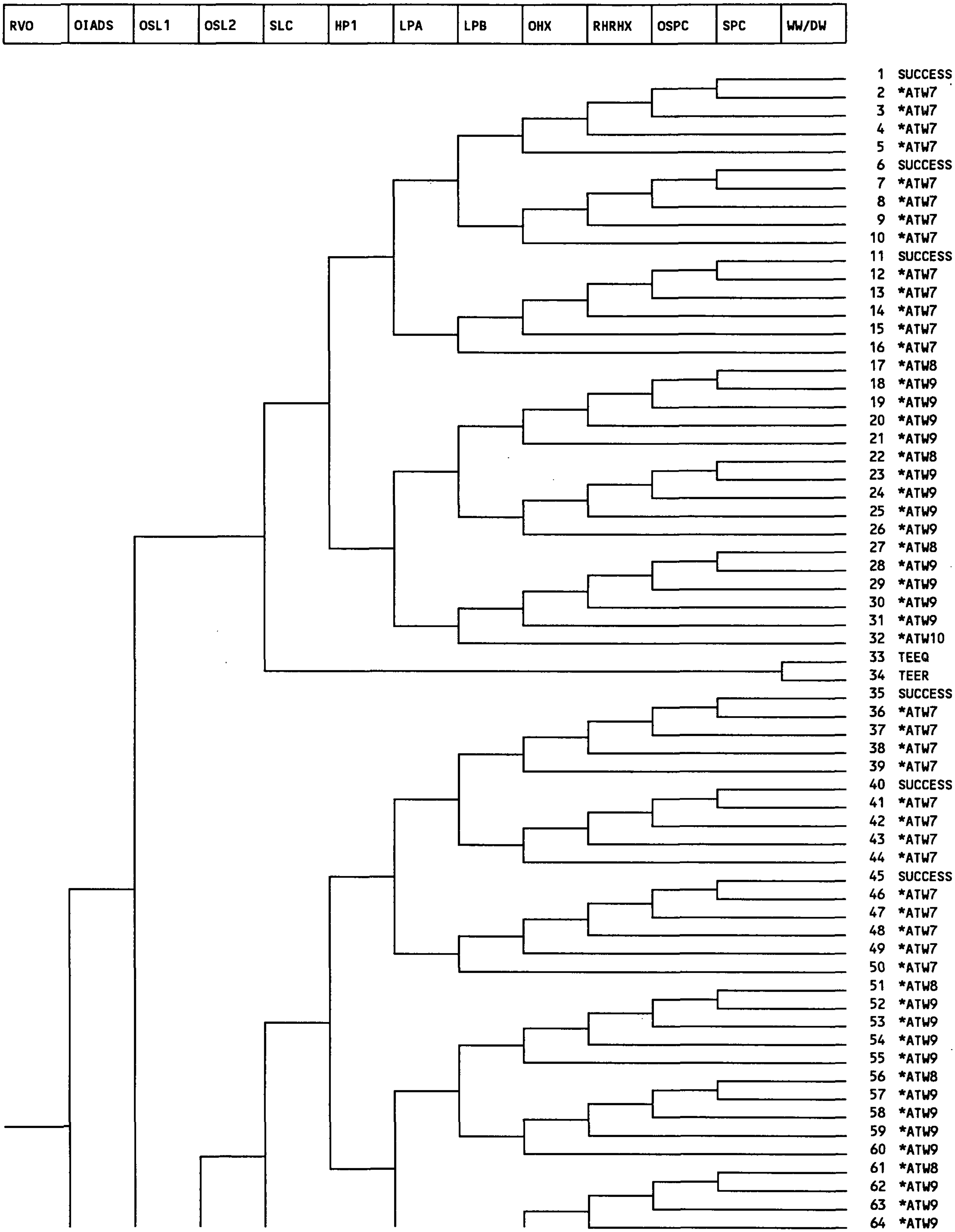
63 *ATW9
64 *ATW9


RVO
OIADS
OSL1
OSL2
SLC
HP1
LPA
LPB
OHX
RHRHX
OSPC
SPC
WW/DW
65 *ATW9
66 *ATW10
67 TEEQ
68 TEER
69 TEEQ
70 TEER
71 TEEQ
72 TEER

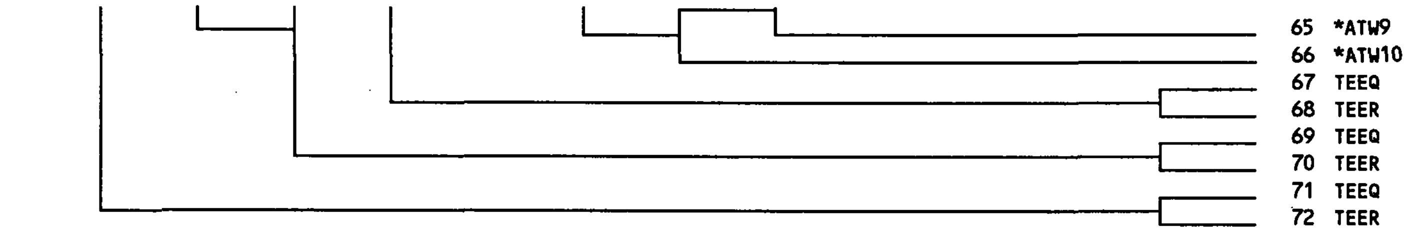
08/23/96 11:14:43 CADET 1.00

ATW4

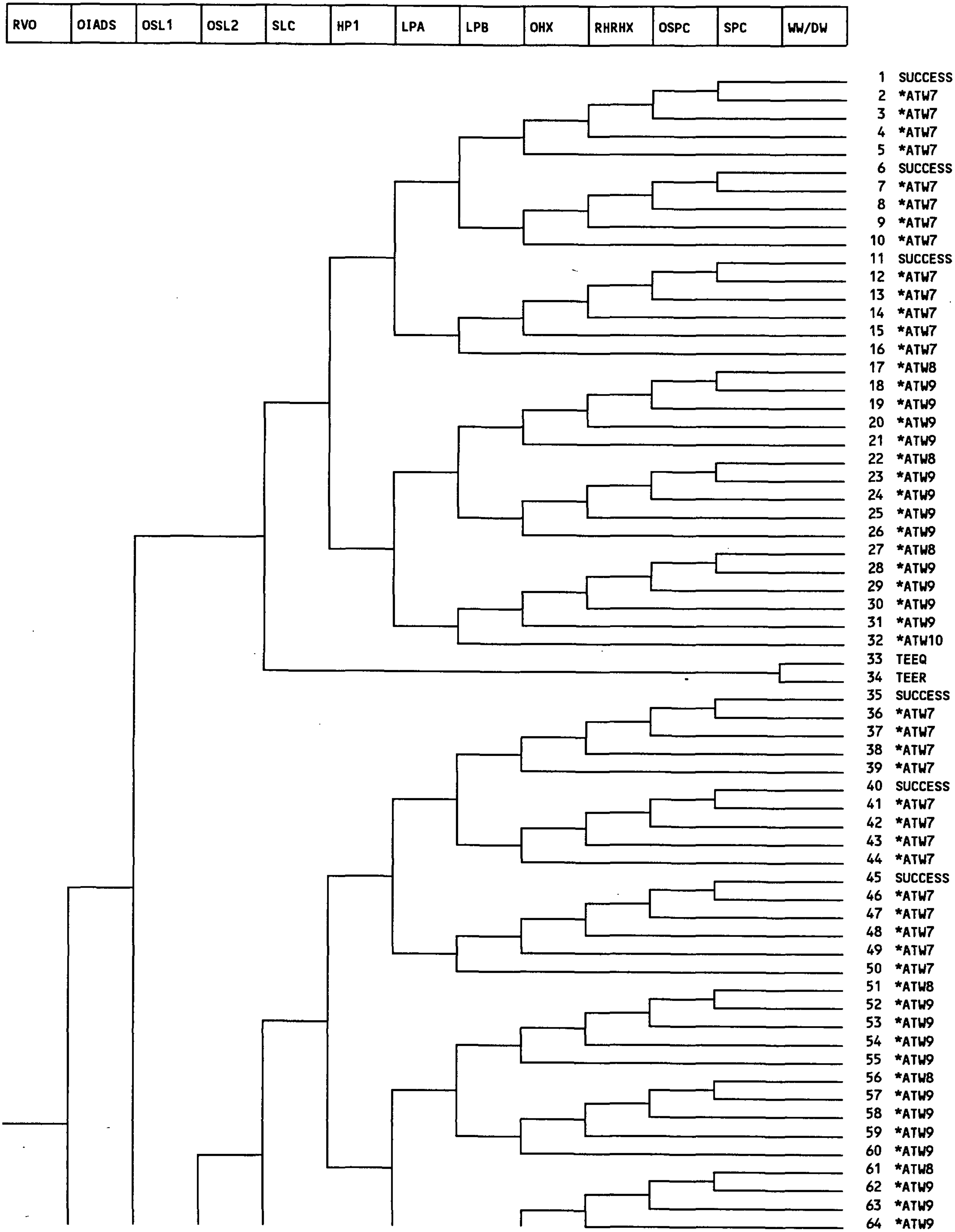
ATWS - FW FAILURE, MC SUCCESS, RPT SUCCESS, ARI FAILED



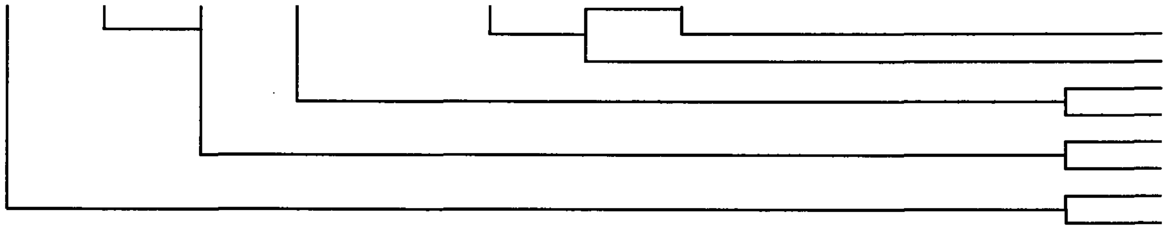
RVO	OIADS	OSL1	OSL2	SLC	HP1	LPA	LPB	OHX	RHRHX	OSPC	SPC	WW/DW



08/23/96 11:14:43 CADET 1.00
ATW5
ATWS - FW AND MC FAILURE, RPT SUCCESS, ARI FAILURE



RVO	OIADS	OSL1	OSL2	SLC	HP1	LPA	LPB	OHX	RHRHX	OSPC	SPC	WW/DW



65 *ATW9
66 *ATW10
67 TEEQ
68 TEER
69 TEEQ
70 TEER
71 TEEQ
72 TEER

ATWS TREE 7

ATWS - OIADS, SBLC AND HPCI SUCCESS, RHR HEAT REMOVAL FAILURE

ATWS PRT ENTRY CONDITION SUMMARY

	EQUIPMENT					
ATWS PRT[1]	OIADS	(OSL1 or OSL2)/SBLC	HP1	LPA/LPB	OHX/ RHRHX	OSPC/ SPC
ATW-7	S	S	S	Note 6	Note 6	Note 6
ATW-8	S	S	F	S[5]	S/S	S/S
ATW-9	S	S	F	S[5]	Note 7	Note 7
ATW-10	S	S	F	F/F	N	N

Notes:

1. Trees 7 through 10 are entered from Trees 3, 4, and 5.
2. F = Failure
3. S = Success
4. N = Not demanded
5. Success of at least one of these nodes is required.
6. One of the nodes OHX, RHRHX, OSPC, SPC, or LPA <u>and</u> LPB fails. The status of the other nodes is not applicable.
7. One of the nodes OHX, RHRHX, OSPC, and SPC fails. The status of the other nodes is not applicable.

ATWS TREE 8
ATWS - OIADS AND SBLC SUCCESS, HPCI FAILURE, RHR HEAT REMOVAL SUCCESS
ATWS PRT ENTRY CONDITION SUMMARY

	EQUIPMENT					
ATWS PRT[1]	OIADS	(OSL1 or OSL2)/SBLC	HP1	LPA/LPB	OHX/ RHRHX	OSPC/ SPC
ATW-7	S	S	S	Note 6	Note 6	Note 6
ATW-8	S	S	F	S[5]	S/S	S/S
ATW-9	S	S	F	S[5]	Note 7	Note 7
ATW-10	S	S	F	F/F	N	N

Notes:

1. Trees 7 through 10 are entered from Trees 3, 4, and 5.
2. F = Failure
3. S = Success
4. N = Not demanded
5. Success of at least one of these nodes is required.
6. One of the nodes OHX, RHRHX, OSPC, SPC, or LPA <u>and</u> LPB fails. The status of the other nodes is not applicable.
7. One of the nodes OHX, RHRHX, OSPC, and SPC fails. The status of the other nodes is not applicable.

ATWS TREE 9
ATWS - OIADS AND SBLC SUCCESS, HPCI AND RHR COOLING FAILURE
ATWS PRT ENTRY CONDITION SUMMARY

	EQUIPMENT					
ATWS PRT[1]	OIADS	(OSL1 or OSL2)/SBLC	HP1	LPA/LPB	OHX/ RHRHX	OSPC/ SPC
ATW-7	S	S	S	Note 6	Note 6	Note 6
ATW-8	S	S	F	S[5]	S/S	S/S
ATW-9	S	S	F	S[5]	Note 7	Note 7
ATW-10	S	S	F	F/F	N	N

Notes:

1. Trees 7 through 10 are entered from Trees 3, 4, and 5.
2. F = Failure
3. S = Success
4. N = Not demanded
5. Success of at least one of these nodes is required.
6. One of the nodes OHX, RHRHX, OSPC, SPC, or LPA and LPB fails. The status of the other nodes is not applicable.
7. One of the nodes OHX, RHRHX, OSPC, and SPC fails. The status of the other nodes is not applicable.

ATWS TREE 10
ATWS - OIADS AND SBLC SUCCESS, HPCI FAILURE, LPCI FAILURE
ATWS PRT ENTRY CONDITION SUMMARY

	EQUIPMENT					
ATWS PRT[1]	OIADS	(OSL1 or OSL2)/SBLC	HP1	LPA/LPB	OHX/ RHRHX	OSPC/ SPC
ATW-7	S	S	S	Note 6	Note 6	Note 6
ATW-8	S	S	F	S[5]	S/S	S/S
ATW-9	S	S	F	S[5]	Note 7	Note 7
ATW-10	S	S	F	F/F	N	N

Notes:

1. Trees 7 through 10 are entered from Trees 3, 4, and 5.
2. F = Failure
3. S = Success
4. N = Not demanded
5. Success of at least one of these nodes is required.
6. One of the nodes OHX, RHRHX, OSPC, SPC, or LPA <u>and</u> LPB fails. The status of the other nodes is not applicable.
7. One of the nodes OHX, RHRHX, OSPC, and SPC fails. The status of the other nodes is not applicable.

TABLE 4.2.2-1
INITIATOR/FRONTLINE SYSTEMS DEPENDENCY MATRIX

INITIATING EVENT	FW	COND	RPS
Transient w/ FW and MC available			
Transient w/ FW available & MC unavailable		C(2)	
Transient w/FW unavailable & MC available	C(2)		
Transient FW & MC unavailable	C(2)	C(2)	
LOOP (single unit)	C(3)	P(4)	P(6)
LOOP (dual unit)	C(3)	P(4)	P(6)
IORV	D(5)	D(5)	
LLOCA	D(5)	D(5)	
MLOCA	D(5)	D(5)	
SLOCA	D(5)	D(5)	
ISLOCA	D(5)	D(5)	
Loss of 125 VDC (Reserve Bus)	P(7)	P(7)	P(13)
Loss of Service Water	D(8)	D(8)	
Loss of 4 kV Bus 11	P(3)		
Loss of 4 kV Bus 12	P(3)		
Loss of 4 kV Bus 13		P(4)	A(6)
Loss of 4 kV Bus 14		P(4)	
Loss of 480 VAC Bus 18			P(9)
Loss of 480 VAC MCC 18-2			P(10)
Loss of Instrument Air	P(11)	D(12)	

Notes for Table 4.2.2-1:

1. The format of each dependency matrix is similar. The first column on the left is the system or event to be examined. An entry in one of the subsequent columns indicates that the system listed at the top of the table is dependent to some degree upon the system or event listed in the first column. A blank intersection means that the system at the top is independent of the system or event at the left. The following items define the characters used in the matrices:

   C The system at the top of the column is completely dependent upon the system or event listed on the left. If the system at the left fails or is unavailable, or if the initiating event occurs, the system at the top of the matrix will be unavailable. The system is assumed to be completely dependent if manual actions are required to align to an alternate (or backup) source.

   P Partial dependence of the system at the top on the event or the system at the left. The event may degrade a system's ability to perform its function but will not completely fail the system. The system is assumed to be partially dependent if automatic actions will occur to align to an alternate (or backup) source.

   D Delayed impact on the system at the top of the matrix by the event or system failure at the left. Delayed dependence is exemplified by the support system providing room cooling. Loss of room cooling may not immediately cause a failure of other systems, but may, over an extended period, cause a system to fail during its mission time.

   R A relationship or interconnection between systems that is not technically defined as a dependency but is utilized by the system at the top of the matrix in some way. Components (other than piping) are shared.

Notes for Table 4.2.2-1 (Continued):

A System on the left can be used as an alternate (or backup) source to the system at the top of the matrix. For example, the primary water source for RCIC suction is the CCSTs. Alternate suction is from the suppression pool.

A blank row/column intersection means that the system at the top is not directly dependent on the system/event at the left. "Cascading" dependencies may still exist, however. For example, system Y has a delayed dependency on system X, and system Z has a partial dependence on system Y, but no other dependence on system X. In this case, the dependency of Y on X would be shown by the matrix, as would the dependency of Z on Y; the Z/X intersection would generally be left blank, however.

An additional column U 1/2 DEP (Unit 1/2 Dependency) has been added to each system/system matrix to indicate the degree of inter-unit dependence for each system for Units 1 and 2. The systems at the Quad Cities Site are noted on the matrix as belonging to one of three categories: I = Independent, S = Shared, X = Cross-tied. Independent systems are those which function solely at the unit where they are located. For example, the High Pressure Coolant Injection (HPCI) systems at Units 1 and 2 are independent. Shared systems normally serve both units simultaneously, e.g., the service water system. Cross-tied systems are those which normally operate independently at each unit but are capable of serving as a backup to the opposite unit's system by supplying support through the cross-tie between the systems, e.g., the instrument air system at each unit.

2. Frontline system has failed as part of the IE definition. For the purposes of identification, frontline systems are defined as those systems that are operational at the time of the transient. Safety systems are those systems required to respond to the event. The "Additional Systems" are defined as those systems which are identified in the Emergency Operating Procedures or which prove useful in accident management strategies. Support systems are those systems providing support in the form of AC or DC electrical power or other motive power (such as air), or cooling/lubrication to components of the front line, safety or additional systems. These are not plant designations. They are used only to simplify the application of dependency analysis.

3. Feed pumps are powered from 4160 Buses 11, 12.

4. Condensate pumps are powered from 4160 Buses 13, 14. The diesel generators provide alternate power. A LOOP will leave the system in a degraded state.

5. A LOCA will deplete the inventory of the condenser leading to a loss of suction to the Reactor Feedwater and Condensate pumps.

6. The RPS motor-generators (MGs) are powered from 480VAC MCCs 18-2 and 19-2. The diesel generators provide alternate power to these MCCs. A LOOP will leave the system power supply in a degraded state; RPS is a fail-safe system, however, and a loss of power will lead to a reactor scram. Reserve power to the RPS buses is supplied from MCC 15-2.

7. 125VDC is required to close the 4160VAC circuit breakers for the pumps.

8. Service Water is the heat sink for the TBCCW system. Loss of SW would result in a delayed failure of TBCCW. Loss of TBCCW system would result in loss of cooling to RFP oil coolers and overheating of RFP bearings.

9. Bus 18 feeds MCC 18-2 which powers RPS MG set A which is the normal power supply to RPS Bus A. Reserve power to the RPS buses is supplied from MCC 15-2.

10. MCC 18-2 powers RPS MG set A which is the normal power supply to RPS Bus A. Reserve power to the RPS buses is supplied from MCC 15-2.

11. Instrument air supplies all air-operated valves in FW&C systems. Feedwater Regulating Valve fails as-is on loss of IA or control signal. The RFP recirculation valves fail open on loss of instrument air or control power.

12. Makeup to the hotwell is via two air-operated valves (LCV 1-3301, 3302), air service is instrument air, and control power is from the Essential Service Bus.

13. Backup scram valves are powered from 125 VDC Bus 1B-1.

TABLE 4.2.2-2
INITIATOR/SAFETY SYSTEMS DEPENDENCY MATRIX

INITIATING EVENT	RPT	ARI	ERV	TR/SRV	SV	HPCI	ADS	RHR	CS	RCIC
Transient w/ FW and MC available			R(7)	R(7)	R(7)					
Transient w/ FW available & MC unavailable			R(7)	R(7)	R(7)					
Transient w/ FW unavailable & MC available			R(7)	R(7)	R(7)					
Transient FW & MC unavailable			R(7)	R(7)	R(7)					
LOOP (single unit)								P(6)	P(6)	
LOOP (dual unit)								P(6)	P(6)	
IORV			P(8)	P(8)	P(8)	D(2)				D(5)
LLOCA						C(2,3)		P(3)	P(3)	C(5,3)
MLOCA						D(2,3)		P(3)	P(3)	D(5,3)
SLOCA						D(2,3)		P(3)	P(3)	D(5,3)
ISLOCA						D(2,3)		P(4)	P(4)	D(5,3)
Loss of 125 VDC (Reserve Bus)	P(9)	P(9)	A(10)			C(11)	P(12)	P(13)	P(14)	P(15)
Loss of Service Water										
Loss of 4 kV Bus 11										
Loss of 4 kV Bus 12										
Loss of 4 kV Bus 13										
Loss of 4 kV Bus 14										
Loss of 480 VAC Bus 18								P(13)	P(14)	
Loss of 480 VAC MCC 18-2										
Loss of Instrument Air						P(16)				P(17)

Notes for Table 4.2.2-2:

1. See Note 1 of Table 4.2.2-1 for descriptions of the dependency codes used in this table.

2. HPCI requires steam from the reactor to operate; depending on the time to lose steam pressure, HPCI may be available for a short time.

3. A LOCA may create a path by which injected coolant can bypass the core and flow directly out the break to the containment.

4. These systems interface with the RPV and are evaluated as initiators of ISLOCAs. An ISLOCA could cause at least one train of these systems to be unavailable.

Notes for Table 4.2.2-2 (Continued):

5. RCIC requires steam from the reactor to operate; depending on the time to lose steam pressure, RCIC may be available for a short time.

6. These systems involve pumps that are powered from ESF AC buses. The alternate power source is from DG 1 and DG 1/2. A LOOP will leave the system in a degraded state.

7. The rapid pressure surge in the reactor vessel due to an ATWS would challenge the relief valve system and could cause them to stick open.

8. IORV fails at least one of the ERVs or SVs or the TR/SRV.

9. The ATWS Division I logic is powered from 125VDC RB DC Panel 1 with alternate power from the 120VAC Instrument Bus. ATWS Division II logic is powered from TB Reserve Bus 1B-1 with alternate power from the 120VAC Essential Services Bus. The ATWS valves and the RPT field breaker relay are powered by 125VDC from Bus 11, 12 control power.

10. Electromatic relief valves require 125VDC power to open. Normal supply is from TB MN Bus 1A with alternate power available from TB Reserve Bus 1B-1.

11. Power supplies for HPCI components and logic are as follows:

125VDC TB 1A-1	Reserve Feed to HPCI Sys Logic and Control Panel 901-39	
125VDC TB 1B-1	Main Feed to HPCI Sys Logic and Control Panel 901-39	
120/240VAC Ess Bus	HPCI Instrument Panels 901-3, 901-19	
120/240VAC Inst Bus	Panels 901-3, 901-4, 901-19, 901-38, 901-39	
250VDC RB MCC 1A	Gland Seal Hotwell Pump	
	Gland Seal Leak-Off Blower	
	Auxiliary Lube Oil Pump	
	Emergency Lube Oil Pump	
	MO1-2301-4	MO1-2301-3
	MO1-2301-6	MO1-2301-8
	MO1-2301-14	MO1-2301-35
	MO1-2301-36	MO1-2301-48
	MO1-2301-9	MO1-2301-10
	MO1-2301-49	

Notes for Table 4.2.2-2 (Continued):

12. 125VDC power is required for ADS logic and valve solenoids as noted:

Logic:

Div. I	TB Main Bus 1A-1
Div. II	TB Reserve Bus 1B-1 (TB MN Bus 1A-1 alternate)

Valves:

Normal	TB Main Bus 1A-1
Alternate	TB Reserve Bus 1B-1

13. RHR system components are powered by the following buses:

AC Power:

4160V

Bus 13-1	RHR Pumps 1A, 1B-1002
Bus 14-1	RHR Pumps 1C, 1D-1002

480V

MCC 18-1B	MO1-1001-7A	MO1-1001-7B
	MO1-1001-16A	MO1-1001-18A
	MO1-1001-19A	MO1-1001-23A
	MO1-1001-26A	MO1-1001-34A
	MO1-1001-36A	MO1-1001-37A
	MO1-1001-43A	MO1-1001-43B
	MO1-1001-50	
MCC 18/19-5	MO1-1001-28A	MO1-1001-28B
	MO1-1001-29A	MO1-1001-29B
MCC 19-4	MO1-1001-7C	MO1-1001-7D
	MO1-1001-16B	MO1-1001-18B
	MO1-1001-19B	MO1-1001-23B
	MO1-1001-26B	MO1-1001-34B
	MO1-1001-36B	MO1-1001-37B
	MO1-1001-43C	MO1-1001-43D

Notes for Table 4.2.2-2 (Continued):

RHR (Continued)

DC Power:

250V

MCC 1B	MO1-1001-47

125V

RB DC PNL 1	RHR Pumps 1A, 1B circuit breaker control power (main feed) RHR Pumps 1C, 1D circuit breaker control power (alternate feed)
TB Reserve Bus 1B-1	RHR Pumps 1C, 1D circuit breaker control power (main feed) RHR Pumps 1A, 1B circuit breaker control power (alternate feed) RHR Loop B System II initiation circuitry Permissive circuitry
TB MN Bus 1A-1	RHR Loop A System I initiation circuitry Permissive circuitry

14. CS System Components are powered by the following buses:

AC Power:

4160V

SWGR 13-1	Power to Pump 1A-1401
SWGR 14-1	Power to Pump 1B-1401

480V

MCC 18-1A	Power to Valves MO1-1402-3A, 4A, 24A, 25A
MCC 19-1	Power to Valves MO1-1402-3B, 4B, 24B, 25B

120V

MCC 18-1A-1	MO1-1402-38A
MCC 19-1-1	MO1-1402-38B

Notes for Table 4.2.2-2 (Continued):

CS (Continued)

DC Power:

125V

RB DC PNL 1	Power to close circuit breaker for CS Pump 1A-1401
TB Reserve Bus 1B-1	Power to close circuit breaker for CS Pump 1B-1401; also used for System II initiation control logic
MN Bus 1A-1	Power for System I initiation control logic

15. 125VDC power is required as follows:

125VDC Bus 1A-2 supplies control power to the following RCIC air-operated valves:

AO1-1301-12 AO1-1301-34
AO1-1301-13 AO1-1301-35
AO1-1301-32

125VDC Bus 1B-1 supplies power to the turbine trip and isolation circuits.

16. Instrument Air is required to maintain HPCI steamline drain valves 29, 30, 64, and 65 open and steam line drain valves 28 and 31 closed. Valves automatically reposition on HPCI initiation and on loss of IA. The effect of the reposition is to bypass the steamline trap, and redirect condensate to the torus.

17. Instrument Air supplies the following RCIC valves:

AO1-1301-12 AO1-1301-34
AO1-1301-13 AO1-1301-35
AO1-1301-32

TABLE 4.2.2-3
INITIATOR/ADDITIONAL SYSTEMS DEPENDENCY MATRIX

INITIATING EVENT	SBLC	SBCS	CRD	SSMP	FP	SBGT	APCV
Transient w/ FW and MC available							
Transient w/ FW available & MC unavailable							
Transient w/ FW unavailable & MC available							
Transient FW & MC unavailable							
LOOP (single unit)	P(2)	P(2)	P(2)	P(2)	P(3)	P(2)	P(2)
LOOP (dual unit)	P(2)	P(2)	P(2)	P(2)	P(3)	P(2)	P(2)
IORV							
LLOCA							
MLOCA							
SLOCA							
ISLOCA							
Loss of 125 VDC (Reserve Bus)							
Loss of Service Water		C(4)		D(5)	P(6)		
Loss of 4 kV Bus 11							
Loss of 4 kV Bus 12							
Loss of 4 kV Bus 13		P(4)	P(7)				
Loss of 4 kV Bus 14		P(4)	P(7)				
Loss of 480 VAC Bus 18	P(8)						
Loss of 480 VAC MCC 18-2							
Loss of Instrument Air						C(9)	D(10)

Notes for Table 4.2.2-3:

1. See Note 1 of Table 4.2.2-1 for descriptions of the dependency codes used in this table.

2. These systems involve equipment that is powered from ESF buses. Alternate power sources are the DG 1 and DG 1/2. A LOOP will leave the system in a degraded state using emergency power sources.

3. Fire Protection uses diesel-driven fire pumps. A loss of AC would leave the system in a degraded state, relying on battery power to start the diesel engines.

Notes for Table 4.2.2-3 (Continued):

4. Secondary water makeup source to hotwell (Primary is CCST). Requires operable SW pump. SW Pumps are powered by 4 kV Buses 13, 14, 23, and 24.

5. SW supplies cooling water to the room air cooler, fire main is an alternate source of cubicle cooling.

6. Unit 1 service water maintains pressure in fire main during normal plant conditions in addition to serving as a backup for the fire protection water system. Valve MO1/2-3906 provides this backup and is powered from 250VDC MCC 1.

7. CRD pump motor 1A is powered from SWGR 13.
   CRD pump motor 1B is powered from SWGR 14.

8. SBLC Pump A and Explosive Valve 1-1106A supplied by 480VAC MCC 18-1A.
   SBLC Pump B and Explosive Valve 1-1106B supplied by 480VAC MCC 19-1.
   SBLC Tank heater 1-1103 supplied by 480VAC MCC 19-1.

9. The containment vent valves to SGBT valves are supplied by Instrument Air. Control power to these valves is supplied from the ESS Bus.

10. The valves in the Augmented Primary Containment vent system are all air-operated valves supplied from Instrument Air. Accumulators on each valve operator allow actuations on a loss of Instrument Air.

TABLE 4.2.2-4
INITIATOR/SUPPORT SYSTEMS DEPENDENCY MATRIX

INITIATING EVENT	TBCCW	SW	RHRHX	DGCW	ECCS-FILL	CLEAN DEMIN	CONDENSATE TRANSFER	INST AIR
Transient w/ FW and MC available								
Transient w/ FW available & MC unavailable								
Transient w/ FW unavailable & MC available								
Transient FW & MC unavailable								
LOOP (single unit)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)
LOOP (dual unit)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)
IORV								
LLOCA								
MLOCA								
SLOCA								
ISLOCA								
Loss of 125 VDC (Reserve Bus)		P(10)	P(11)					
Loss of Service Water	D(12)							D(13)
Loss of 4 kV Bus 11								
Loss of 4 kV Bus 12								
Loss of 4 kV Bus 13	P(14)	P(10)	P(11)					
Loss of 4 kV Bus 14	P(14)	P(10)	P(11)			P(17)	P(18)	
Loss of 480 VAC Bus 18		D(10)	P(11)	P(19)	C(20)		P(18)	A(21)
Loss of 480 VAC MCC 18-2		D(10)					P(18)	A(21)
Loss of Instrument Air	D(22)	D(15)					A(16)	

TABLE 4.2.2-4 (Continued)
INITIATOR/SUPPORT SYSTEMS DEPENDENCY MATRIX

INITIATING EVENT	DRYWELL PNEUMATICS	SA	HVAC	CCST	SUPP POOL	HW	CAS	OFF-SITE POWER	DG 1	DG 1/2
Transient w/ FW and MC available								P(5)		
Transient w/ FW available & MC unavailable								P(5)		
Transient w/ FW unavailable & MC available								P(5)		
Transient FW & MC unavailable								P(5)		
LOOP (single unit)	P(2)	P(2)	P(2)					C(8)	D(3)	D(3)
LOOP (dual unit)	P(2)	P(2)	P(2)					C(8)	D(3)	D(3)
IORV								P(5)		
LLOCA	D(9)							P(5)		
MLOCA	D(9)							P(5)		
SLOCA	D(9)							P(5)		
ISLOCA								P(5)		
Loss of 125 VDC (Reserve Bus)							P(23)	P(5)	C(24)	A(25)
Loss of Service Water	D(26)	D(27)	P(28)			A(29)		P(5)		
Loss of 4 kV Bus 11								P(5)		
Loss of 4 kV Bus 12								P(5)		
Loss of 4 kV Bus 13		P(30)	P(28)					P(5)		
Loss of 4 kV Bus 14		P(30)	P(28)					P(5)		
Loss of 480 VAC Bus 18	P(32)		P(28)					P(5)	P(24)	C(25)
Loss of 480 VAC MCC 18-2			P(28)					P(5)	P(24)	
Loss of Instrument Air						C(31)		P(5)		

TABLE 4.2.2-4 (Continued)
INITIATOR/SUPPORT SYSTEMS DEPENDENCY MATRIX

INITIATING EVENT	NON-ESF AC BUSES	NON-EMERGENCY ESF AC BUSES	4160VAC BUS 13-1	4160VAC BUS 14-1	480VAC BUS 18	480VAC BUS 19	250VDC TBMCC1	250VDC TBMCC2
Transient w/ FW and MC available	P(6)	P(6)						
Transient w/ FW available & MC unavailable	P(6)	P(6)						
Transient w/ FW unavailable & MC available	P(6)	P(6)						
Transient FW & MC unavailable	P(6)	P(6)						
LOOP (single unit)	C(4)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)
LOOP (dual unit)	C(4)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)	P(2)
IORV	P(6)	P(6)						
LLOCA	P(6)	P(6)						
MLOCA	P(6)	P(6)						
SLOCA	P(6)	P(6)						
ISLOCA	P(6)	P(6)						
Loss of 125 VDC (Reserve Bus)	P(6)	P(6)	A(33)	C(33)	A(34)	C(34)		
Loss of Service Water	P(6)	P(6)						
Loss of 4 kV Bus 11	P(6)	P(6)						
Loss of 4 kV Bus 12	P(6)	P(6)						
Loss of 4 kV Bus 13	P(6)	P(6)						
Loss of 4 kV Bus 14	P(6)	P(6)						
Loss of 480 VAC Bus 18	P(6)	P(6)				A(35)	A(36)	A(37)
Loss of 480 VAC MCC 18-2	P(6)	P(6)					A(36)	A(37)
Loss of Instrument Air	P(6)	P(6)						

TABLE 4.2.2-4 (Continued)
INITIATOR/SUPPORT SYSTEMS DEPENDENCY MATRIX

INITIATING EVENT	125VDC MN BUS 1A	125VDC MN BUS 2A	INSTR BUS	ESS SERVICE BUS	RPS BUSES
Transient w/ FW and MC available					
Transient w/ FW available & MC unavailable					
Transient w/ FW unavailable & MC available					
Transient FW & MC unavailable					
LOOP (single unit)	P(2)	P(2)	P(2)	P(2)	C(7)
LOOP (dual unit)	P(2)	P(2)	P(2)	P(2)	C(7)
IORV					
LLOCA					
MLOCA					
SLOCA					
ISLOCA					
Loss of 125 VDC (Reserve Bus)					
Loss of Service Water					
Loss of 4 kV Bus 11					
Loss of 4 kV Bus 12					
Loss of 4 kV Bus 13			A(39)		A(41)
Loss of 4 kV Bus 14				A(40)	
Loss of 480 VAC Bus 18	A(38)		P(39)	P(40)	P(41)
Loss of 480 VAC MCC 18-2	A(38)		P(39)	A(40)	P(41)
Loss of Instrument Air					

Notes for Table 4.2.2-4:

1. See Note 1 of Table 4.2.2-1 for descriptions of the dependency codes used in this table.
2. LOOP initiator eliminates one of the redundant components in the system. Other components may still be operable under emergency power supplies.
3. The loss of AC power supplies create the operating requirement for the DGs.
4. Non-ESF AC buses (4160VAC 11, 12) do not have an emergency power supply backup.
5. All initiators cause a turbine trip, which removes the unit generator from supplying the grid.

Notes for Table 4.2.2-4 (Continued):

6. A turbine trip causes 4160VAC bus feed from the UAT to transfer to the RAT, and removes the alternate power supply from the other 4160VAC buses.

7. The RPS motor-generators (MGs) are powered from 480VAC Buses 18 & 19. The diesel generators provide alternate power to these buses. A LOOP will leave the system power supply in a degraded state; however, RPS is a fail-safe system and a loss of power will lead to a reactor scram.

8. Support System has failed as a part of the IE definition.

9. LOCA events are expected to lead to a Group II PCIS Isolation.

10. Power for SW system components is provided by the following buses:

AC Power:

4160V

SWGR 13	Power to Pump 1-3901A
SWGR 14	Power to Pump 1-3901B, 1/2-3901

120V

MCC 18-2	Power for Strainers 1-3902, 1/2-3902 (stepped down)

DC Power:

125V

TB BUS 1A-1 (1B-1 alternate)	Power to close CB for Pump 1A
TB BUS 1B-1 (1A-1 alternate)	Power to close CB for Pumps 1B, 1/2

11. Power for RHRSW system components is provided by:

AC Power:

4160V

SWGR 13	Power to Pumps A, B
SWGR 14	Power to Pumps C, D

Notes for Table 4.2.2-4 (Continued):

480V

MCC 18-1B	Power for MO1-1001-4A, -5A, and -16A
MCC 19-4	Power for MO1-1001-4B, -5B, and -16B
MCC 18-2	RHRSW cubicle cooler fans for pumps 1-1001-65A, -65B
MCC 19-2	RHRSW cubicle cooler fans for pumps 1-1001-65C, -65D

DC Power:

125V

TB MN Bus 1A-1 (1B-1 alternate)	Control Power to close CB for RHRSW Pumps A, B
TB Res Bus 1B-1 (1A-1 alternate)	Control Power to close CB for RHRSW Pumps C, D

12. Cooling water for the TBCCW heat exchangers is provided by the service water system. Delayed failure on loss of SW.

13. The SW system provides cooling water to IA compressors and aftercoolers. Air compressors are assumed to fail immediately upon loss of cooling; systems with air receivers will continue to perform their function for some period after compressor failure.

14. Power to Pump 1A-3801 is provided by MCC 15-1. Power to Pump 1B-3801 is provided by MCC 17-2. 120VAC power for solenoid LCV-1-3801 is provided by MCC 17-1-1.

15. Instrument Air is necessary to shift the SW pump strainers and operate the flush and drain valves.

16. The alternate motive source is the condensate booster pumps (Unit 1 only). PCV 1-3499-32 provides this source and will fail closed on a loss of IA.

17. Clean demin system is completely shared between units. Power for the clean demineralizer pumps is provided by 480VAC power:

    480V

    MCC 16/26-1 Power to Clean Demin Pumps 1/2 A, B, and C

18. The condensate transfer pumps are powered from:

MCC 15-2	Pump 1/2-4301A and Pump 1/2-4301C
MCC 25-2	Pump 1/2-4301B
MCC 18-2	Power to the condensate transfer jockey pump and condensate transfer Pump 1A
MCC 19-2	Power to condensate transfer Pump 1B

Notes for Table 4.2.2-4 (Continued):

19. Power for the DGCW system pumps is provided by 480VAC buses:

480V

MCC 19-2	Power for DGCW Pump 1
MCC 18-3	Power for DGCW Pump 1/2
MCC 28	Alternate feed for DGCW Pump 1/2

20. Power for the ECCS Keep-Fill jockey pump is from 480VAC Bus MCC 18-1A.

21. Power to components of the IA system is as follows:

AC Power:

480V

SWGR 17	Power to Compressor 1-4709A
SWGR 28	Power to Compressor 1-4709B

120V

MCC 18-2-1	Control power for IA/SA crosstie valve AOV 1-4799-221

22. Instrument Air is required to open expansion tank level control valve. Manual bypass around level control valve is provided.

23. The Common Actuation System (CAS) provides actuation signals to HPCI, RHR, RCIC, ADS, and CS. The power supplies to CAS are:

Division I -	125VDC TB MN BUS 1A-1
Division II -	125VDC TB RES BUS 1B-1

Notes for Table 4.2.2-4 (Continued):

24. Power for components required for DG 1 support are as follows:

AC Power:

480V

MCC 19	Normal Power for Diesel Oil Transfer Pump 1, DGCW Pump 1, and DG Room Supply Fan 1
MCC 18-2	Power for Diesel Starting Air Compressor 1A, Turbocharger Lubricating Oil Pump, Circulating Lubricating Oil Pump, and cooling water immersion heater
MCC 19-2	Diesel starting air compressors 1A and 1B

DC Power:

125V

TB Res Bus 1B-1	Control and excitation, and related loads for DG 1
RB Dist Panel 1	Backup for control and excitation

Note: The DGs require DC to start (to run the fuel oil priming pump until 200 rpm), and to provide initial field excitation at 800 rpm. Breaker control power is also DC. Once running the DG does not require DC power. Breaker control is possible locally by manual action.

25. Power for components required for DG 1/2 support are as follows:

AC Power:

480V

MCC 18/28-1A	Power for Diesel Oil Transfer Pump 1/2, Power for DG Room Supply Fan 1/2, and starting air compressor 1/2B
MCC 18/28-3	Normal Power for DGCW Pump 1/2
MCC 18/28-3	Power for Turbocharger Lubricating Oil pump, circulating lubricating oil pump, and cooling water immersion heater.

DC Power:

125V

RB DC Panel 1	Control and excitation, and related loads for DG 1/2
RB DC Panel 2	Alternate power

Note: The DGs require DC to start (to run the fuel oil priming pump until 200 rpm), and to provide initial field excitation at 800 rpm. Breaker control power is also DC. Once running the DG does not require DC power. Breaker control is possible locally by manual action.

26. RBCCW supplies cooling water to Drywell Pneumatics compressors. RBCCW depends on SW and 480VAC Buses 18 and 19. There is a delayed dependency on a loss of RBCCW.

Notes for Table 4.2.2-4 (Continued):

27. TBCCW provides cooling water to SA compressors and aftercoolers. TBCCW depends on SW. Air compressors are assumed to fail immediately upon loss of cooling; systems with air receivers will continue to perform their function for some period after compressor failure.

28. Power supplies for various HVAC components are noted under the frontline or support systems listed as having a dependency upon HVAC; e.g., RHR, RHRSW, etc.

29. Condensate transfer system provides makeup to the hotwell. Backup is provided by the Service Water Standby Coolant Supply (SBCS) line.

30. Power to components of the SA system is as follows:

    480VAC

SWGR 17	Power to Compressor 1-4601A
SWGR 15	Power to Compressor 1-4601B

31. Makeup to the hotwell is via two air-operated valves (LCV 1-3301, 3302); air source is Instrument Air and control power is from the Essential Service Bus.

32. Drywell Pneumatics Air Compressors 1A and 1B are powered by 480VAC MCC 18-1 and 19-1, respectively. Nitrogen makeup supply provides backup to the Drywell Pneumatics system.

33. The 4160VAC non-ESF emergency buses receive DC control power from the 125VDC buses. A primary feed and alternate feed are provided. 4160VAC Bus 13-1 receives 125VDC control power for circuit breakers from RB Bus 1 with alternate power from TB Res Bus 1B-1. 4160VAC Bus 14-1 receives 125VDC control power for circuit breakers from TB Res Bus 1B-1 with alternate power from RB Bus 1. Transfer of control power is a manual action.

34. 125VDC control power for supply to Buses 18 and 19 are provided by RB Bus 1 and TB Res Bus 1B-1, respectively. MCC 18 receives alternate 125VDC control power from TB Res Bus 1B-1; MCC 19 receives alternate 125VDC control power from RB Bus 1.

35. 480VAC Buses 18 and 19 receive power from 4160VAC Buses 13-1 and 14-1, respectively. 480VAC Buses 18 and 19 can be crosstied together.

36. 250VDC Bus TB MCC 1 normally receives power from 250VDC Battery Charger 1 or it can receive power from Battery Charger 1/2, or the 250VDC Unit 1 batteries. Battery Charger 1 is powered by 480VAC MCC 19-2. Battery Charger 1/2 is powered by 480VAC MCC 18-2 (or 28-2). TB MCC 1 supplies 250VDC loads in both Units 1 and 2.

37. 250VDC Bus TB MCC 2 normally receives power from 250VDC Battery Charger 2 or it can receive power from Battery Charger 1/2, or the 250VDC Unit 2 batteries. Battery Charger 2 is powered by 480VAC MCC 29-2 (Unit 2 480VAC supplies are not shown in table). Battery Charger 1/2 is powered by 480VAC MCC 18-2 (or 28-2). TB MCC 2 supplies 250VDC loads in both Units 1 and 2.

Notes for Table 4.2.2-4 (Continued):

38. The 125VDC TB Main Bus 1A receives power from Battery Bus 1 which is normally powered by Battery Charger 1 or it can receive power from Battery Charger 1A, or the Unit 1 125VDC batteries. Battery Charger 1 is powered by 480VAC MCC 19-2. Battery Charger 1A is powered by 480VAC MCC 18-2. TB MN Bus 1A supplies 125VDC loads in both Units 1 and 2. The 125VDC TB Main Bus 2A receives power from Battery Bus 2 which is normally powered by Battery Charger 2 or it can receive power from Battery Charger 2A, or the Unit 2 125VDC batteries. Battery Charger 2 is powered by 480VAC MCC 29-2. Battery Charger 2A is powered by 480VAC MCC 28-2. TB MN Bus 2A supplies 125VDC loads in both Units 1 and 2. (Unit 2 480VAC supplies are not shown in table).

39. The 120/240V main instrument bus normally receives power via a 480VAC/120-240V transformer from 480VAC MCC 18-2. A reserve power supply is available from MCC 15-2 via the reserve instrument and RPS transformer.

40. The essential services bus normally receives power by 480VAC Bus 18 which is rectified to 250VDC and then inverted to 120VAC which is fed to the bus via a static switch and an ABT. The 250VDC TB MCC 1 also taps into the normal feed path between the rectifier and inverter. If the normal supply path fails, alternate AC feeds the static switch from 480V Bus 17 through a 120V regulator. A second alternate AC power supply comes from MCC 18-2 through a power-seeking Automatic Bus Transfer.

41. The RPS Buses A and B receive power via 480V Buses 18-2 and 19-2, respectively. Each uses a motor-generator (MG) to regulate the power supply to the sensors and instrumentation in the RPS. A flywheel on each MG prevents momentary losses of power to the MG set from generating a trip signal to the reactor. MCC 15-2 is an alternate power supply for the RPS Buses.

TABLE 4.2.2-5
SYSTEM/SYSTEM DEPENDENCY MATRIX

SYSTEM	U1/2 DEP	FW	COND	RPS	RPT	ARI	ERV	TR/SRV	SV	HPCI	ADS	RHR	CS	RCIC	SBLC	SBCS	CRD	SSMP	FP	SBGT	APCV
FW	I									R(3)								R(4)			
COND	I	C(2)																			
RPS	I																				
RPT	I					R(11)															
ARI	I				R(11)																
ERV	I										P(10)										
TR/SRV	I										P(10)										
SV	I																				
HPCI	I																	R(5)			
ADS	I											P(6)	P(6)								
RHR	X																				
CS	I																				
RCIC	I																				
SBLC	I																				
SBCS	I	D(7)	D(7)																		
CRD	I																				
SSMP	S																				
FP	S											A(8)						A(12)			
SBGT	I																				
APCV	I																			A(9)	

Notes for Table 4.2.2-5:

1. See Note 1 of Table 4.2.2-1 for descriptions of the dependency codes used in this table. Unit 1/Unit 2 Dependency (U1/2 DEP), is used in the system/system matrices to indicate the degree of inter-unit dependence for each system. The systems at the Quad Cities Site are noted on the matrix as belonging to one of three categories:

   I Independent systems are those that function solely at the unit where they are located. For example, the HPCI systems at Units 1 and 2 are independent.

   S Shared systems normally serve both units simultaneously (e.g., the service water system).

   X Crosstied systems are those that normally operate independently at each unit but are capable of serving as a backup to the opposite unit's system by supplying support through a crosstie between the systems (e.g., the Instrument Air system at each unit).

2. Feedwater suction is dependent on the condensate pumps.

3. HPCI flows through two "B" feedwater header check valves.

4. SSMP flows through two "B" feedwater header check valves.

5. SSMP uses one HPCI check valve.

6. For high-pressure sequences, RHR and CS will not be able to inject without ADS.

7. In limited scenarios, feedwater and condensate pumps will fail after pumping down the hotwell without SBCS.

8. FP can be used as an alternate water injection source to the RPV through RHR piping and a spool piece.

9. The APCV provides an alternate means of depressurizing the containment.

10. ADS uses the ERV and TR/SRV in ADS mode of operation.

11. RPT and ARI use the same ATWS system sensors. Upon exceeding 1250 psig in the reactor vessel or upon reaching -59 inches reactor water level, RPT trips the recirculation pumps and ARI energizes valves to vent the scram air header, initiating a reactor scram.

12. FP is an alternate suction source and room cooler supply.

TABLE 4.2.2-6
FRONTLINE SYSTEM/SUPPORT SYSTEM DEPENDENCY MATRIX - UNIT 1

SUPPORT SYSTEM	U1/2 DEP	FW	COND	RPS
TBCCW	I	C(2)	D(2)	
SW	S			
RHRHX	I			
DGCW	X			
ECCS FILL	I			
CLEAN DEMIN	S			
COND TRANS	S			
INST AIR	X	P(3)	D(3)	
DRYWELL PNEUMATICS	I			
SERVICE AIR	S			
HVAC	I	D(4)		
CCST	S	D(5)	D(3)	
SUPP POOL	I			
HOTWELL	I	C(6)	C(5)	
CAS	I			
OFFSITE POWER	S			
DG 1	I			
DG 1/2	S			
NON-ESF AC BUSES	I	C(7)		
NON-EMERGENCY ESF AC BUSES	I		C(4)	A(2)
4160VAC BUS 13-1	I			
4160VAC BUS 14-1	X			
480VAC BUS 18	I		D(4)	P(2)
480VAC BUS 19	I		D(4)	P(2)
250VDC TBMCC1	S			
250VDC TBMCC2	S			
125VDC MN BUS 1A	S	P(8)	P(4)	P(3)
125VDC MN BUS 2A	S	P(8)	P(4)	P(3)
120/240 MN INST BUS	I			
ESSENTIAL SERVICE BUS	I	P(3)		
RPS BUSES	I			C(2)

Notes for Table 4.2.2-6:

1. See Note 1 of Table 4.2.2-1 and Note 1 of Table 4.2.2-5 for descriptions of the dependency codes used in this table.

FW

2. Loss of flow in TBCCW system would result in loss of cooling to RFP oil coolers and overheating of RFP bearings due to hot feedwater that is being pumped.

3. Instrument Air supplies all air-operated valves in FW&C systems. Feedwater Regulating Valves fail as-is on loss of IA or control signal. The RFP recirculation valves fail open on loss of Instrument Air or control power and the makeup and emergency makeup valves fail closed.

4. Interlocked with RFPs to prevent start unless ventilation fan is running. Ventilation fans required for air cooling for RFP motors. RFP vent fan 1A is powered from MCC 18-3; RFP vent fan 1B is powered from MCC 19-2. One of the two vent fans are required for two RFPs.

5. Low hotwell level results in delayed dependence upon the CCST. Makeup water is usually supplied by vacuum drag from the CCST, via the normal makeup valve. Emergency makeup water supplied by two hotwell makeup pumps when emergency makeup valve opens on a low-low level signal for hotwell.

6. Feedwater pumps require condensate and condensate booster pumps to provide sufficient NPSH, or pumps trip on low suction pressure. Depletion of hotwell would result in low suction pressure.

7. RFP 1A is supplied by 4160VAC SWGR 11; RFP 1B is supplied by 4160VAC SWGR 12; RFP 1C can be powered from either 4160VAC SWGR 11 or 12.

8. 125VDC power is required to close 4160VAC pump circuit breakers to provide motive power to start non-operating pumps. Assuming RFP 1A and 1B are normally operating, standby pump RFP 1C, when it is to be powered from SWGR 11, requires 125VDC Main Bus 1A-2 to close its breaker; when powered from SWGR 12, it requires 125VDC Reserve Bus 1B-2.

COND

2. Loss of flow in TBCCW system would result in delayed failure of condensatė pumps due to loss of pump seal cooling.

3. Low hotwell level results in delayed dependence upon the CCST. Makeup water is usually supplied by vacuum drag from the CCST, via the normal makeup valve (operated by Instrument Air). Emergency makeup water supplied by two hotwell makeup pumps when emergency makeup valve opens on a low-low level signal for hotwell.

4. Condensate system components are powered by the following buses:

AC Power:

4160V

SWGR 13	Power to condensate and condensate booster Pumps 1A and 1B
SWGR 14	Power to condensate and condensate booster Pumps 1C and 1D

480V

SWGR 16	Power to condenser vacuum pump
SWGR 17	Power to condensate demineralizer air compressor
MCC 15-2	Power to various pumps and motor-operated valves
MCC 16-1	Power to various pumps and motor-operated valves
MCC 17-1	Power to various pumps and motor-operated valves

Notes for Table 4.2.2-6 (Continued):

COND (continued)

125VDC Power:

MN Bus 1A-1	Control power to components powered by SWGR 13
Reserve Bus 1B-1	Control power to components powered by SWGR 14

5. The operator will trip the condensate pumps on low hotwell level.

RPS

2. Normal power to RPS Buses A and B is supplied by two MG sets. The A and B MG sets are powered from MCCs 18-2 and 19-2, respectively. Reserve power to RPS buses is supplied from MCC 15-2.

3. Backup scram valve solenoids are powered from 125VDC Buses 1A-1 and 1B-1.

TABLE 4.2.2-7
SAFETY SYSTEM/SUPPORT SYSTEM DEPENDENCY MATRIX

SUPPORT SYSTEM	U1/2 DEP	RPT	ARI	ERV	TR/SRV	SV	HPCI	ADS	RHR A	RHR B	CS A	CS B	RCIC
TBCCW	I												
SW	S												
RHRHX	X								D(7)	D(7)	D(7)	D(7)	
DGCW	X								D(6)	D(6)	D(6)	D(6)	D(7)
ECCS FILL	I								P(3)	P(3)	P(2)	P(2)	
CLEAN DEMIN	S												
COND TRANS	S												
INST AIR	X						P(5)						P(4)
DRYWELL PNEUMATICS	I							P(2)					
SERVICE AIR	S												
HVAC	I						D(3)		D(6)	D(6)	D(7)	D(7)	D(7)
CCST	S						C(6)		A(5)	A(5)	A(4)	A(4)	C(6)
SUPP POOL	I						A(6)		C(5)	C(5)	C(4)	C(4)	A(6)
HOTWELL	I												
CAS	I						C(4)	C(3)	C(4)	C(4)	C(3)	C(3)	C(8)
OFFSITE POWER	S												
DG 1	I												
DG 1/2	S												
NON-ESF AC BUSES	I												
NON-EMERGENCY ESF AC BUSES	I												
4160VAC BUS 13-1	I								C(8)		C(5)		
4160VAC BUS 14-1	X									C(8)		C(5)	
480VAC BUS 18	I								C(8)		C(5)		
480VAC BUS 19	I						D(3)			C(8)		C(5)	
250VDC TBMCC1	S								P(8)	P(8)			C(2)
250VDC TBMCC2	S						C(7)						
125VDC MN BUS 1A	S	P(2)	P(2)	C(2)			A(7)	P(4)	C(8)	P(8)	C(5)		C(3)
125VDC MN BUS 2A	S	P(2)	P(2)	A(2)			C(7)	P(4)	P(8)	C(8)		C(5)	P(3)
120/240 MN INST BUS	I	A(2)	A(2)										
ESSENTIAL SERVICE BUS	I	A(2)	A(2)										
RPS BUSES	I												

Notes for Table 4.2.2-7:

1. See Note 1 of Table 4.2.2-1 and Note 1 of Table 4.2.2-5 for descriptions of the dependency codes used in this table.

RPT & ARI

2. The ATWS Division I logic is powered from 125VDC RB DC Panel 1 with alternate power from the 120VAC Instrument Bus. ATWS Division II logic is powered from TB Reserve Bus 1B-1 with alternate power from the 120VAC Essential Services Bus. The ATWS valves and the RPT field breaker relay are powered by 125VDC from Bus 11, 12 control power.

ERV

2. Electromatic relief valves require 125VDC power to open. Normal supply is from TB MN Bus 1A with alternate power available from TB Reserve Bus 1B-1.

HPCI

2. Failure of Keep-Fill system could potentially cause damage to front-line system at startup due to water hammer if HPCI is lined-up to take a suction from the torus.

3. Delayed dependency on HVAC for room cooling, but is insignificant for HPCI mission time. Power for HPCI area cooler supplied by 480V MCC 19-4 and cooling is provided by service water with DGCW as an alternate.

4. HPCI system auto-starts on low-low reactor water level (-59 inches) or high drywell pressure (2.5 psig).

5. Instrument Air is required to maintain HPCI steamline drain valves 29, 30, 64, and 65 open and steam line drain valves 28 and 31 closed. Valves automatically reposition on HPCI initiation and on loss of IA. The effect of the reposition is to bypass the steamline trap, and redirect condensate to the torus.

6. Initial source of water is CCST. An alternate suction is the suppression pool. Operators are directed by QGA 100 to use the CCST suction if available and will defeat the high torus level transfer.

7. Power supplies for HPCI components and logic are as follows:

125VDC TB 1A-1	Reserve Feed to HPCI Sys Logic and Control Panel 901-39
125VDC TB 1B-1	Main Feed to HPCI Sys Logic and Control Panel 901-39
120/240VAC Ess Bus	HPCI Instrument Panels 901-3, 901-19
120/240VAC Inst Bus	Panels 901-3, 901-4, 901-19, 901-38, 901-39

Notes for Table 4.2.2-7 (Continued):

<u>HPCI</u> (Continued)

250VDC RB MCC 1A	Gland Seal Hotwell Pump	
	Gland Seal Leak-Off Blower	
	Auxiliary Lube Oil Pump	
	Emergency Lube Oil Pump	
	MO1-2301-4	MO1-2301-3
	MO1-2301-6	MO1-2301-8
	MO1-2301-14	MO1-2301-35
	MO1-2301-36	MO1-2301-48
	MO1-2301-9	MO1-2301-10
	MO1-2301-49	

<u>ADS</u>

2. The Target Rock SRV requires air from the Drywell Pneumatics System to open valve for ADS mode of operation. The TR/SRV has an accumulator at the valve, which allows up to five valve operations when the drywell is at atmospheric pressure.

3. ADS auto-initiates after 120 seconds on reactor water level less than -59 inches <u>AND</u> drywell pressure above 2.5 psig if a low pressure ECCS pump is running with 100-psig discharge; or on low-low reactor water level (-59 inches) alone after 8.5 minutes if a low pressure ECCS pump is running.

4. 125VDC power is required for ADS logic and valve solenoids as noted:

   Logic:

Div. I	TB Main Bus 1A-1
Div. II	TB Reserve Bus 1B-1 (TB MN Bus 1A-1 alternate)

   Valves:

Normal	TB Main Bus 1A-1
Alternate	TB Reserve Bus 1B-1

Notes for Table 4.2.2-7 (Continued):

RHR

2. RHR used in the injection cooling mode requires the RHR heat exchangers (RHRHX) for transfer of decay heat to ultimate heat sink. In this mode, RHR provides containment heat removal function, as well.

3. Failure of ECCS Fill system could potentially cause damage to frontline system at startup due to water hammer. Tech Specs require the ECCS Fill system to be in operation in order to consider RHR operable.

4. Common Actuation sensors auto-start RHR pumps on high drywell pressure (2.5 psig) OR low-low reactor water level (-59 inches) AND reactor low pressure (<325 psig) OR low-low reactor water level maintained for greater than 8.5 minutes.

5. Initial source of water is the suppression pool; suction can be switched to CCST manually if required.

6. Room cooling is supplied by DGCW. Power supplies to the AHUs are MCC 18-1A for 1-5746A and MCC 19-1 for 1-5746B.

7. The RHR pumps will eventually fail due to loss of NPSH when the suppression pool heats up.

8. RHR system components are powered by the following buses:

AC Power:

4160V

Bus 13-1	RHR Pumps 1A, 1B-1002
Bus 14-1	RHR Pumps 1C, 1D-1002

480V

MCC 18-1B	MO1-1001-7A	MO1-1001-7B
	MO1-1001-16A	MO1-1001-18A
	MO1-1001-19A	MO1-1001-23A
	MO1-1001-26A	MO1-1001-34A
	MO1-1001-36A	MO1-1001-37A
	MO1-1001-43A	MO1-1001-43B
	MO1-1001-50	
MCC 18/19-5	MO1-1001-28A	MO1-1001-28B
	MO1-1001-29A	MO1-1001-29B

Notes for Table 4.2.2-7 (Continued):

RHR (Continued)

MCC 19-4	MO1-1001-7C	MO1-1001-7D
	MO1-1001-16B	MO1-1001-18B
	MO1-1001-19B	MO1-1001-23B
	MO1-1001-26B	MO1-1001-34B
	MO1-1001-36B	MO1-1001-37B
	MO1-1001-43C	MO1-1001-43D

DC Power:

250V

MCC 1B	MO1-1001-47

125V

RB DC PNL 1	RHR Pumps 1A, 1B circuit breaker control power (main feed)
	RHR Pumps 1C, 1D circuit breaker control power (alternate feed)
TB Reserve Bus 1B-1	RHR Pumps 1C, 1D circuit breaker control power (main feed)
	RHR Pumps 1A, 1B circuit breaker control power (alternate feed)
	RHR Loop B System II initiation circuitry
	Permissive circuitry
TB MN Bus 1A-1	RHR Loop A System I initiation circuitry
	Permissive circuitry

Notes for Table 4.2.2-7 (Continued):

CS

2. Failure of ECCS Fill system could potentially cause damage to frontline system at startup due to water hammer. Tech Specs require the ECCS Fill system to be in operation in order to consider Core Spray operable.

3. Common Actuation sensors auto-start Core Spray pumps on high drywell pressure (2.5 psig) OR low-low reactor water level (-59 inches) AND reactor low pressure (<325 psig) OR low-low reactor water level maintained for greater than 8.5 minutes.

4. Initial source of water is the suppression pool; suction can be switched to CCST manually if required.

5. CS System Components are powered by the following buses:

AC Power:

4160V

SWGR 13-1	Power to Pump 1A-1401
SWGR 14-1	Power to Pump 1B-1401

480V

MCC 18-1A	Power to Valves MO1-1402-3A, 4A, 24A, 25A
MCC 19-1	Power to Valves MO1-1402-3B, 4B, 24B, 25B

120V

MCC 18-1A-1	MO1-1402-38A
MCC 19-1-1	MO1-1402-38B

DC Power:

125V

RB DC PNL 1	Power to close circuit breaker for CS Pump 1A-1401
TB Reserve Bus 1B-1	Power to close circuit breaker for CS Pump 1B-1401; also used for System II initiation control logic
MN Bus 1A-1	Power for System I initiation control logic

6. Room cooling is supplied by DGCW. Power supplies to the AHUs are MCC 18-1A for 1-5748A and MCC 19-1 for 1-5748B.

7. The CS pumps will eventually fail due to loss of NPSH when the suppression pool heats up.

Notes for Table 4.2.2-7 (Continued):

RCIC

2. 250VDC MCC 1B power is required as follows:

   RCIC Turbine Control Power

   MO1-1301-17 MO1-1301-49
   MO1-1301-22 MO1-1301-53
   MO1-1301-25 MO1-1301-60
   MO1-1301-26 MO1-1301-61
   MO1-1301-48 MO1-1301-62

   Gland Seal Condenser Condensate Pump

   Gland Seal Condenser Vacuum Pump

3. 125VDC power is required as follows:

   125VDC Bus 1A-2 supplies control power to the following air-operated valves:

   AO1-1301-12 AO1-1301-34
   AO1-1301-13 AO1-1301-35
   AO1-1301-32

   125VDC Bus 1B-1 supplies power to the turbine trip and isolation circuits.

4. Instrument Air supplies the following valves:

   AO1-1301-12 AO1-1301-34
   AO1-1301-13 AO1-1301-35
   AO1-1301-32

5. MO1-1301-16 is supplied from 480VAC MCC 18-1A-1.

6. Initial source of water is CCST. An alternate suction is the suppression pool. Operators are directed by QGA-100 to use the CCST suction if available and will defeat the high torus level transfer.

7. Room cooling is supplied from 480VAC MCC 18-1A and is cooled by DGCW.

8. Common Actuation sensors auto-start the RCIC pump low-low reactor water level (-59 inches).

TABLE 4.2.2-8
ADDITIONAL SYSTEMS/SUPPORT SYSTEM DEPENDENCY MATRIX

SUPPORT SYSTEM	U1/2DEP	SBLC	SBCS	CRD	SSMP	FP	SBGT	APCV
TBCCW	I			D(3)				
SW	S		C(2)		D(4)	P(2)		
RHRSW	I							
DGCW	X							
ECCS FILL	I							
CLEAN DEMIN	S							
COND TRANS	S							
INST AIR	X						C(3)	D(2)
DRYWELL PNEUMATICS	I							
SERVICE AIR	S							
HVAC	I				D(4)			
CCST	S			A(5)	P(2)			
SUPP POOL	I							
HOTWELL	I			P(5)				
CAS	I							
OFFSITE POWER	S							
DG 1	I							
DG 1/2	S							
NON-ESF AC BUSES	I							
NON-EMERGENCY ESF AC BUSES	I		C(4)	C(2) P(4)				
4160VAC BUS 13-1	I							
4160VAC BUS 14-1	X				P(3)			
480VAC BUS 18	I	P(2)						
480VAC BUS 19	I	P(2)					C(2)	
250VDC TBMCC1	S					P(2)		
250VDC TBMCC2	S							
125VDC MN BUS 1A	S							
125VDC MN BUS 2A	S							
120/240 MN INST BUS	I							
ESSENTIAL SERVICE BUS	I					D(3)	C(3)	C(3)
RPS BUSES	I							

Notes for Table 4.2.2-8:

1. See Note 1 of Table 4.2.2-1 and Note 1 of Table 4.2.2-5 for descriptions of the dependency codes used in this table.

SBLC

2. SBLC Pump A and Explosive Valve 1-1106A supplied by 480VAC MCC 18-1A.
SBLC Pump B and Explosive Valve 1-1106B supplied by 480VAC MCC 19-1.
SBLC Tank heater 1-1103 supplied by 480VAC MCC 19-1.

SBCS

2. Secondary water makeup source to hotwell (Primary is CCST). Requires operable SW pump.

3. 480VAC power MCC 17-1 provides power to open normally closed MOV 1-3901, 3902 valves to Unit 1 hotwell.

CRD

2. CRD pump motor 1A is powered from SWGR 13.
CRD pump motor 1B is powered from SWGR 14.

3. Cooling Water to oil coolers and thrust bearings is provided by TBCCW.

4. CRD valves are powered from:

MCC 15-2	MO1-0301-2A
MCC 17-2	MO1-301-2B

5. CRD suction is maintained by the 60-gpm hotwell reject. If the hotwell reject is not available, CRD suction will be from the CCST.

SSMP

2. Normal suction is from the CCST, alternate suction is from the Fire Main.

3. SSMP pump motor is powered from SWGR 31. MCC 30 provides power to MO1/2-2901-7 (test valve), MO1-2901-8 (Unit 1 reactor vessel supply valve), MO2-2901-8 (Unit 2 reactor vessel supply valve), and flow control valve MO1/2-2901-6. MCC 30 and 31 are supplied from SWGR 14-1 or 24-1.

4. SW supplies cooling water to the room air cooler, fire main is an alternate source of cubicle cooling.

FP

2. Unit 1 service water maintains pressure in fire main during normal plant conditions in addition to serving as a backup for the fire protection water system. Valve MO1/2-3906 provides this backup and is powered from 250VDC MCC 1.

3. Unit 1/2 fire pump and Unit 1 fire pump are started and controlled by DC power from two dedicated 24V batteries. 120VAC from the Instrument Bus is required for the battery charger for these batteries. Delayed dependence.

Notes for Table 4.2.2-8 (Continued):

SBGT

2. Isolation valve, air supply valve and Unit 1/2 Fan (A) is powered from MCC 19-4. Fan discharge valve and electric heater is powered from MCC 19-4.

3. The containment vent valves to SGBT valves are supplied by Instrument Air. Control power to these valves is supplied from the ESS Bus.

APCV

2. The valves in the Augmented Primary Containment vent system are all air-operated valves supplied from Instrument Air. Accumulators on each valve operator allow actuations on a loss of Instrument Air.

3. Control power is supplied from the Essential Services Bus.

TABLE 4.2.2-9
SUPPORT SYSTEM/SUPPORT SYSTEM DEPENDENCY MATRIX

SUPPORT SYSTEM	U1/2 DEP	TBCCW	SW	RHRHX	DGCW	ECCS FILL	CLEAN DEMIN	COND TRANS	INST AIR	DW AIR	SA	HVAC	CCST	SUPP POOL	HW	CAS	DG 1	DG 1/2	NON-ESF BUS	NON-EMERG ESF BUS	4160 13-1	4160 14-1	480V 18	480V 19	250V TB MCC1	250V TB MCC2	125VDC MN BUS 1A	125VDC MN BUS 2A	INST BUS	ESS BUS RPS BUS
TBCCW	I										D(2)																			
SW	S	D(2)							D(2)	D(2)		P(3)			A(2)															
RHRHX	I																													
DGCW	X											P(3)					C(2)	C(2)												
ECCS FILL	I																													
CLEAN DEMIN	S	D(3)																												
COND TRANS	S					A(2)									C(2)															
INST AIR	X	D(4)	D(3)					A(4)							C(3)															
DRYWELL PNEUMATICS	I																													
SERVICE AIR	S								A(3)																					
HVAC	I			D(2)													D(3)	D(3)												
CCST	S							P(3)																						
SUPP POOL	I																													
HOTWELL	I							A(4)																						
CAS	I																													
OFFSITE PWR	S																		C(2)	C(2)	P(2)	P(2)								
DG 1	I																			A(3)		A(3)								
DG 1/2	S																			A(3)	A(3)									
NON-ESF AC BUSES	I																													
NON-EMERGENCY ESF AC BUSES	I	C(5)	C(2)	C(3)			C(2)	C(2)	P(4)		C(3)	P(2)																	A(2)	A(2) A(2)
4160VAC BUS 13-1	I																						C(2)							
4160VAC BUS 14-1	X																							C(2)						
480VAC BUS 18	I		D(2)	P(3)	P(2)	C(3)		P(2)	A(4)	P(2,3)		P(2)					P(4)	C(4)						A(2)	A(2)	A(2)	A(2)		P(2)	P(2) P(2)
480VAC BUS 19	I			P(3)	C(2)					P(2,3)		P(2)					C(4)						A(2)		P(2)		D(2)			 P(2)
250VDC TBMCC1	S		P(4)																											A(2)
250VDC TBMCC2	S																													
125VDC MN BUS 1A	S		P(2)	P(3)												P(2)	A(4)	C(4)	P(3)	P(4)	C(4)	A(4)	C(3)	A(3)						
125VDC MN BUS 2A	S		P(2)	P(3)												P(2)	C(4)	A(4)	P(3)	P(4)	A(4)	C(4)	A(3)	C(3)						
120/240 MN INST BUS	I																													
ESSENTIAL SERVICE BUS	I														C(3)															
RPS BUSES	I																													

Notes for Table 4.2.2-9:

1. See Note 1 of Table 4.2.2-1 and Note 1 of Table 4.2.2-5 for descriptions of the dependency codes used in this table.

TBCCW

2. Cooling water for the TBCCW heat exchangers is provided by the service water system. Delayed failure on loss of SW.

3. Clean demineralized water is used for makeup to the TBCCW Expansion Tank. Delayed failure if level cannot be maintained.

4. Instrument Air is required to open expansion tank level control valve. Manual bypass around level control valve is provided.

5. Power to Pump 1A-3801 is provided by MCC 15-1. Power to Pump 1B-3801 is provided by MCC 17-2. 120VAC power for solenoid LCV-1-3801 is provided by MCC 17-1-1.

SW

2. Power for SW system components is provided by the following buses:

   AC Power:

   4160V

   SWGR 13 Power to Pump 1-3901A
   SWGR 14 Power to Pump 1-3901B, 1/2-3901

   120V

   MCC 18-2 Power for Strainers 1-3902, 1/2-3902 (stepped down)

   DC Power:

   125V

   TB BUS 1A-1 (1B-1 alternate) Power to close CB for Pump 1A
   TB BUS 1B-1 (1A-1 alternate) Power to close CB for Pumps 1B, 1/2

3. Instrument Air is necessary to shift the SW pump strainers and operate the flush and drain valves.

4. Power for MOV 1A-3903, 1B-3903, 1-3904, 1-3905 is supplied by 250VDC MCC 1. These valves are used to isolate large non-essential loads from the SW header.

Notes for Table 4.2.2-9 (Continued):

<u>RHRHX</u>

2. Delayed dependency on loss of HVAC to RHRSW pumps cubicle coolers.

3. Power for RHRSW system components is provided by:

AC Power:

4160V

SWGR 13	Power to Pumps A, B
SWGR 14	Power to Pumps C, D

480V

MCC 18-1B	Power for MO1-1001-4A, -5A, and -16A
MCC 19-4	Power for MO1-1001-4B, -5B, and -16B
MCC 18-2	RHRSW cubicle cooler fans for pumps 1-1001-65A, -65B
MCC 19-2	RHRSW cubicle cooler fans for pumps 1-1001-65C, -65D

DC Power:

125V

TB MN Bus 1A-1 (1B-1 alternate)	Control Power to close CB for RHRSW Pumps A, B
TB Res Bus 1B-1 (1A-1 alternate)	Control Power to close CB for RHRSW Pumps C, D

<u>DGCW</u>

2. Power for the DGCW system pumps is provided by 480VAC buses:

480V

MCC 19-2	Power for DGCW Pump 1
MCC 18-3	Power for DGCW Pump 1/2
MCC 28	Alternate feed for DGCW Pump 1/2

Notes for Table 4.2.2-9 (Continued):

ECCS Keep-Fill

2. The condensate transfer jockey pump provides a backup supply for the ECCS Keep-Fill system.

3. Power for the ECCS Keep-Fill jockey pump is from 480VAC Bus MCC 18-1A.

Clean Demin

2. Clean demin system is completely shared between units. Power for the clean demineralizer pumps is provided by 480VAC power:

   480V

   MCC 16/26-1 Power to Clean Demin Pumps 1/2 A, B, and C

Cond Trans

2. The condensate transfer pumps are powered from:

MCC 15-2	Pump 1/2-4301A and Pump 1/2-4301C
MCC 25-2	Pump 1/2-4301B
MCC 18-2	Power to the condensate transfer jockey pump and condensate transfer Pump 1A
MCC 19-2	Power to condensate transfer Pump 1B

3. The suction for the condensate transfer pumps is the CCST.

4. The alternate motive source is the condensate booster pumps (Unit 1 only). PCV 1-3499-32 provides this source and will fail closed on a loss of IA.

Notes for Table 4.2.2-9 (Continued):

<u>Inst Air</u>

2. The SW system provides cooling water to IA compressors and aftercoolers. Air compressors are assumed to fail immediately upon loss of cooling; systems with air receivers will continue to perform their function for some period after compressor failure.

3. Service Air system provides a backup to the IA system.

4. Power to components of the IA system is as follows:

   AC Power:

   480V

   SWGR 17 Power to Compressor 1-4709A
   SWGR 28 Power to Compressor 1-4709B

   120V

   MCC 18-2-1 Control power for IA/SA crosstie valve AOV 1-4799-221

<u>Drywell Pneumatics</u>

2. RBCCW supplies cooling water to Drywell Pneumatics compressors. RBCCW depends on SW and 480VAC Buses 18 and 19. There is a delayed dependency on a loss of RBCCW.

3. Drywell Pneumatics Air Compressors 1A and 1B are powered by 480VAC MCC 18-1 and 19-1, respectively. Nitrogen makeup supply provides backup to the Drywell Pneumatics system.

Notes for Table 4.2.2-9 (Continued):

Service Air

2. TBCCW provides cooling water to SA compressors and aftercoolers. Air compressors are assumed to fail immediately upon loss of cooling; systems with air receivers will continue to perform their function for some period after compressor failure.

3. Power to components of the SA system is as follows:

   480VAC

   SWGR 17 Power to Compressor 1-4601A
   SWGR 15 Power to Compressor 1-4601B

HVAC

2. Power supplies for various HVAC components are noted under the frontline or support systems listed as having a dependency upon HVAC; e.g., RHR, RHRSW, etc.

3. HPCI emergency air coolers normally supplied by SW can be supplied by DGCW through check valves on loss of SW.

HW

2. Condensate transfer system provides makeup to the hotwell. Backup is provided by the Service Water Standby Coolant Supply (SBCS) line.

3. Makeup to the hotwell is via two air-operated valves (LCV 1-3301, 3302); air source is Instrument Air and control power is from the Essential Service Bus.

CAS

2. The Common Actuation System (CAS) provides actuation signals to HPCI, RHR, RCIC, ADS, and CS. The power supplies to CAS are:

   Division I - 125VDC TB MN BUS 1A-1

   Division II - 125VDC TB RES BUS 1B-1

Notes for Table 4.2.2-9 (Continued):

<u>DG 1</u>

2. Diesel Cooling Water (DGCW) Pump 1 provides cooling to DG 1. Failure after a short delay if lost. The Service Water return piping is used, but requires no valves for return water to the river.

3. DG Room Supply Fan 1 provides HVAC for DG 1. Leads to loss of engine control circuitry on prolonged loss of ventilation.

4. Power for components required for DG 1 support are as follows:

   AC Power:

   480V

   MCC 19 Normal Power for Diesel Oil Transfer Pump 1, DGCW Pump 1, and DG Room Supply Fan 1
   MCC 18-2 Power for Diesel Starting Air Compressor 1A, Turbocharger Lubricating Oil Pump, Circulating Lubricating Oil Pump, and cooling water immersion heater
   MCC 19-2 Diesel starting air compressors 1A and 1B

   DC Power:

   125V

   TB Res Bus 1B-1 Control and excitation, and related loads for DG 1
   RB Dist Panel 1 Backup for control and excitation

   Note: The DGs require DC to start (to run the fuel oil priming pump until 200 rpm), and to provide initial field excitation at 800 rpm. Breaker control power is also DC. Once running the DG does not require DC power. Breaker control is possible locally by manual action.

Notes for Table 4.2.2-9 (Continued):

DG 1/2

2. Diesel Cooling Water (DGCW) pump 1/2 provides cooling to DG 1/2. Failure after a short delay if lost. The service water return piping is used, but requires no valves for return water to the river.

3. DG room supply fan 1/2-5727 provides HVAC for DG 1/2. Leads to loss of engine control circuitry on prolonged loss of ventilation.

4. Power for components required for DG 1/2 support are as follows:

AC Power:

480V

MCC 18/28-1A	Power for Diesel Oil Transfer Pump 1/2, Power for DG Room Supply Fan 1/2, and starting air compressor 1/2B
MCC 18/28-3	Normal Power for DGCW Pump 1/2
MCC 18/28-3	Power for Turbocharger Lubricating Oil pump, circulating lubricating oil pump, and cooling water immersion heater.

DC Power:

125V

RB DC Panel 1	Control and excitation, and related loads for DG 1/2
RB DC Panel 2	Alternate power

Note: The DGs require DC to start (to run the fuel oil priming pump until 200 rpm), and to provide initial field excitation at 800 rpm. Breaker control power is also DC. Once running the DG does not require DC power. Breaker control is possible locally by manual action.

Non-ESF Buses

2. Offsite power sources provides power for 4160VAC Buses 11 and 12 via the unit auxiliary transformer (T11) if the unit generator is operating or the reserve auxiliary transformer (T12).

3. Control power is supplied from the 125VDC buses. The normal control power for 4160VAC Bus 11 is supplied from TB 125VDC Bus 1A-2 with alternate power from TB Res Bus 1B-2. The normal control power for 4160VAC Bus 12 is supplied from TB Res Bus 1B-2 with alternate power from TB 125VDC Bus 1A-2. Transfer of control power is a manual action.

Notes for Table 4.2.2-9 (Continued):

Non-emergency ESF Buses and 4160VAC Buses 13-1 and 14-1

2. Offsite power sources provide power for 4160VAC Buses 13 and 14 via the unit auxiliary transformer (T11) or the reserve auxiliary transformer (T12). Buses 13 and 14 provide normal feed to 4160VAC Buses 13-1 and 14-1.

3. 4160VAC Buses 13 and 14 can receive power from the onsite emergency DGs through manual connections with Buses 13-1 and 14-1, respectively. Bus 13-1 is automatically connected to Unit 1 emergency swing DG 1/2. Bus 14-1 is automatically connected to the Unit 1 emergency DG 1.

4. The 4160VAC non-ESF emergency buses receive DC control power from the 125VDC buses. A primary feed and alternate feed are provided. 4160VAC Bus 13-1 receives 125VDC control power for circuit breakers from RB Bus 1 with alternate power from TB Res Bus 1B-1. 4160VAC Bus 14-1 receives 125VDC control power for circuit breakers from TB Res Bus 1B-1 with alternate power from RB Bus 1. Transfer of control power is a manual action.

480VAC Buses 18 and 19

2. 480VAC Buses 18 and 19 receive power from 4160VAC Buses 13-1 and 14-1, respectively. 480VAC Buses 18 and 19 can be crosstied together.

3. 125VDC control power for supply to Buses 18 and 19 are provided by RB Bus 1 and TB Res Bus 1B-1, respectively. MCC 18 receives alternate 125VDC control power from TB Res Bus 1B-1; MCC 19 receives alternate 125VDC control power from RB Bus 1.

250VDC TBMCC1

2. 250VDC Bus TB MCC 1 normally receives power from 250VDC Battery Charger 1 or it can receive power from Battery Charger 1/2, or the 250VDC Unit 1 batteries. Battery Charger 1 is powered by 480VAC MCC 19-2. Battery Charger 1/2 is powered by 480VAC MCC 18-2 (or 28-2). TB MCC 1 supplies 250VDC loads in both Units 1 and 2.

250VDC TBMCC2

2. 250VDC Bus TB MCC 2 normally receives power from 250VDC Battery Charger 2 or it can receive power from Battery Charger 1/2, or the 250VDC Unit 2 batteries. Battery Charger 2 is powered by 480VAC MCC 29-2 (Unit 2 480VAC supplies are not shown in table). Battery Charger 1/2 is powered by 480VAC MCC 18-2 (or 28-2). TB MCC 2 supplies 250VDC loads in both Units 1 and 2.

125VDC TB MN Bus 1A and TB MN Bus 2A

2. The 125VDC TB Main Bus 1A receives power from Battery Bus 1 which is normally powered by Battery Charger 1 or it can receive power from Battery Charger 1A, or the Unit 1 125VDC batteries. Battery Charger 1 is powered by 480VAC MCC 19-2. Battery Charger 1A is powered by 480VAC MCC 18-2. TB MN Bus 1A supplies 125VDC loads in both Units 1 and 2. The 125VDC TB Main Bus 2A receives power from Battery Bus 2 which is normally powered by Battery Charger 2 or it can receive power from Battery Charger 2A, or the Unit 2 125VDC batteries. Battery Charger 2 is powered by 480VAC MCC 29-2. Battery Charger 2A is powered by 480VAC MCC 28-2. TB MN Bus 2A supplies 125VDC loads in both Units 1 and 2. (Unit 2 480VAC supplies are not shown in table).

Notes for Table 4.2.2-9 (Continued):

Inst Bus

2. The 120/240V main instrument bus normally receives power via a 480VAC/120-240V transformer from 480VAC MCC 18-2. A reserve power supply is available from MCC 15-2 via the reserve instrument and RPS transformer.

Essential Service Bus

2. The essential services bus normally receives power by 480VAC Bus 18 which is rectified to 250VDC and then inverted to 120VAC which is fed to the bus via a static switch and an ABT. The 250VDC TB MCC 1 also taps into the normal feed path between the rectifier and inverter. If the normal supply path fails, alternate AC feeds the static switch from 480V Bus 17 through a 120V regulator. A second alternate AC power supply comes from MCC 18-2 through a power-seeking Automatic Bus Transfer.

RPS Buses

2. The RPS Buses A and B receive power via 480V Buses 18-2 and 19-2, respectively. Each uses a motor-generator (MG) to regulate the power supply to the sensors and instrumentation in the RPS. A flywheel on each MG prevents momentary losses of power to the MG set from generating a trip signal to the reactor. MCC 15-2 is an alternate power supply for the RPS Buses.

TABLE 4.4.1-3
QUAD CITIES-SPECIFIC COMPONENT FAILURE RATES
Periods of Record: Jan 1985 through Dec 1991, or Jan 1993 through Dec 1995 (*)

COMPONENT TYPE GROUPING AND FAILURE MODE	SYSTEM	COMPONENT TYPE	FAILURE CODE[4]	TOTAL NUMBER OF COMPONENTS[5]	NUMBER OF FAILURES[3]	NUMBER OF DEMANDS OR HOURS	FAILURE RATE
U-1 Diesel Generator Failure to Start	DG	DG	A	1	3	212D	1.42×10^{-2}/D *
U-1 Diesel Generator Failure to Run	DG	DG	X	1	1/2	490H	1.02×10^{-3}/H *
U-2 Diesel Generator Failure to Start	DG	DG	A	1	3	212D	1.42×10^{-2}/D *
U-2 Diesel Generator Failure to Run	DG	DG	X	1	1/2	490H	1.02×10^{-3}/H *
1/2 Diesel Generator Failure to Start	DG	DG	A	1	3	212D	1.42×10^{-2}/D *
1/2 Diesel Generator Failure to Run	DG	DG	X	1	1/2	490H	1.02×10^{-3}/H *
Diesel Generator Cooling Water Pump Failure to Start	DG	PM	A	3	4	301D	1.33×10^{-2}/D *
Diesel Generator Cooling Water Pump Failure to Run/Function	DG	PM	X	3	1	552H	1.81×10^{-3}/H *
Diesel Generator Output Breaker Fails to Function	DG	CB	F	4	1	191H	5.24×10^{-3}/H *
HPCI Turbine Failure to Start	HI	PT	A	2	2	53D	3.77×10^{-2}/D *
HPCI Turbine Failure to Run[2]	HI	PT	X	2	1/2	27H	2.77×10^{-3}/H * (7)
Condensate/Condensate Booster Pumps Failure to Start[1]	FW	PM	A	8	1/2	866D	5.77×10^{-4}/D
Condensate/Condensate Booster Pumps Failure to Run[1]	FW	PM	X	16	1/2	592242H	8.44×10^{-7}/H
RHR Pumps Failure to Start	RH	PM	A	8	1/2	1128D	4.43×10^{-4}/D *
RHR Pump Failure to Run	RH	PM	X	8	1/2	8666H	5.77×10^{-5}/H *
Reactor Feed Pump Failure to Start	FW	PM	A	6	1/2	244D	2.05×10^{-3}/D
Reactor Feed Pump Failure to Run	FW	PM	X	6	1/2	193260H	2.59×10^{-6}/H
Motor Operated Valve Failure to Open/Close	ALL	MV	D, K	169	46	31652D	1.45×10^{-3}/D
Core Spray Pumps Failure to Start	CS	PM	A	4	1	290D	3.45×10^{-3}/D
Core Spray Pumps Failure to Run	CS	PM	X	4	1	544H	1.84×10^{-3}/H
Service Water Pumps Failure to Start[2]	SW	PM	A	5	1/2	84D	2.78×10^{-3}/D (IEEE)
Service Water Pumps Failure to Run	SW	PM	X	5	2	306720H	6.52×10^{-6}/H

TABLE 4.4.1-3 (Continued)
QUAD CITIES-SPECIFIC COMPONENT FAILURE RATES
Periods of Record: Jan 1985 through Dec 1991, or Jan 1993 through Dec 1995 (*)

COMPONENT TYPE GROUPING AND FAILURE MODE	SYSTEM	COMPONENT TYPE	FAILURE CODE[4]	TOTAL NUMBER OF COMPONENTS[5]	NUMBER OF FAILURES[3]	NUMBER OF DEMANDS OR HOURS	FAILURE RATE
CRD Pumps Failure to Start[2]	CR	PM	A	4	1/2	82D	4.80×10^{-3}/D (IEEE)
CRD Pumps Failure to Run	CR	PM	X	4	1/2	102416H	4.88×10^{-6}/H
RHRSW Pumps Failure to Start	RS	PM	A	8	1	1740D	5.75×10^{-4}/D *
RHRSW Pumps Failure to Run[2]	RS	PM	X	8	1	14398H	6.95×10^{-5}/H *
All Air Operated Valves Failure to Open/Close	ALL	AV	D, K	40	18	9078D	1.98×10^{-3}/D
SLC Pumps Failure to Start	SL	PM	A	4	1/2	406D	1.23×10^{-3}/D
SLC Pumps Failure to Run[2]	SL	PM	X	4	1/2	29H	1.18×10^{-5}/H (IEEE)
TBCCW Pumps Failure to Run	TB	PM	X	4	1	245376H	4.07×10^{-6}/H
RCIC Pumps Failure to Start	RC	PT	A	2	1	93D	1.08×10^{-2}/D *
RCIC Pumps Failure to Run[2]	RC	PT	X	2	1/2	25H	2.84×10^{-3}/H * (7)
Safe Shutdown Makeup Pump Failure to Start	SS	PM	A	1	1/2	34D	3.69×10^{-4}/D * (7)
Safe Shutdown Makeup Pump Failure to Run[2]	SS	PM	X	1	1/2	25H	2.99×10^{-5}/H * (7)
Core Spray Room Coolers Failure to Function	CS	RF	F	4	1	682H	1.47×10^{-3}/H
Service Water Strainer Failure to Switch	SW	FL	F	6	13	15336D	8.48×10^{-4}/D
Diesel Fire Pump Fails to Start	FP	PD	A	2	3	740D	4.05×10^{-3}/D
Diesel Fire Pump Fails to Run	FP	PD	X	2	1/2	368H	1.36×10^{-3}/H

Notes for Table 4.4.1-3:

1. Fails to start is considered common to both pumps in a Condensate/Condensate Booster Pump pair. Fails to run is on an individual pump basis. Further explanation is provided in Section 7.1 of the subject system notebook.

2. Generic Data used in accordance with "Special Conditions" of Data Collection Guide.

3. All "Number of Failures" shown as 0.5 indicate that a failure did not occur for the subject components during the review period.

4. A list of failure codes is provided in Appendix C of the Data Collection Notebook.

5. This represents the sum of components in both units unless specified otherwise.

6. Supporting Documentation is provided in Volume II of the Data Collection Notebook.

7. Bayesian update estimate using 1993 through 1995 data and NUREG/CR-4550 generic data.

TABLE 4.4.1-4
SUMMARY OF QUAD CITIES-SPECIFIC MAINTENANCE UNAVAILABILITIES
Periods of Record: Jan 1985 through Dec 1991, or Jan 1994 through Dec 1995 (*)

COMPONENT TYPE	SYSTEM	COMPONENT TYPE	FAILURE CODE[2]	NUMBER OF COMPONENTS[3]	NO. OF EVENTS (X)	TIME REQUIRED OPERATIONAL PER COMPONENT (HR)	AVERAGE OOS (HR/EVENT)	MAINT UNAVAIL
U-1 Diesel Generator	DG	DG	M	1				1.90×10^{-2} * (5)
U-2 Diesel Generator	DG	DG	M	1				1.90×10^{-2} * (5)
1/2 Diesel Generator	DG	DG	M	1				1.90×10^{-2} * (5)
Diesel Generator Output Breaker	DG	CB	M	4				8.83×10^{-4} * (6)
HPCI Turbine	HI	PT	M	2				1.80×10^{-2} * (5)
RHRSW Pump	RS	PM	M	8				1.76×10^{-3} * (6,7)
Core Spray Pump	CS	PM	M	4	6	48309	19.5	6.06×10^{-4}
Service Water Pump	SW	PM	M	5	45	61344	86.8	1.27×10^{-2}
Condensate/Condensate Booster Pump[1]	FW	PM	M	8	84	48309	77.0	1.67×10^{-2}
Reactor Feed Pump	FW	PM	M	6	126	48309	66.2	2.88×10^{-2}
TBCCW Pump	TB	PM	M	4	8	61344	51.9	1.69×10^{-3}
Control Rod Drive Pump	CR	PM	M	4	31	48309	83.5	1.34×10^{-2}
Motor Operated Valves	ALL	MV	M	169	292	48309	38.1	1.36×10^{-3}
RHR Pumps	RH	PM	M	8				3.37×10^{-3} * (6)
All Air Operated Valves	ALL	AV	M	40	29	48309	75.9	1.14×10^{-3}
Diesel Generator Cooling Water Pumps	DG	PM	M	3				5.03×10^{-3} * (6)
SLC Pumps	SL	PM	M	4	9	48309	25.0	1.16×10^{-3}

TABLE 4.4.1-4 (Continued)
SUMMARY OF QUAD CITIES-SPECIFIC MAINTENANCE UNAVAILABILITIES
Periods of Record: Jan 1985 through Dec 1991, or Jan 1993 through Dec 1995 (*)

COMPONENT TYPE	SYSTEM	COMPONENT TYPE	FAILURE CODE[2]	NUMBER OF COMPONENTS[3]	NO. OF EVENTS (X)	TIME REQUIRED OPERATIONAL PER COMPONENT (HR)	AVERAGE OOS (HR/EVENT)	MAINT UNAVAIL
RCIC Pumps	RC	PT	M	2				1.27×10^{-2} * (5)
Safe Shutdown Makeup Pump	SS	PM	M	1				4.72×10^{-3} * (6)
Service Water Strainer	SW	FL	M	6	37	61344	28.1	2.83×10^{-3}
Diesel Fire Pump	FP	PD	M	2	10	61344	370.3	3.02×10^{-2}

Notes for Table 4.4.1-4:

1. Unavailabilities are considered common to both pumps in a Condensate/Condensate Booster Pump pair. Further explanation is provided in Section 7.1 of the subject system notebook.

2. A list of failure codes is provided in Appendix C of the Data Collection Notebook.

3. This represents the sum of components in both units unless specified otherwise.

4. Supporting Documentation is provided in Volume II of the Data Collection Notebook.

5. From 1994 through 1995 INPO Safety System Performance Indicator (SSPI) data.

6. From 1994 through 1995 OOS and PIF data gathered for the March 15, 1996, submittal to the NRC. Average OOS times and required operational hours were not calculated.

7. During the preparation of final documents for the Modified IPE, it was determined that the value for RHRSW pump maintenance unavailability used in the quantification was incorrect. The value used, 1.76E-03, was a factor of 10 low and should have been 1.76E-02. A re-quantification with the correct value showed no change (to three decimal places) in the overall core damage frequency and no change in the order of the top 100 sequences. Small (second and third decimal place) changes in some sequence frequencies were noted, but this has no effect on the overall results or conclusions presented in this report. Therefore, this insignificant error will be corrected later during the Quad Cities PRA update effort.

TABLE 4.4.1-5
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
AD	AT	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Signal comparator fails to function (per hr), Rate: 8.00E-08	1.20E-09	HR	TOPS
AM	SS	CC event (2 out of 2) MGL factor: 2.20E-01, Basis event: Flow-indicating controller (amplifier) fails to function (per hr), Rate: 3.00E-06	6.60E-07	HR	IEEE
AM	ALL	CC event (2 out of 2) MGL factor: 2.20E-01, Basis event: Safety valve prematurely opens (per hr), Rate: 3.00E-06	6.60E-07	HR	IEEE
AS	PC	CC event (2 out of 4) MGL factor: 1.07E-01, Basis event: PORV fails to open (per dem), Rate: 3.57E-02	3.82E-03	D	QUAD
AS	PC	CC event (3 out of 4) MGL factor: 5.76E-02, Basis event: PORV fails to open (per dem), Rate: 3.57E-02	2.06E-03	D	QUAD
AS	PC	CC event (4 out of 4) MGL factor: 1.11E-01, Basis event: PORV fails to open (per dem), Rate: 3.57E-02	3.96E-03	D	QUAD
AV	CD	CC event (2 out of 2) MGL factor: 2.20E-01, Basis event: All air-operated valves failure to open (per dem), Rate: 1.98E-03	4.35E-04	D	QUAD
AV	CV	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: All air-operated valves failure to open (per dem), Rate: 1.98E-03	2.97E-05	D	QUAD
BS	AC	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: Bus fails all modes (per hr), Rate: 3.00E-08	2.10E-10	HR	2815
BS	AC	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: Bus fails all modes (per hr), Rate: 3.00E-08	1.42E-10	HR	2815
BS	AC	CC event (4 out of 4) MGL factor: 9.14E-03, Basis event: Bus fails all modes (per hr), Rate: 3.00E-08	2.74E-10	HR	2815
BY	FP	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Battery (wet cell) fails to provide proper output (per hr), Rate: 2.00E-06	3.00E-08	HR	2815
CB	AC	CC event (2 out of 4) MGL factor: 2.40E-02, Basis event: Circuit breaker fails to transfer/function/open/close (per dem), Rate: 3.00E-03	7.20E-05	D	IEEE
CB	AC	CC event (3 out of 4) MGL factor: 2.40E-02, Basis event: Diesel generator output breaker fails to function (per hr), Rate: 5.24E-03	1.26E-04	D	QUAD
CB	AC	CC event (3 out of 4) MGL factor: 2.38E-02, Basis event: Diesel generator output breaker fails to function (per hr), Rate: 5.24E-03	1.25E-04	D	QUAD

TABLE 4.4.1-5 (Continued)
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
CB	AC	CC event (2 out of 4) MGL factor: 2.40E-02, Basis event: 4160VAC supply breaker fails to close (per dem), Rate: 1.60E-03	3.84E-05	D	8875
CB	AC	CC event (2 out of 2) MGL factor: 3.90E-02, Basis event: 480V circuit breaker fails spurious open/fails open (per hr), Rate: 1.00E-08	3.90E-10	HR	IEEE
CB	AC	CC event (2 out of 3) MGL factor: 2.80E-02, Basis event: 480V circuit breaker fails spurious open/fails open (per hr), Rate: 1.00E-08	2.80E-10	HR	IEEE
CB	AC	CC event (2 out of 4) MGL factor: 2.40E-02, Basis event: 480V circuit breaker fails spurious open/fails open (per hr), Rate: 1.00E-08	2.40E-10	HR	IEEE
CB	AC	CC event (2 out of 4) MGL factor: 2.40E-02, Basis event: 4160VAC supply breaker spurious opening (per hr), Rate: 1.00E-06	2.40E-08	HR	4550
CB	AC	CC event (3 out of 4) MGL factor: 2.38E-02, Basis event: 480V circuit breaker fails spurious open/fails open (per hr), Rate: 1.00E-08	2.38E-10	HR	IEEE
CB	AC	CC event (3 out of 4) MGL factor: 2.38E-02, Basis event: 4160VAC supply breaker spurious opening (per hr), Rate: 1.00E-06	2.38E-08	HR	4550
CB	AC	CC event (4 out of 4) MGL factor: 6.13E-02, Basis event: 480V circuit breaker fails spurious open/fails open (per hr), Rate: 1.00E-08	6.13E-10	HR	IEEE
CO	AT	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Relay coil shorted/fails to function (per hr), Rate: 3.90E-06	5.84E-08	HR	IEEE
CV	ALL	CC event (2 out of 2) MGL factor: 3.10E-02, Basis event: CV fails to open (per dem), Rate: 8.00E-05	2.48E-06	D	IEEE
CV	DG	CC event (2 out of 3) MGL factor: 2.15E-02, Basis event: CV fails to open (per dem), Rate: 8.00E-05	1.72E-06	D	IEEE
CV	RH	CC event (2 out of 4) MGL factor: 1.60E-02, Basis event: CV fails to open (per dem), Rate: 8.00E-05	1.28E-06	D	IEEE
CV	DG	CC event (3 out of 3) MGL factor: 1.89E-02, Basis event: CV fails to open (per dem), Rate: 8.00E-05	1.51E-06	D	IEEE
CV	RH	CC event (3 out of 4) MGL factor: 1.09E-02, Basis event: CV fails to open (per dem), Rate: 8.00E-05	8.72E-07	D	IEEE
CV	RH	CC event (4 out of 4) MGL factor: 1.21E-02, Basis event: CV fails to open (per dem), Rate: 8.00E-05	9.68E-07	D	IEEE
DC	AT	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: DC power supply fails to function (per hr), Rate: 5.30E-06	7.95E-08	HR	TOPS

TABLE 4.4.1-5 (Continued)
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
DG	DG	CC event (2 out of 3) MGL factor: 5.00E-03, Basis event: U-1 diesel generator failure to start (per dem), Rate: 1.42E-02	7.10E-05	D	QUAD
DG	DG	CC event (2 out of 3) MGL factor: 5.00E-03, Basis event: 1/2 diesel generator failure to start (per dem), Rate: 1.42E-02	7.10E-05	D	QUAD
DG	DG	CC event (3 out of 3) MGL factor: 5.80E-03, Basis event: U-1 diesel generator failure to start (per dem), Rate: 1.42E-02	8.24E-05	D	QUAD
DG	DG	CC event (2 out of 3) MGL factor: 5.00E-03, Basis event: U-1 diesel generator failure to run (per hr), Rate: 1.02E-03	5.10E-06	HR	QUAD
DG	DG	CC event (2 out of 3) MGL factor: 5.00E-03, Basis event: U-2 diesel generator failure to run (per hr), Rate: 1.02E-03	5.10E-06	HR	QUAD
DG	DG	CC event (3 out of 3) MGL factor: 5.80E-03, Basis event: U-1 diesel generator failure to run (per hr), Rate: 1.02E-03	5.92E-06	HR	QUAD
DM	DG	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: Modulating damper fails to open (per dem), Rate: 3.00E-03	2.85E-05	D	2815
DM	DG	CC event (3 out of 3) MGL factor: 1.10E-02, Basis event: Modulating damper fails to open (per dem), Rate: 3.00E-03	3.30E-05	D	2815
EF	AT	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Electrical filter fails to function (per hr), Rate: 7.10E-06	1.06E-07	HR	IEEE
ES	ALL	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Level switch fails to open/close/function (per hr), Rate: 5.30E-06	7.95E-08	HR	IEEE
EV	SL	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Explosive valve does not open (per dem), Rate: 3.00E-03	4.50E-05	D	2815
FL	SW	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: Service Water strainers fail to switch (per dem), Rate: 8.48E-04	8.05E-06	D	QUAD
FL	SW	CC event (3 out of 3) MGL factor: 1.10E-02, Basis event: Service Water strainers fail to switch (per dem), Rate: 8.48E-04	9.32E-06	D	QUAD
FL	SW	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: Strainer/filter plugged (per hr), Rate: 3.00E-05	2.85E-07	HR	2815
FL	SW	CC event (3 out of 3) MGL factor: 1.10E-02, Basis event: Strainer/filter plugged (per hr), Rate: 3.00E-05	3.30E-07	HR	2815

TABLE 4.4.1-5 (Continued)
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
FN	DG	CC event (2 out of 3) MGL factor: 2.90E-02, Basis event: HVAC fan fails to start (per dem), Rate: 3.00E-04	8.70E-06	D	4550
FN	DG	CC event (3 out of 3) MGL factor: 4.93E-02, Basis event: HVAC fan fails to start (per dem), Rate: 3.00E-04	1.47E-05	D	4550
FN	FW	CC event (2 out of 2) MGL factor: 5.40E-02, Basis event: HVAC fan fails to run (per hr), Rate: 1.00E-05	5.40E-07	HR	4550
FN	DG	CC event (2 out of 3) MGL factor: 2.90E-02, Basis event: HVAC fan fails to run (per hr), Rate: 1.00E-05	2.90E-07	HR	4550
FN	DG	CC event (3 out of 3) MGL factor: 4.93E-02, Basis event: HVAC fan fails to run (per hr), Rate: 1.00E-05	4.93E-07	HR	4550
IV	AT	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Inverter fails to operate (per hr), Rate: 6.00E-05	9.00E-07	HR	2815
MS	AC	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Contactor fails during operation (per hr), Rate: 5.10E-07	7.64E-09	HR	TOPS
MV	ALL	CC event (2 out of 2) MGL factor: 1.10E-02, Basis event: Motor-operated valve failure to open (per dem), Rate: 1.45E-03	1.59E-05	D	QUAD
MV	ALL	CC event (2 out of 2) MGL factor: 1.10E-02, Basis event: Motor-operated valve failure to close (per dem), Rate: 1.45E-03	1.59E-05	D	QUAD
MV	RH	CC event (2 out of 2) MGL factor: 1.10E-02, Basis event: MOV spuriously closes (per hr), Rate: 1.00E-07	1.10E-09	HR	4550
PD	FP	CC event (2 out of 2) MGL factor: 1.00E-02, Basis event: Fire Protection diesel-driven pump fails to start (per dem), Rate: 4.05E-03	4.05E-05	D	QUAD
PD	FP	CC event (2 out of 2) MGL factor: 1.00E-02, Basis event: Fire Protection diesel-driven pump fails to run (per hr), Rate: 1.36E-03	1.36E-05	HR	QUAD
PM	CR	CC event (2 out of 2) MGL factor: 1.00E-02, Basis event: CRD pump failure to start (per dem), Rate: 4.80E-03	4.80E-05	D	IEEE
PM	CS	CC event (2 out of 2) MGL factor: 8.10E-02, Basis event: Core Spray pump failure to start (per dem), Rate: 3.45E-03	2.79E-04	D	QUAD
PM	SL	CC event (2 out of 2) MGL factor: 1.00E-02, Basis event: SLC pump failure to start (per dem), Rate: 1.23E-03	1.23E-05	D	QUAD

TABLE 4.4.1-5 (Continued)
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
PM	DG	CC event (2 out of 3) MGL factor: 6.50E-03, Basis event: Diesel generator cooling water pump failure to start (per dem), Rate: 1.33E-02	8.65E-05	D	QUAD
PM	FW	CC event (2 out of 3) MGL factor: 6.50E-03, Basis event: Reactor feed pump failure to start (per dem), Rate: 2.05E-03	1.33E-05	D	QUAD
PM	CD	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: Cond/cond booster pump failure to start (per dem), Rate: 5.77E-04	4.03E-06	D	QUAD
PM	RH	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: RHR pump failure to start (per dem), Rate: 4.43E-04	3.10E-06	D	QUAD
PM	RS	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: RHRSW pump failure to start (per dem), Rate: 5.75E-04	4.03E-06	D	QUAD
PM	SW	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: Service Water pump failure to start (per dem), Rate: 5.95E-03	4.16E-05	D	IEEE
PM	DG	CC event (3 out of 3) MGL factor: 4.55E-03, Basis event: Diesel generator cooling water pump failure to start (per dem), Rate: 1.33E-02	6.05E-05	D	QUAD
PM	FW	CC event (3 out of 3) MGL factor: 4.55E-03, Basis event: Reactor feed pump failure to start (per dem), Rate: 2.05E-03	9.32E-06	D	QUAD
PM	CD	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: Cond/cond booster pump failure to start (per dem), Rate: 5.77E-04	2.74E-06	D	QUAD
PM	RH	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: RHR pump failure to start (per dem), Rate: 4.43E-04	2.11E-06	D	QUAD
PM	RS	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: RHRSW pump failure to start (per dem), Rate: 5.75E-04	2.74E-06	D	QUAD
PM	SW	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: Service Water pump failure to start (per dem), Rate: 5.95E-03	2.83E-05	D	IEEE
PM	CD	CC event (4 out of 4) MGL factor: 9.14E-03, Basis event: Cond/cond booster pump failure to start (per dem), Rate: 5.77E-04	5.27E-06	D	QUAD
PM	RH	CC event (4 out of 4) MGL factor: 9.14E-03, Basis event: RHR pump failure to start (per dem), Rate: 4.43E-04	4.05E-06	D	QUAD
PM	RS	CC event (4 out of 4) MGL factor: 9.14E-03, Basis event: RHRSW pump failure to start (per dem), Rate: 5.75E-04	5.26E-06	D	QUAD

TABLE 4.4.1-5 (Continued)
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
PM	CR	CC event (2 out of 2) MGL factor: 1.00E-02, Basis event: CRD pump failure to run (per hr), Rate: 4.88E-06	4.88E-08	HR	QUAD
PM	CS	CC event (2 out of 2) MGL factor: 8.10E-02, Basis event: Core Spray pump failure to run (per hr), Rate: 1.84E-03	1.49E-04	HR	QUAD
PM	SL	CC event (2 out of 2) MGL factor: 1.00E-02, Basis event: SLC pump failure to run (per hr), Rate: 1.18E-05	1.18E-07	HR	IEEE
PM	DG	CC event (2 out of 3) MGL factor: 6.50E-03, Basis event: Diesel generator cooling water pump failure to run (per hr), Rate: 1.81E-03	1.18E-05	HR	QUAD
PM	FW	CC event (2 out of 3) MGL factor: 6.50E-03, Basis event: Reactor feed pump failure to run (per hr), Rate: 2.59E-06	1.68E-08	HR	QUAD
PM	CD	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: Cond/cond booster pump failure to run (per hr), Rate: 8.44E-07	5.90E-09	HR	QUAD
PM	RH	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: RHR pump failure to run (per hr), Rate: 5.77E-05	4.04E-07	HR	QUAD
PM	RS	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: Alternating motor-driven pump fails to run (per hr), Rate: 6.95E-05	4.87E-07	HR	IEEE
PM	SW	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: Service Water pump failure to run (per hr), Rate: 6.52E-06	4.56E-08	HR	QUAD
PM	DG	CC event (3 out of 3) MGL factor: 4.55E-03, Basis event: Diesel generator cooling water pump failure to run (per hr), Rate: 1.81E-03	8.24E-06	HR	QUAD
PM	FW	CC event (3 out of 3) MGL factor: 4.55E-03, Basis event: Reactor feed pump failure to run (per hr), Rate: 2.59E-06	1.17E-08	HR	QUAD
PM	CD	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: Cond/cond booster pump failure to run (per hr), Rate: 8.44E-07	4.01E-09	HR	QUAD
PM	RH	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: RHR pump failure to run (per hr), Rate: 5.77E-05	2.75E-07	HR	QUAD
PM	RS	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: Alternating motor-driven pump fails to run (per hr), Rate: 6.95E-05	3.31E-07	HR	IEEE
PM	SW	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: Service Water pump failure to run (per hr), Rate: 6.52E-06	3.10E-08	HR	QUAD

TABLE 4.4.1-5 (Continued)
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
PM	CD	CC event (4 out of 4) MGL factor: 9.14E-03, Basis event: Cond/cond booster pump failure to run (per hr), Rate: 8.44E-07	7.71E-09	HR	QUAD
PM	RH	CC event (4 out of 4) MGL factor: 9.14E-03, Basis event: RHR pump failure to run (per hr), Rate: 5.77E-05	5.27E-07	HR	QUAD
PM	RS	CC event (4 out of 4) MGL factor: 9.14E-03, Basis event: Alternating motor-driven pump fails to run (per hr), Rate: 6.95E-05	6.35E-07	HR	IEEE
PS	ALL	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Pressure switch fails to operate/open/close (per hr), Rate: 2.00E-07	3.00E-09	HR	2815
RE	ALL	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Relay fails during operation (per hr), Rate: 5.10E-07	7.64E-09	HR	TOPS
SV	AT	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: SOV fails to operate/open/close (per dem), Rate: 9.40E-07	1.41E-08	D	IEEE
SV	AT	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: SOV fails to operate/open/close (per dem), Rate: 9.40E-07	8.93E-09	D	IEEE
SV	AT	CC event (3 out of 3) MGL factor: 1.10E-02, Basis event: SOV fails to operate/open/close (per dem), Rate: 9.40E-07	1.03E-08	D	IEEE
SW	RH	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Switch (general type) fails (per hr), Rate: 1.70E-07	2.55E-09	HR	TOPS
TC	DG	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: Temperature controller fails to function (per hr), Rate: 5.78E-06	5.49E-08	HR	IEEE
TC	DG	CC event (3 out of 3) MGL factor: 1.10E-02, Basis event: Temperature controller fails to function (per hr), Rate: 5.78E-06	6.35E-08	HR	IEEE
TI	ALL	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: Timing relay fails to function (per hr), Rate: 5.10E-07	4.84E-09	HR	TOPS
TI	ALL	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: Timing relay fails to function (per hr), Rate: 5.10E-07	3.57E-09	HR	TOPS
TI	ALL	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: Timing relay fails to function (per hr), Rate: 5.10E-07	2.42E-09	HR	TOPS
TP	AT	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Pressure transmitter fails to function (per hr), Rate: 1.70E-06	2.55E-08	HR	IEEE

TABLE 4.4.1-5 (Continued)
COMMON CAUSE FAILURE DATA FOR QUAD CITIES IPE

COMP	SYSTEM	EVENT DESCRIPTION	VALUE	UNIT	SOURCE
TR	AC	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: AC transformers failure to function - UAT & RAT (per hr), Rate: 7.10E-07	6.74E-09	HR	2815
TR	AC	CC event (3 out of 3) MGL factor: 1.10E-02, Basis event: AC transformers failure to function - UAT & RAT (per hr), Rate: 7.10E-07	7.81E-09	HR	2815
TS	HI	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Temperature switch fails to open/close/function (per hr), Rate: 3.90E-07	5.85E-09	HR	IEEE
TS	RI	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Temperature switch fails to open/close/function (per hr), Rate: 3.90E-07	5.85E-09	HR	IEEE
TT	DG	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: Temperature transmitter fails to function (per hr), Rate: 2.39E-06	2.27E-08	HR	IEEE
TT	DG	CC event (3 out of 3) MGL factor: 1.10E-02, Basis event: Temperature transmitter fails to function (per hr), Rate: 2.39E-06	2.62E-08	HR	IEEE
VR	AC	CC event (2 out of 3) MGL factor: 9.50E-03, Basis event: UV protective relay fails to function (per hr), Rate: 5.10E-07	4.84E-09	HR	TOPS
VR	AC	CC event (2 out of 4) MGL factor: 7.00E-03, Basis event: UV protective relay fails to function (per hr), Rate: 5.10E-07	3.57E-09	HR	TOPS
VR	AC	CC event (3 out of 4) MGL factor: 4.76E-03, Basis event: UV protective relay fails to function (per hr), Rate: 5.10E-07	2.42E-09	HR	TOPS
XV	RH	CC event (2 out of 2) MGL factor: 1.50E-02, Basis event: Manual valve fails to open (per dem), Rate: 6.00E-05	9.00E-07	D	8875

TABLE 4.4.1-6
QUAD CITIES IPE GENERIC FAILURE DATA

COMP	SYSTEM	FAILURE MODE	FAILURE CODE	FAILURE RATE/UNITS	SOURCE
AD	ALL	Signal Comparator Loss of Function	F	2.900E-06/H	TOPS
AM	EP	Flow-Indicating Controller Fails to Function	F	3.100E-07/H	IEEE
AM	ALL	Safety Valve Prematurely Opens	U	3.000E-06/H	IEEE
AV	ALL	AOV Spuriously Closes	V	1.000E-07/H	4550
BC	ALL	Battery Charger Fails to Operate	F	6.000E-07/H	2815
BL	ALL	Air Cooler Fails to Operate	F	1.000E-06/D	2815
BS	ALL	Bus Fails All Modes	F	3.000E-08/H	2815
BY	ALL	Battery (Wet Cell) Fails to Provide Proper Output	F	2.000E-06/H	2815
CB	ALL	480VAC Circuit Breaker Fails to Open	D	9.950E-05/D	IEEE
CB	ALL	480VAC Circuit Breaker Fails to Close	K	9.950E-05/D	IEEE
CB	ALL	480VAC Circuit Breaker Spurious Open	U	1.000E-08/H	IEEE
CB	AC	480VAC Circuit Breaker Fails to Function	F	1.200E-06/H	IEEE
CB	ALL	Large AC Circuit Breaker Fails Spurious Open	U	1.000E-06/H	4550
CB	DC	Molded Case Circuit Breaker Fails Spurious Open	U	4.000E-07/H	IEEE
CB	AC	Large Circuit Breaker Fails to Open	D	1.100E-03/D	8875
CB	AC	Large Circuit Breaker Fails to Close	K	1.600E-03/D	8875
CM	IA	Air Compressor Fails to Run (Single-Stage Reciprocating)	X	1.000E-04/H	8875
CM	IA	Air Compressor Fails to Run (Rotary Type)	X	2.000E-04/H	2815
CM	IA	Air Compressor Fails to Start (Single-Stage Reciprocating)	A	5.000E-03/D	8875
CN	ALL	Contact Spurious Open	U	8.700E-08/H	TOPS
CO	ALL	Solenoid Coil Fails to Function	F	8.500E-06/H	IEEE
CV	ALL	CV Leakage	L	3.000E-06/H	2815
CV	ALL	Check Valve Fails to Open/Close	D, K	8.000E-05/D	IEEE
DC	ALL	DC Power Supply Fails to Function	F	5.800E-06/H	TOPS
DE	ALL	Diode or Rectifier Fails to Function	F	6.900E-07/H	IEEE
DM	ALL	Damper Fails to Open	D	3.000E-03/D	2815
DP	ALL	Distribution Panel Fails to Function	F	8.000E-08/H	IEEE
EF	ALL	Electrical Filter Fails to Function	F	7.100E-06/H	IEEE

TABLE 4.4.1-6 (Continued)
QUAD CITIES IPE GENERIC FAILURE DATA

COMP	SYSTEM	FAILURE MODE	FAILURE CODE	FAILURE RATE/UNITS	SOURCE
ER	ALL	Voltage Regulator Fails to Function	F	7.110E-07/H	IEEE
ES	ALL	Level Switch Fails to Function	F	5.300E-06/H	IEEE
EV	ALL	Explosive Valve Does Not Open	D	3.000E-03/D	2815
FL	ALL	Strainer/Filter Plugged	P	3.000E-05/H	2815
FN	HVAC	HVAC Fan Fails to Start	A	3.000E-04/D	4550
FN	HVAC	HVAC Fan Fails to Run	X	1.000E-05/H	4550
FR	ALL	Under-frequency Protective Relay Fails to Function	F	5.100E-07/H	TOPS
FS	ALL	Flow Switch Fails to Function	F	3.000E-07/H	IEEE
FU	ALL	Fuse Opens Prematurely	B	3.000E-06/H	2815
GE	AC	Generator Fails to Function (18.75 KVA)	F	4.340E-06/H	IEEE
HE	ALL	Heat Exchanger Blockage	P	5.700E-06/H	4550
HV	ALL	HOV Fails to Close	K	2.000E-03/D	2815
HV	ALL	HOV Fails to Operate	F	1.000E-05/H	2815
IV	ALL	Inverter Fails to Operate	F	6.000E-05/H	2815
KV	ALL	Stop Check Valve Fails to Open	D	1.000E-04/D	IREP
LS	ALL	Limit Switch Fails to Operate	F	6.000E-06/H	2815
MS	ALL	Relay Contacts Fail to Open/Close	D, K	8.500E-06/D	TOPS
MS	ALL	Contactor Fails to Function	F	5.100E-07/H	TOPS
MV	ALL	MOV Spuriously Closes	V	1.000E-07/H	4550
PA	ALL	Auxiliary Shaft Driven Pump Fails to Run	X	1.000E-04/H	RE06
PM	ALL	Motor Driven Pump Fails to Run	X	1.060E-04/H	2815
PM	ALL	Motor Driven Pump Fails to Start	A	3.900E-03/D	IEEE
PR	ALL	Overcurrent Protective Relay Spurious Opening	U	8.700E-08/H	TOPS
PS	ALL	Pressure Switch Fails to Operate	F	2.000E-07/H	2815
RD	ALL	Rupture Disc Ruptures Prematurely	R	3.900E-06/H	2815
RE	ALL	Relay Coil Opens (short circuit)	Q	1.000E-07/H	TOPS
RE	ALL	Relay Contacts Spuriously Open	U	8.700E-08/H	TOPS
RE	ALL	Relay Fails During Operation, Composite of 451, 453, 454	F	5.100E-07/H	TOPS
SD	ALL	Solid State Device Fails to Function	F	6.900E-07/H	IEEE
SV	ALL	SOV Fails to Open	D	9.400E-07/D	IEEE

TABLE 4.4.1-6 (Continued)
QUAD CITIES IPE GENERIC FAILURE DATA

COMP	SYSTEM	FAILURE MODE	FAILURE CODE	FAILURE RATE/UNITS	SOURCE
SW	ALL	Switch (General Type) Fails, Spuriously Open	F, U	1.700E-07/H	TOPS
TC	ALL	Temperature Controller (Thermostat) Fails to Function	F	5.780E-06/H	IEEE
TF	ALL	Flow Transmitter Fails to Function	F	5.900E-06/H	IEEE
TI	ALL	Timing Relay Fails to Function	F	6.700E-07/H	TOPS
TP	ALL	Pressure Transmitter Loss of Function	F	1.700E-06/H	IEEE
TR	ALL	Power Transformer Fails All Modes	F	6.000E-07/H	2815
TS	ALL	Temperature Switch Fails to Loss of Function	F	3.900E-07/H	IEEE
TT	ALL	Temperature Transmitter Fails to Function	F	2.390E-06/H	IEEE
VR	ALL	Undervoltage Protective Relay Spuriously Open	U	1.790E-06/H	IEEE
VR	ALL	Undervoltage Protective Relay Fails to Function	F	5.100E-07/H	TOPS
XV	ALL	Manual Valve Fails to Open	D	4.300E-04/D	8875
AD	ALL	Maintenance Unavailability ATWS Signal Comparator	M	4.900E-06/D	IEEE
AM	SS	Maintenance Unavailability Flow-Indicating Controller	M	1.000E-03/D	4550
BS	DC	Maintenance Unavailability DC Bus	M	9.500E-08/D	IEEE
BS	ALL	Maintenance Unavailability Bus	M	8.040E-07/D	IEEE
CB	AC	Maintenance Unavailability 480VAC Circuit Breaker	M	2.160E-06/D	IEEE
CB	AC	Maintenance Unavailability Large Circuit Breaker	M	6.720E-05/D	IEEE
CM	IA	Maintenance Unavailability Instrument Air Compressor	M	2.000E-03/D	4550
DM	ALL	Maintenance Unavailability Dampers	M	8.000E-04/D	4550
DP	SS	Maintenance Unavailability Distribution Panel	M	2.400E-06/D	4550
FN	ALL	Maintenance Unavailability Fans	M	2.000E-03/D	4550
GE	AC	Maintenance Unavailability 18.75 KVA Generator	M	7.810E-06/D	IEEE
TR	AC	Maintenance Unavailability Transformer	M	1.620E-06/D	IEEE
UP	AC	Maintenance Unavailability UPS (Static-35KVA)	M	5.000E-06/D	IEEE
XV	ALL	Maintenance Unavailability Manual Valve	M	8.000E-04/D	4550

Notes for Table 4.4.1-6:

COMP	Component type identification code (from fault tree analysis guidelines).
FAILURE CODE	Provided in Appendix C of the Data Collection Notebook.
SYSTEM	System identification code. System Acronyms are provided in Appendix B of the Data Collection Notebook.
ALL	Indicates that the entry can apply to all systems for which system- or component-specific data are not available.

SOURCE

2815	NUREG/CR-2815
4550	NUREG/CR-4550
8875	EGG-SSRE-8875
RE01	This note has been deleted.
RE02	This note has been deleted.
RE03	This note has been deleted.
RE04	This note has been deleted.
RE05	This note has been deleted.
RE06	No data for shaft driven pump FTR in NUREGs, use failure rate for motor driven pump FTR from NUREG/CR-2815.
RE07	This note has been deleted.
RE08	This note has been deleted.
TOPS	WCAP-10271 (Westinghouse proprietary class 2)
IEEE	IEEE Std. 500-1984
IREP	NUREG/CR-2728

TABLE 4.5.2-1
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
AC ELECTRIC POWER	
LOSS OF BUS 14 AFTER 13, 14-1 AVAILABLE (HAND CALC)	1.17E-01
LOSS OF BUS 14 AFTER 13, 14-1 AVAILABLE (COND CALC)	1.73E-04
LOSS OF BUS 14 AFTER 13, 14-1 AVAILABLE (LB13)	4.57E-03
LOSS OF BUS 14 AFTER 13, 14-1 AVAILABLE (LB14)	4.57E-03
LOSS OF BUS 16 AFTER 15, 13 & 14 AVAILABLE (HAND CALC)	4.69E-01
LOSS OF BUS 16 AFTER 15, 13 & 14 AVAILABLE (COND CALC)	7.25E-07
LOSS OF BUS 17 AFTER 16, 14 AVAILABLE (HAND CALC)	3.43E-01
LOSS OF BUS 17 AFTER 16, 14 AVAILABLE (COND CALC)	7.25E-07
LOSS OF BUS 17 AFTER 16, 14 UNAVAILABLE (HAND CALC)	2.13E-02
LOSS OF BUS 17 AFTER 16, 14 UNAVAILABLE (COND CALC)	1.18E-04
LOSS OF BUS 17 AFTER 15, 13 & 14 AVAILABLE (HAND CALC)	4.69E-01
LOSS OF BUS 17 AFTER 15, 13 & 14 AVAILABLE (COND CALC)	7.25E-07
LOSS OF BUS 17 AFTER 15 & 16, 13 & 14 AVAILABLE (HAND CALC)	1.00E+00
LOSS OF BUS 17 AFTER 15 & 16, 13 & 14 AVAILABLE (COND CALC)	7.38E-07
LOSS OF BUS 19 AFTER 18, 13-1 & 14-1 AVAILABLE (HAND CALC)	2.74E-01
LOSS OF BUS 19 AFTER 18, 13-1 & 14-1 AVAILABLE (COND CALC)	7.25E-07
LOSS OF BUS 19 AFTER 18, 13-1 & 14-1 AVAILABLE LB18 (HAND CALC)	4.01E-05
LOSS OF BUS 19 AFTER 18, 13-1 & 14-1 AVAILABLE LB18 (COND CALC)	4.01E-05
LOSS OF BUS 14-1 AFTER 13-1, 14 AVAILABLE (HAND CALC)	1.26E-02
LOSS OF BUS 14-1 AFTER 13-1, 14 AVAILABLE (COND CALC)	2.31E-06
LOSS OF DG1 AFTER DG1/2, 6 HRS (HAND CALC)	9.68E-02
LOSS OF DG1 AFTER DG1/2, 6 HRS (COND CALC)	8.75E-03
LOSS OF DG1 AFTER DG1/2, 24 HRS (HAND CALC)	1.54E-01
LOSS OF DG1 AFTER DG1/2, 24 HRS (COND CALC)	2.11E-02
LOSS OF DG1/2 AFTER DG1, 6 HRS (HAND CALC)	9.68E-02
LOSS OF DG1/2 AFTER DG1, 6 HRS (COND CALC)	8.75E-03
LOSS OF DG1/2 AFTER DG2, 6 HRS (HAND CALC)	1.12E-01
LOSS OF DG1/2 AFTER DG2, 6 HRS (COND CALC)	8.75E-03
LOSS OF DG1/2 AFTER DG1 AND DG2, 6 HRS (HAND CALC)	1.35E-01

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
LOSS OF DG1/2 AFTER DG1 AND DG2, 6 HRS (COND CALC)	1.18E-03
LOSS OF BUS 13 (345KV AVAILABLE)	1.02E-04
LOSS OF BUS 13 (345KV AVAILABLE) LB13	1.00E+00
LOSS OF BUS 13 (345KV UNAVAILABLE)	1.48E-03
LOSS OF BUS 13 (345KV UNAVAILABLE) LB13	1.00E+00
LOSS OF BUS 14 (345KV AVAILABLE)	2.80E-03
LOSS OF BUS 14 (345KV AVAILABLE) LB14	1.00E+00
LOSS OF BUS 14 (345KV UNAVAILABLE)	1.40E-03
LOSS OF BUS 14 (345KV UNAVAILABLE) LB14	1.00E+00
LOSS OF BUS 15, 13 & 16 & 17 AVAILABLE	1.55E-06
LOSS OF BUS 15, 13 UNAVAILABLE	2.16E-04
LOSS OF BUS 15, 16 & 17 UNAVAILABLE	1.10E-04
LOSS OF BUS 16, 14 & 15 AVAILABLE	2.12E-06
LOSS OF BUS 16, 14 UNAVAILABLE	5.49E-03
LOSS OF BUS 16, 15 UNAVAILABLE	1.10E-04
LOSS OF BUS 17, 14 & 15 AVAILABLE	2.12E-06
LOSS OF BUS 17, 14 UNAVAILABLE	5.49E-03
LOSS OF BUS 17, 15 UNAVAILABLE	1.10E-04
LOSS OF BUS 18, 13-1 & 19 AVAILABLE	2.64E-06
LOSS OF BUS 18, 13-1 & 19 AVAILABLE LB18	1.00E+00
LOSS OF BUS 18, 13-1 UNAVAILABLE	1.13E-02
LOSS OF BUS 18, 13-1 UNAVAILABLE LB18	1.00E+00
LOSS OF BUS 18, 19 UNAVAILABLE	1.10E-04
LOSS OF BUS 18, 19 UNAVAILABLE LB18	1.00E+00
LOSS OF BUS 19, 14-1 & 18 AVAILABLE	2.64E-06
LOSS OF BUS 19, 14-1 UNAVAILABLE	1.13E-02
LOSS OF BUS 19, 18 UNAVAILABLE	1.10E-04
LOSS OF BUS 13-1, 13 AVAILABLE	1.84E-04
LOSS OF BUS 13-1, 13 UNAVAILABLE	1.52E-06
LOSS OF BUS 14-1, 14 AVAILABLE	1.84E-04

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
LOSS OF BUS 14-1, 14 & DG1 UNAVAILABLE, CROSSTIE AVAILABLE	1.17E-02
LOSS OF BUS 14-1, 14 & CROSSTIE UNAVAILABLE	1.52E-06
LOSS OF POWER FROM DG1 TO BUS 14-1, 6 HRS	9.04E-02
LOSS OF POWER FROM DG1 TO BUS 14-1, 24 HRS	1.37E-01
LOSS OF POWER FROM DG1/2, 6 HRS	9.03E-02
LOSS OF POWER FROM DG1/2, 24 HRS	1.37E-01
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49), 18 UNAVAILABLE	2.79E-05
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49), 17 UNAVAILABLE	3.65E-06
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49), 17 UNAVAILABLE L182	1.48E-03
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49), 250VDC UNAVAIL	3.62E-06
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49), 250VDC UNAVAIL L182	2.77E-05
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49), 17 & 18 UNAVAILABLE	1.51E-03
LOSS OF ESS SERVICE BUS (PANEL 901-49), 18 & 250VDC UNAVAILABLE	2.03E-04
LOSS OF ESS SERVICE BUS (PANEL 901-49), 17 & 250VDC UNAVAILABLE	3.64E-06
LOSS OF ESS SERVICE BUS (PANEL 901-49), 17 & 250VDC UNAVAIL L182	1.50E-03
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49)	3.62E-06
LOSS OF ESSENTIAL SERVICE BUS (PANEL 901-49) L182	2.77E-05
LOSS OF INSTRUMENT BUS (PANEL 901-50), 18 UNAVAILABLE	9.94E-06
LOSS OF INSTRUMENT BUS (PANEL 901-50), L182	9.94E-06
LOSS OF INSTRUMENT BUS (PANEL 901-50), 15 UNAVAILABLE	2.67E-05
LOSS OF INSTRUMENT BUS (PANEL 901-50), 15 UNAVAILABLE L182	1.00E+00
LOSS OF INSTRUMENT BUS (PANEL 901-50)	3.61E-06
LOSS OF DG2 AFTER DG1, 6 HRS (HAND CALC)	9.68E-02
LOSS OF DG2 AFTER DG1, 6 HRS (COND CALC)	8.75E-03
LOSS OF BUS 24 AFTER 23, 23-1 & 24-1 AVAILABLE (HAND CALC)	1.17E-01
LOSS OF BUS 23 (345KV UNAVAILABLE)	5.57E-03
LOSS OF BUS 24 (345KV UNAVAILABLE)	5.50E-03
LOSS OF BUS 23-1, 23 UNAVAILABLE	1.52E-06

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
LOSS OF BUS 24-1, 24 & DG2 UNAVAILABLE, X-TIE AVAILABLE	1.17E-02
LOSS OF BUS 24-1, 24 & X-TIE UNAVAILABLE	1.52E-06
LOSS OF POWER FROM DG2 TO BUS 24-1, 6 HRS	7.83E-02
COMMON ACTUATION SYSTEM	
CAS FAILS TO ACTUATE RHR OR CS (ISLOCA)	7.23E-05
CAS FAILS TO ACTUATE RHR OR CS (ISLOCA), 1A UNAVAILABLE	1.19E-02
CAS FAILS TO ACTUATE RHR OR CS (ISLOCA), 1B-1 UNAVAILABLE	1.19E-02
CAS FAILS TO ACTUATE RHR OR CS (LLOCA)	7.33E-05
CAS FAILS TO ACTUATE RHR OR CS (LLOCA), 1A UNAVAILABLE	1.24E-02
CAS FAILS TO ACTUATE RHR OR CS (LLOCA), 1B-1 UNAVAILABLE	1.24E-02
CAS FAILS TO INITIATE HPCI (SLOCA)	1.46E-07
CAS FAILS TO INITIATE HPCI (SLOCA), 1A UNAVAILABLE	2.00E-05
CAS FAILS TO INITIATE HPCI (SLOCA), 1B-1 UNAVAILABLE	1.19E-04
FAILURE TO ACTUATE HPCI OR RCIC (TRANS)	1.45E-04
FAILURE TO ACTUATE HPCI OR RCIC (TRANS), 1A UNAVAILABLE	1.17E-02
FAILURE TO ACTUATE HPCI OR RCIC (TRANS), 1B-1 UNAVAILABLE	1.18E-02
DC ELECTRIC POWER	
LOSS OF 125VDC TB MAIN BUS 1A	7.34E-05
LOSS OF 125VDC TB RESERVE BUS 1B-1	8.38E-05
LOSS OF TB 250VDC MCC 1	1.10E-05
LOSS OF BUS 2A, 24HR	7.34E-05
LOSS OF BUS 2B-1, 24HR	8.38E-05
INSTRUMENT AIR SYSTEM	
IA FAILS (LOOP, DLOOP)	1.96E-02
IA FAILS	4.17E-05
IA FAILS, 15 UNAVAILABLE	1.58E-04
IA FAILS, 17 UNAVAILABLE	1.37E-02
IA FAILS, 18 UNAVAILABLE	9.60E-03

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
SERVICE WATER	
SBCS FAILS; ALL SUPPORTS AVAILABLE	5.61E-03
FAILURE OF SW (DLOOP), 13 UNAVAILABLE	2.85E-02
FAILURE OF SW (DLOOP), 13 & 16 UNAVAILABLE	8.35E-02
FAILURE OF SW (DLOOP), 13 & 14 & 16 UNAVAILABLE	8.37E-02
FAILURE OF SW (DLOOP), 13 & 23 UNAVAILABLE	2.87E-02
FAILURE OF SW (DLOOP), 13 & 23 & 16 UNAVAILABLE	8.37E-02
FAILURE OF SW (DLOOP), 13 & 24 UNAVAILABLE	2.87E-02
FAILURE OF SW (DLOOP), 13 & 24 & 16 UNAVAILABLE	8.37E-02
FAILURE OF SW (DLOOP), 13 & 14 & 23 & 16 UNAVAILABLE	1.11E-01
FAILURE OF SW (DLOOP), 13 & 23 & 24 UNAVAILABLE	5.74E-02
FAILURE OF SW (DLOOP), 13 & 23 & 24 & 16 UNAVAILABLE	1.11E-01
FAILURE OF SW (DLOOP), 23 UNAVAILABLE	2.85E-02
FAILURE OF SW (DLOOP), 23 & 16 UNAVAILABLE	8.35E-02
FAILURE OF SW (DLOOP), 14 & 23 UNAVAILABLE	2.87E-02
FAILURE OF SW (DLOOP), 14 & 23 & 16 UNAVAILABLE	8.37E-02
FAILURE OF SW (DLOOP), 23 & 24 UNAVAILABLE	2.87E-02
FAILURE OF SW (DLOOP), 23 & 24 & 16 UNAVAILABLE	8.37E-02
FAILURE OF SW (LOOP)	4.59E-02
FAILURE OF SW (LOOP), 16 UNAVAILABLE	1.00E-01
FAILURE OF SW (TRANS), 1A & 14 & SWITCHING UNAVAILABLE	9.27E-02
FAILURE OF SW (TRANS), 1A & 14 & SWITCHING UNAVAILABLE LOIA	3.72E-02
FAILURE OF SW (TRANS)	1.53E-05
FAILURE OF SW (TRANS) LOIA	8.14E-06
FAILURE OF SW (TRANS), 13 OR 1A UNAVAILABLE	4.15E-04
FAILURE OF SW (TRANS), 13 OR 1A UNAVAILABLE LOIA	4.07E-04
FAILURE OF SW (TRANS), 14 OR 1B-1 UNAVAILABLE	4.15E-04
FAILURE OF SW (TRANS), 14 OR 1B-1 UNAVAILABLE LOIA	4.07E-04
FAILURE OF SW (TRANS), 16 UNAVAILABLE	5.77E-02
FAILURE OF SW (TRANS), 16 UNAVAILABLE LOIA	3.02E-05

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
FAILURE OF SW (TRANS), 13 OR 1A/16 UNAVAILABLE	5.81E-02
FAILURE OF SW (TRANS), 13 OR 1A/16 UNAVAILABLE LOIA	4.30E-04
FAILURE OF SW (TRANS), 14 OR 1B-1/16 UNAVAILABLE	5.81E-02
FAILURE OF SW (TRANS), 14 OR 1B-1/16 UNAVAILABLE LOIA	4.30E-04
FAILURE OF SW (TRANS), 13/14 OR 1B-1/16 UNAVAILABLE	9.94E-02
FAILURE OF SW (TRANS), 13/14 OR 1B-1/16 UNAVAILABLE LOIA	4.42E-02
FAILURE OF SW (TRANS), 1A UNAVAILABLE	1.03E-05
FAILURE OF SW (TRANS), 1A UNAVAILABLE LOIA	3.19E-06
FAILURE OF SW (TRANS), 1A & 13 UNAVAILABLE	3.67E-02
FAILURE OF SW (TRANS), 1A & 13 UNAVAILABLE LOIA	3.67E-02
FAILURE OF SW (TRANS), 1A & SWITCHING UNAVAILABLE	5.77E-02
FAILURE OF SW (TRANS), 1A & SWITCHING UNAVAILABLE LOIA	2.53E-05
FAILURE OF SW (TRANS), 1A & 13 & SWITCHING UNAVAILABLE	9.23E-02
FAILURE OF SW (TRANS), 1A & 13 & SWITCHING UNAVAILABLE LOIA	3.67E-02
FAILURE OF SW (TRANS), 15 UNAVAILABLE	2.20E-05
FAILURE OF SW (TRANS), 15 UNAVAILABLE LOIA	8.14E-06
FAILURE OF SW (TRANS), 13 OR 1A/15 UNAVAILABLE	4.21E-04
FAILURE OF SW (TRANS), 13 OR 1A/15 UNAVAILABLE LOIA	4.07E-04
FAILURE OF SW (TRANS), 1A & 15 UNAVAILABLE	1.70E-05
FAILURE OF SW (TRANS), 1A & 15 UNAVAILABLE LOIA	3.20E-06
FAILURE OF SW (TRANS), 17 UNAVAILABLE	8.04E-04
FAILURE OF SW (TRANS), 17 UNAVAILABLE LOIA	8.44E-06
FAILURE OF SW (TRANS), 13 OR 1A/17 UNAVAILABLE	1.20E-03
FAILURE OF SW (TRANS), 13 OR 1A/17 UNAVAILABLE LOIA	4.08E-04
FAILURE OF SW (TRANS), 1A & 17 UNAVAILABLE	7.99E-04
FAILURE OF SW (TRANS), 1A & 17 UNAVAILABLE LOIA	3.50E-06
FAILURE OF SW (TRANS), 18 UNAVAILABLE	5.67E-04
FAILURE OF SW (TRANS), 18 UNAVAILABLE LOIA	8.35E-06
FAILURE OF SW (TRANS), 13 OR 1A/18 UNAVAILABLE	9.66E-04
FAILURE OF SW (TRANS), 13 OR 1A/18 UNAVAILABLE LOIA	4.08E-04

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
FAILURE OF SW (TRANS), 1A & 18 UNAVAILABLE	5.62E-04
FAILURE OF SW (TRANS), 1A & 18 UNAVAILABLE LOIA	3.41E-06
TURBINE BUILDING CLOSED COOLING WATER	
TBCCW FAILS	3.12E-06
TBCCW FAILS, 17 UNAVAILABLE	1.01E-04
TBCCW FAILS, 15 UNAVAILABLE	7.24E-03
ATWS SYSTEM	
ARI FAILS; ALL SUPPORTS AVAILABLE	5.67E-08
ARI FAILS; 1R1 AVAILABLE	9.57E-05
ARI FAILS; 1M1 AVAILABLE	9.83E-05
ATWS2 ACTUATION FAILS AFTER ATWS1 FAILURE	1.30E-02
ATWS1 ACTUATION FAILS	1.28E-02
ATWS1 ACTUATION FAILS (COND CALC)	1.66E-04
ATWS2 ACTUATION FAILS (GIVEN ATWS1 SUCCESS)	1.28E-02
AUTO RPT FAILS; ALL SUPPORTS AVAILABLE	1.99E-04
AUTO RPT FAILS; 1R1 AVAILABLE	2.38E-04
AUTO RPT FAILS; 1M1 AVAILABLE	2.40E-04
MANUAL RPT FAILS	1.99E-04
FAILURE OF RPS	3.00E-05

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
AUTOMATIC DEPRESSURIZATION SYSTEM	
ADS FAILS; ALL SUPPORTS AVAILABLE	9.14E-06
ADS FAILS; 1R1 AVAILABLE	9.20E-06
ADS FAILS; 1M1 AVAILABLE	8.20E-05
CCST INVENTORY CONTROL	
CCST INVENTORY NOT MAINTAINED; NON-LOOP	5.97E-06
CCST INVENTORY NOT MAINTAINED; LOOP OR DLOOP	4.03E-04
CONTROL ROD DRIVE SYSTEM	
CRD FAILS; ALL SUPPORTS AVAILABLE	2.13E-03
CRD FAILS; 14, 1R1, 17, TBCCW AVAILABLE	2.85E-02
CRD FAILS; BUS 13, TBCCW AVAILABLE	2.24E-03
CRD FAILS (LOOP, DLOOP); ALL SUPPORTS AVAILABLE	2.28E-03
CRD FAILS (LOOP, DLOOP); 14, 17, 1R1, TBCCW AVAILABLE	2.46E-02
CRD FAILS (LOOP, DLOOP); 13, 1M1, TBCCW AVAILABLE	7.03E-03
CORE SPRAY SYSTEM	
CS FAILS; ALL SUPPORT SYSTEMS AVAILABLE	6.66E-03
CS FAILS; 13-1, 18, 1M1 AVAILABLE	5.24E-02
CS FAILS; 14-1, 19, 1R1 AVAILABLE	5.24E-02
FEEDWATER AND CONDENSATE	
FW FAILS; ALL SUPPORTS AVAILABLE	2.05E-03
FW FAILS; ALL SUPPORTS AVAILABLE LB11	4.29E-03
FW FAILS; ALL SUPPORTS AVAILABLE LB12	5.14E-03
FW FAILS; 1M1, 1R1, 14, 15, 18, 19 AVAILABLE	2.21E-03
FW FAILS; 1M1, 1R1, 14, 15, 18, 19 AVAILABLE LB11	4.45E-03
FW FAILS; 1M1, 1R1, 14, 15, 18, 19 AVAILABLE LB12	5.30E-03
FW FAILS; 1M1, 1R1, 13, 15, 18, 19 AVAILABLE	2.37E-03
FW FAILS; 1M1, 1R1, 13, 15, 18, 19 AVAILABLE LB11	4.61E-03
FW FAILS; 1M1, 1R1, 13, 15, 18, 19 AVAILABLE LB12	5.46E-03
FW FAILS; 1M1, 1R1, 13, 15, 18, 19 AVAILABLE LB13	1.00E+00
FW FAILS; 1R1, 13, 14, 15, 19 AVAILABLE	2.69E-03

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
FW FAILS; 1R1, 13, 14, 15, 19 AVAILABLE LB11	4.81E-03
FW FAILS; 1R1, 13, 14, 15, 19 AVAILABLE LB12	1.00E+00
FW FAILS; 1M1, 13, 14, 15, 18, 19 AVAILABLE	4.99E-03
FW FAILS; 1M1, 13, 14, 15, 18, 19 AVAILABLE LB11	1.00E+00
FW FAILS; 1M1, 13, 14, 15, 18, 19 AVAILABLE LB12	5.14E-03
FW FAILS; 1M1, 1R1, 13, 14, 15, 19 AVAILABLE	2.58E-03
FW FAILS; 1M1, 1R1, 13, 14, 15, 19 AVAILABLE LB11	4.82E-03
FW FAILS; 1M1, 1R1, 13, 14, 15, 19 AVAILABLE LB12	5.67E-03
FW FAILS; 1R1, 14, 15, 19 AVAILABLE	2.85E-03
FW FAILS; 1R1, 14, 15, 19 AVAILABLE LB11	4.97E-03
FW FAILS; 1R1, 14, 15, 19 AVAILABLE LB12	1.00E+00
FW FAILS; 1M1, 14, 15, 18, 19 AVAILABLE	5.14E-03
FW FAILS; 1M1, 14, 15, 18, 19 AVAILABLE LB11	1.00E+00
FW FAILS; 1M1, 14, 15, 18, 19 AVAILABLE LB12	5.29E-03
FW FAILS; 1M1, 1R1, 14, 15, 19 AVAILABLE	2.74E-03
FW FAILS; 1M1, 1R1, 14, 15, 19 AVAILABLE LB11	4.98E-03
FW FAILS; 1M1, 1R1, 14, 15, 19 AVAILABLE LB12	5.83E-03
FW FAILS; 1R1, 13, 15, 19 AVAILABLE	3.05E-03
FW FAILS; 1R1, 13, 15, 19 AVAILABLE LB11	5.17E-03
FW FAILS; 1R1, 13, 15, 19 AVAILABLE LB12	1.00E+00
FW FAILS; 1M1, 13, 15, 18, 19 AVAILABLE	5.30E-03
FW FAILS; 1M1, 13, 15, 18, 19 AVAILABLE LB11	1.00E+00
FW FAILS; 1M1, 13, 15, 18, 19 AVAILABLE LB12	5.45E-03
FW FAILS; 1M1, 1R1, 13, 15, 19 AVAILABLE	2.90E-03
FW FAILS; 1M1, 1R1, 13, 15, 19 AVAILABLE LB11	5.14E-03
FW FAILS; 1M1, 1R1, 13, 15, 19 AVAILABLE LB12	5.99E-03
FW FAILS; 1M1, 13, 14, 15, 19 AVAILABLE	5.52E-03
FW FAILS; 1M1, 13, 14, 15, 19 AVAILABLE LB11	1.00E+00
FW FAILS; 1M1, 13, 14, 15, 19 AVAILABLE LB12	5.67E-03
FW FAILS; 1M1, 13, 15, 19 AVAILABLE	5.83E-03

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
FW FAILS; 1M1, 13, 15, 19 AVAILABLE LB11	1.00E+00
FW FAILS; 1M1, 13, 15, 19 AVAILABLE LB12	5.97E-03
FW FAILS; 1M1, 14, 15, 19 AVAILABLE	5.67E-03
FW FAILS; 1M1, 14, 15, 19 AVAILABLE LB11	1.00E+00
FW FAILS; 1M1, 14, 15, 19 AVAILABLE LB12	5.82E-03
FW FAILS (LOCA); ALL SUPPORTS AVAILABLE	2.00E-04
FW FAILS (LOCA); 1R1, 14 AVAILABLE	1.28E-03
FW FAILS (LOCA); 1M1, 13 AVAILABLE	4.45E-04
FIRE PROTECTION SYSTEM	
FP FAILS; ALL SUPPORTS AVAILABLE	1.52E-02
HIGH PRESSURE COOLANT INJECTION	
HP FAILS; ALL SUPPORTS AVAILABLE	8.32E-02
HP FAILS; 1M1, 1T2 AVAILABLE	8.33E-02
HP FAILS; 1R1, 1T2 AVAILABLE	8.32E-02
HP FAILS AFTER CA FAILURE; ALL SUPPORTS AVAILABLE	8.43E-02
HP FAILS AFTER CA FAILURE; 1M1, 1T2 AVAILABLE	8.44E-02
HP FAILS AFTER CA FAILURE; 1R1, 1T2 AVAILABLE	8.43E-02
HP FAILS (MULTIPLE START); ALL SUPPORTS AVAILABLE	2.30E-01
HP FAILS (MULTIPLE START); 1M1, 1T2 AVAILABLE	2.30E-01
HP FAILS (MULTIPLE START); 1R1, 1T2 AVAILABLE	2.30E-01
REACTOR CORE ISOLATION COOLING SYSTEM	
RCIC FAILS; ALL SUPPORTS AVAILABLE	1.01E-01
RCIC FAILS; ALL SUPPORTS AVAILABLE, CAS FAILURE	1.03E-01
RESIDUAL HEAT REMOVAL SYSTEM	
CCST SUCTION VALVES FAIL, RHR PUMP A SUCCESS	6.14E-03
CCST SUCTION VALVES FAIL, RHR PUMP B SUCCESS	6.14E-03
CCST SUCTION VALVES FAIL, CS SUCCESS	5.75E-03
CNTS FAILS; ALL SUPPORTS AVAILABLE	1.35E-05
CNTS FAILS; BUS 18, 1M1, 1R1 AVAILABLE	1.54E-04
CNTS FAILS; BUS 18, 1M1 AVAILABLE	3.97E-04

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
CNTS FAILS; BUS 19, 1R1 AVAILABLE	5.55E-04
CNTS FAILS; BUS 19, 1M1, 1R1 AVAILABLE	8.10E-05
RHR HX FAILS/RHR A/B PUMPS AVAILABLE; ALL SUPPORTS AVAILABLE	3.39E-04
RHR HX FAILS/RHR A/B PUMPS AVAILABLE; 14, 19, 1R1 AVAILABLE	9.09E-03
RHR HX FAILS/RHR A/B PUMPS AVAILABLE; 13, 18, 1M1 AVAILABLE	9.16E-03
RHR HX FAILS/RHR A PUMP AVAILABLE; 13, 18, 1M1 AVAILABLE	9.16E-03
RHR HX FAILS/RHR A PUMP AVAILABLE; 13, 18, 1M1 AVAILABLE L182	1.00E+00
RHR HX FAILS/RHR A PUMP AVAILABLE; 13, 18, 1M1 AVAILABLE LB13	1.00E+00
RHR HX FAILS/RHR A/B PUMPS AVAILABLE; ALL SUPPORTS AVAILABLE L182	9.30E-03
RHR HX FAILS/RHR A/B PUMPS AVAILABLE; 13, 18, 1M1 AVAILABLE L182	1.00E+00
RHR HX FAILS/RHR B PUMP AVAILABLE; 14, 19, 1R1 AVAILABLE	9.09E-03
RHR HX FAILS/RHR B PUMP AVAILABLE; 14, 19, 1R1 AVAILABLE LB14	1.00E+00
RHR B FAILS/A FAILURE; ALL SUPPORTS AVAILABLE	1.02E-02
RHR A FAILS ; ALL SUPPORTS AVAILABLE	3.10E-03
RHR A FAILS/RHR B FAILS (COND CALC)	3.17E-05
RHR A FAILS; 13-1, 1M1, 18 AVAILABLE	3.22E-03
RHR B FAILS/A SUCCESS; ALL SUPPORTS AVAILABLE	2.97E-03
RHR B FAILS; 14-1, 1R1, 19 AVAILABLE	3.10E-03
INJ VALVES FAIL; ALL SUPPORTS AVAILABLE	2.89E-03
INJ VALVES FAIL; 18, 19, 1M1 AVAILABLE	5.01E-01
SDC FAILS; RHR A PUMP AVAILABLE	2.46E-02
SDC FAILS; RHR B PUMP AVAILABLE	1.36E-02
SPC VALVE FAIL; ALL SUPPORTS AVAILABLE	5.69E-05
SPC VALVE FAIL; 19, 1R1 AVAILABLE	5.63E-03
SPC VALVE FAIL; 18, 1M1 AVAILABLE	5.70E-03
SAFE SHUTDOWN MAKEUP PUMP SYSTEM	
SSMP\CST FAILS; ALL SUPPORTS AVAILABLE	1.60E-02
SSMP\CST FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	1.88E-02
SSMP\CST FAILS; 14-1, 1M1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	1.60E-02
SSMP\CST FAILS; 14-1, 1R1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	1.60E-02

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
SSMP\CST FAILS; 1M1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	1.88E-02
SSMP\CST FAILS; 1R1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	1.88E-02
SSMP\CST FAILS (DLOOP); 1R1, 2R1, 2ES, 14-1, 1T2 AVAILABLE	1.60E-02
SSMP\CST FAILS (DLOOP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE	1.71E-02
SSMP\CST FAILS (DLOOP); 1R1, 2R1, 14-1, 1T2 AVAILABLE	1.72E-02
SSMP\CST FAILS (DLOOP); 1R1, 2R1, 24-1, 1T2 AVAILABLE	1.99E-02
SSMP\CST FAILS (DLOOP); 1R1, 14-1, 2ES, 1T2 AVAILABLE	1.60E-02
SSMP\CST FAILS (DLOOP); 1R1, 14-1, 1T2 AVAILABLE	1.72E-02
SSMP\CST FAILS (DLOOP); 2R1, 24-1, 1T2 AVAILABLE	2.00E-02
SSMP\FP FAILS; ALL SUPPORTS AVAILABLE	3.17E-02
SSMP\FP FAILS; 1R1, 1M1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	3.44E-02
SSMP\FP FAILS; 14-1, 1M1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	3.17E-02
SSMP\FP FAILS; 14-1, 1R1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	3.17E-02
SSMP\FP FAILS; 1M1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	3.45E-02
SSMP\FP FAILS; 1R1, 1T2 AVAILABLE (DLOOP: 241, 2ES)	3.44E-02
SSMP\FP FAILS (DLOOP); 1R1, 2R1, 2ES, 14-1, 1T2 AVAILABLE	3.17E-02
SSMP\FP FAILS (DLOOP); 1R1, 2R1, 24-1, 14-1, 1T2 AVAILABLE	3.28E-02
SSMP\FP FAILS (DLOOP); 1R1, 2R1, 14-1, 1T2 AVAILABLE	3.29E-02
SSMP\FP FAILS (DLOOP); 1R1, 2R1, 24-1, 1T2 AVAILABLE	3.56E-02
SSMP\FP FAILS (DLOOP); 1R1, 14-1, 2ES, 1T2 AVAILABLE	3.17E-02
SSMP\FP FAILS (DLOOP); 1R1, 14-1, 1T2 AVAILABLE	3.29E-02
SSMP\FP FAILS (DLOOP); 2R1, 24-1, 1T2 AVAILABLE	3.56E-02
SAFETY/RELIEF VALVES	
RVC FAILS; RELIEF VALVES CLOSE (TRANS)	2.70E-02
RVC FAILS; RELIEF VALVES CLOSE (ATWS W/MC)	1.60E-01
RVC FAILS; RELIEF VALVES CLOSE (ATWS W/O MC)	3.60E-01
RVC FAILS; RELIEF VALVES OPEN (TRANS)	1.00E-11
RVC FAILS; RELIEF VALVES OPEN (ATWS W/MC)	1.40E-11
RVC FAILS; RELIEF VALVES OPEN (ATWS W/O MC)	2.00E-07

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
STANDBY LIQUID CONTROL	
SLC FAILS; 1/2 PUMPS; ALL SUPPORTS AVAILABLE	2.35E-04
SLC FAILS; 1/2 PUMPS; BUS 19 AVAILABLE	5.65E-03
SLC FAILS; 1/2 PUMPS; BUS 18 AVAILABLE	5.65E-03
SLC FAILS; 2/2 PUMPS	1.12E-02
TORUS/DRYWELL VENT SYSTEM	
LARGE DRYWELL AND TORUS VENTS FAIL	6.73E-01
LARGE DRYWELL AND TORUS VENTS FAIL L182	7.35E-01
LARGE TORUS VENT FAILS; ALL SUPPORTS AVAILABLE	5.93E-03
LARGE TORUS VENT FAILS; L182	7.32E-03
LARGE VENTS FAIL; IA, ES, 18, 19 AVAILABLE	6.94E-01
LARGE TORUS VENT FAILS; IA, ES, 18, 19 AVAILABLE	6.33E-03
LARGE TORUS VENT FAILS; IA, ES, 18, 19 AVAILABLE L182	1.00E+00
LARGE VENTS FAIL; IA, ES, 15, 19 AVAILABLE	7.42E-01
LARGE TORUS VENT FAILS; IA, ES, 15, 19 AVAILABLE	7.51E-03
LARGE VENTS FAIL; IA, ES, 15, 18 AVAILABLE	7.42E-01
LARGE TORUS VENT FAILS; IA, ES, 15, 18 AVAILABLE	7.51E-03
LARGE TORUS VENT FAILS; IA, ES, 15, 18 AVAILABLE L182	1.00E+00
LARGE VENTS FAIL; ALL SUPPORT AVAILABLE (COND CALC)	3.99E-03
LARGE VENTS FAIL; 15 UNAVAILABLE (COND CALC)	4.39E-03
LARGE VENTS FAIL; 18 UNAVAILABLE (COND CALC)	5.58E-03
LARGE VENTS FAIL; 19 UNAVAILABLE (COND CALC)	5.58E-03
LARGE VENTS FAIL; ALL SUPPORT AVAILABLE (COND CALC) L182	5.38E-03
LARGE VENTS FAIL; 15 UNAVAILABLE (COND CALC) L182	1.00E+00
LARGE VENTS FAIL; 19 UNAVAILABLE (COND CALC) L182	1.00E+00
LARGE VENTS FAIL; IA, ES, 15, 18 AVAILABLE L182	1.00E+00
LARGE VENTS FAIL; IA, ES, 18, 19 AVAILABLE L182	1.00E+00
OPERATOR ACTIONS	
OPERATOR ALLOWS ADS (0)	0.00
OPERATOR FAILS TO INITIATE ADS (1)	1.30E-02

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
OPERATOR FAILS TO INITIATE ADS (2)	1.30E-03
OPERATOR FAILS TO INITIATE ADS (4)	5.10E-02
OPERATOR FAILS TO INITIATE ADS (9)	1.60E-02
OPERATOR FAILS TO INITIATE ADS (10)	1.60E-03
OPERATOR FAILS TO INITIATE ADS (11)	6.50E-02
OPERATOR FAILS TO INITIATE ADS (12)	5.20E-02
OPERATOR FAILS TO INITIATE ADS (17)	1.60E-02
OPERATOR FAILS TO INITIATE ADS (18)	1.60E-03
OPERATOR FAILS TO INITIATE ADS (19)	6.50E-02
OPERATOR FAILS TO INITIATE ADS (20)	5.20E-02
OPERATOR FAILS TO INITIATE ADS (21)	1.60E-01
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (2)	1.30E-03
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (4)	5.10E-02
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (9)	1.30E-02
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (10)	1.30E-03
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (11)	6.20E-02
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (12)	5.10E-02
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (17)	1.30E-02
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (18)	1.30E-03
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (18)	1.50E-02
OPERATOR FAILS TO INITIATE DEPRESSURIZATION (19)	6.20E-02
OPERATOR FAILS TO CONTROL RV LEVEL AFTER ATWS (9)	9.40E-02
OPERATOR FAILS TO CONTROL RV LEVEL AFTER ATWS (11)	1.40E-01
OPERATOR FAILS TO INITIATE ARI SYSTEM (17)	6.00E-03
OPERATOR FAILS TO INITIATE ARI SYSTEM (21)	1.50E-01
OPERATOR FAILS TO DEPRESSURIZE TO MAIN CONDENSER (2)	5.50E-04
OPERATOR FAILS TO DEPRESSURIZE TO MAIN CONDENSER (4)	6.00E-03
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (1)	1.20E-02
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (3)	6.10E-02
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (9)	2.50E-02

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (10)	2.70E-03
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (11)	7.40E-02
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (13)	1.60E-01
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (17)	6.20E-02
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (18)	6.80E-03
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (20)	1.20E-02
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (21)	2.00E-01
OPERATOR FAILS TO INITIATE CONTAINMENT SPRAY (22)	2.20E-02
OPERATOR FAILS TO RESTORE CRD (1)	6.20E-03
OPERATOR FAILS TO RESTORE CRD (2)	6.40E-04
OPERATOR FAILS TO RESTORE CRD (4)	6.10E-03
OPERATOR FAILS TO RESTORE CRD (10)	1.30E-03
OPERATOR FAILS TO RESTORE CRD (12)	6.80E-03
OPERATOR FAILS TO RESTORE CRD (17)	3.10E-02
OPERATOR FAILS TO RESTORE CRD (18)	3.40E-03
OPERATOR FAILS TO ALIGN TO CCST SOURCE (2)	1.40E-03
OPERATOR FAILS TO ALIGN TO CCST SOURCE (10)	1.50E-03
OPERATOR FAILS TO ALIGN TO CCST SOURCE (12)	5.10E-02
OPERATOR FAILS TO ALIGN TO CCST SOURCE (18)	1.90E-03
OPERATOR FAILS TO ALIGN TO CCST SOURCE (20)	5.20E-02
OPERATOR FAILS TO ALIGN TO CCST SOURCE (25)	1.00E+00
OPERATOR FAILS TO RESTART A FW PUMP (0)	8.40E-03
OPERATOR FAILS TO RESTART A FW PUMP (1)	1.40E-02
OPERATOR FAILS TO RESTART A FW PUMP (2)	1.40E-03
OPERATOR FAILS TO MANUALLY SCRAM AND CONTROL FW (9)	1.70E-02
OPERATOR FAILS TO RESTART (F/V) FW FOLLOWING LOOP (1)	1.00E+00
OPERATOR FAILS TO RESTART (F/V) FW FOLLOWING LOOP (17)	1.00E+00
OPERATOR FAILS TO ALIGN COOLING TO RHR (1)	1.00E-04
OPERATOR FAILS TO ALIGN COOLING TO RHR (2)	1.00E-05
OPERATOR FAILS TO ALIGN COOLING TO RHR (6)	1.60E-02

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
OPERATOR FAILS TO ALIGN COOLING TO RHR (9)	1.70E-02
OPERATOR FAILS TO ALIGN COOLING TO RHR (10)	1.90E-03
OPERATOR FAILS TO ALIGN COOLING TO RHR (14)	1.70E-02
OPERATOR FAILS TO ALIGN COOLING TO RHR (18)	4.90E-03
OPERATOR FAILS TO ALIGN COOLING TO RHR (22)	2.00E-02
OPERATOR FAILS TO INHIBIT ADS (1)	1.30E-02
OPERATOR FAILS TO INHIBIT ADS (2)	3.30E-03
OPERATOR FAILS TO INHIBIT ADS (9)	1.30E-02
OPERATOR FAILS TO INHIBIT ADS (26)	3.00E-03
OPERATOR FAILS TO ISOLATE THE BREAK (1)	1.00E+00
OPERATOR FAILS TO ISOLATE THE BREAK (9)	1.00E+00
OPERATOR FAILS TO ISOLATE THE BREAK (10)	1.00E+00
OPERATOR FAILS TO ISOLATE THE BREAK (13)	1.00E+00
OPERATOR FAILS TO ISOLATE THE BREAK (14)	1.00E+00
OPERATOR FAILS TO ISOLATE THE BREAK (17)	1.00E+00
OPERATOR FAILS TO INITIATE RPT (17)	5.10E-02
OPERATOR FAILS TO INITIATE SBCS (2)	1.50E-03
OPERATOR FAILS TO INITIATE SBCS (9)	2.80E-02
OPERATOR FAILS TO INITIATE SBCS (10)	3.10E-03
OPERATOR FAILS TO INITIATE SBCS (12)	8.40E-03
OPERATOR FAILS TO INITIATE SBCS (17)	7.20E-02
OPERATOR FAILS TO INITIATE SBCS (19)	1.20E-01
OPERATOR FAILS TO ALIGN FOR SHUTDOWN COOLING (2)	9.20E-03
OPERATOR FAILS TO INITIATE SLC (1/2 PUMP REQUIRED) (1)	8.70E-03
OPERATOR FAILS TO INITIATE SLC (1/2 PUMP REQUIRED) (17)	4.50E-02
OPERATOR FAILS TO INITIATE SLC (2/2 PUMPS REQUIRED) (16)	5.60E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (1)	1.40E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (2)	1.40E-03
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (3)	6.30E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (4)	5.10E-02

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (9)	1.50E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (10)	1.50E-03
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (11)	6.50E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (12)	5.10E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (17)	1.90E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (18)	1.90E-03
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (19)	6.80E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (NO ECCS CONDITION) (20)	5.20E-02
OPERATOR FAILS TO INIT SSMP FROM FP (NO ECCS CONDITION) (10)	2.00E-03
OPERATOR FAILS TO INIT SSMP FROM FP (NO ECCS CONDITION) (12)	7.40E-03
OPERATOR FAILS TO INIT SSMP FROM FP (NO ECCS CONDITION) (17)	6.30E-02
OPERATOR FAILS TO INIT SSMP FROM FP (NO ECCS CONDITION) (18)	6.90E-03
OPERATOR FAILS TO INIT SSMP FROM FP (NO ECCS CONDITION) (20)	1.20E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (3)	5.60E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (9)	1.40E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (10)	1.50E-03
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (11)	6.30E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (12)	7.00E-03
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (13)	1.50E-01
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (17)	4.80E-03
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (18)	5.30E-03
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (19)	9.60E-02
OPERATOR FAILS TO INIT SSMP FROM CCST (ECCS CONDITION) (20)	1.10E-02
OPERATOR FAILS TO INITIATE SPC (1)	1.00E-04
OPERATOR FAILS TO INITIATE SPC (2)	1.00E-05
OPERATOR FAILS TO INITIATE SPC (4)	5.50E-03
OPERATOR FAILS TO INITIATE SPC (9)	1.30E-03
OPERATOR FAILS TO INITIATE SPC (10)	1.40E-04
OPERATOR FAILS TO INITIATE SPC (17)	6.30E-03
OPERATOR FAILS TO INITIATE SPC (18)	6.90E-04

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
OPERATOR FAILS TO RECOVER SW/TBCCW (1)	4.00E-02
OPERATOR FAILS TO RECOVER SW/TBCCW (2)	4.40E-03
OPERATOR FAILS TO RECOVER SW/TBCCW (4)	9.70E-03
OPERATOR FAILS TO RECOVER SW/TBCCW (9)	8.00E-02
OPERATOR FAILS TO RECOVER SW/TBCCW (10)	8.80E-03
OPERATOR FAILS TO RECOVER SW/TBCCW (11)	1.30E-01
OPERATOR FAILS TO RECOVER SW/TBCCW (12)	1.40E-02
OPERATOR FAILS TO RECOVER SW/TBCCW (17)	2.00E-01
OPERATOR FAILS TO RECOVER SW/TBCCW (18)	2.20E-02
OPERATOR FAILS TO RECOVER SW/TBCCW (19)	2.40E-01
OPERATOR FAILS TO RECOVER SW/TBCCW (20)	2.60E-02
OPERATOR FAILS TO VENT CONTAINMENT (2)	1.00E-02
OPERATOR FAILS TO VENT CONTAINMENT (10)	2.00E-02
OPERATOR FAILS TO VENT CONTAINMENT (18)	5.10E-02
OPERATOR FAILS TO REALIGN DG1/2 TO APPROPRIATE UNIT	1.00E+00
OPERATOR ACTION NOT NEEDED; HW FAILED BY IE	0.00
RECOVERY OF OFFSITE POWER	
FAILURE TO RECOVER OSP	5.09E-02
FAILURE TO RECOVER OSP; SBO, SHORT TIME AVAILABLE	1.00E+00
SPECIAL EVENTS	
FRACTION OF RPS FAILURES THAT ARE MECHANICAL	3.33E-01
FRACTION OF CONTAINMENT FAILURES IN DRYWELL (VS. WETWELL)	2.30E-01
DUMMY ZERO VALUE	0.00
PIPE RUPTURE DUE TO HIGH PRESSURE (ISLOCA)	9.34E-03
RHR SYSTEM RELIEF VALVES FAIL TO RECLOSE	4.31E-02
FW FAILS (FRACTION OF INITIATING EVENTS THAT ARE LOFW)	1.28E-01
MAIN CONDENSER FAILS (GIVEN FW SUCCESS) AFTER ATWS	3.18E-01
MAIN CONDENSER FAILS (GIVEN FW FAILS) AFTER ATWS	6.16E-01
POWER CONVERSION SYSTEM UNAVAILABLE	4.05E-01
EVENT FAILURE	1.00E+00

TABLE 4.5.2-1 (Continued)
UNAVAILABILITY OF SYSTEMS AND PLANT FUNCTIONS

EVENT	UNAVAILABILITY
EVENT CANNOT FAIL (SENSITIVITY)	0.00
NOT AN SBO	0.00
SBO IN UNIT 1, NO SBO IN UNIT 2	1.00E+00
SBO IN UNIT 2, NO SBO IN UNIT 1	1.00E+00
SBO IN UNIT 1, SBO IN UNIT 2	1.00E+00
SBO OCCURS IN UNIT 1	1.00E+00
INDICATED LOSS OF BUS 17 IMPOSSIBLE	0.00
INDICATED LOSS OF BUS 19 IMPOSSIBLE	0.00