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Assessment of ISLOCA Risk–Methodology and Application to a Babcock and Wilcox Nuclear Power Plant

Appendices A–H

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This document presents information essential to understanding the risk associated with inter-system loss-of-coolant accidents (ISLOCAs). The methodology developed and presented in this document provides a state-of-the-art method for identifying and evaluating plant-specific hardware designs, human performance issues, and accident consequence factors relevant to the prediction of the ISLOCA risk. This ISLOCA methodology was developed and then applied to a Babcock and Wilcox (B&W) nuclear power plant. The results from this application are described in detail. For this particular B&W reference plant, the assessment indicated that the probability of a severe ISLOCA is approximately 2.2E − 06/reactor-year.

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HISTORICAL EXPERIENCE RELATED TO ISLOCA EVENTS

A.1. Summary of ISLOCA Precursor Events

A search of the LERs was performed by the INEL to collect and analyze those events that can loosely be categorized as ISLOCA precursors. This search was performed by application of computer software. This appendix is a description of these ISLOCA precursors found in the LER data base. A number of generalizations were made after reviewing the LERs. The ISLOCA precursors that resulted in an overpressurization and/or leak out of the RCS typically involve either: a.) a series of human errors, b.) inadequate procedures or, c.) existing hardware failures in combination with a human error or inadequate procedures.

The work related to a review of Licensee Event Reports (LERs) relevant to the ISLOCA was initiated in a NRC memorandum. This memorandum was from N. Thomasson, Office for Analysis and Evaluation of Operational Data/Trends & Patterns Analysis Branch (AEOD/TPAB) to S. Diab, Office of Nuclear Reactor Regulation/Risk Applications Branch (NRR/RAB). After the issuance of the memorandum, the results of LER searches performed by contractors at the Oak Ridge National Laboratory (ORNL) were made available to the INEL.

The ORNL provided the LERs on twelve systems of interest to the INEL. A review of the LERs regarding ISLOCA events for these twelve systems was performed. This review included items related to human error. The results of the review indicate that some aspect of personnel error were found in the following categories:

(a) Valve transfers open, i.e., spurious opening (214 LERs with personnel involvement),
(b) Failure to close (201 LERs with personnel errors indicated),
(c) Valve problems where maintenance might have been involved (27 LERs),
(d) Possible maintenance and testing errors linked to check valve problems (16 LERS),
(e) Maintenance and testing errors related to non check valve problems (156 LERS),
(f) Instances where valves were not tested as required and subsequently found to be inoperable (5 LERs), and

(g) Root cause analysis of valve failures after 1986 (human error - 23 times cited, design error - cited 10 times, procedural deficiency - cited 10 times, and construction error - cited one time).

Of the 1113 LERS identified by the top most search strategy, i.e. "any valve problems in the twelve interfacing systems," there were 80 LERs indicating valve problems where leaks were involved.

Regarding position/indicator alarm problems, some 42 LERs were identified, and of these, two related instances where position/indicator alarm failures were linked (presumably caused) to human errors.

A review of the LER data base produced several events that can be considered potential precursors to an ISLOCA. The pre ISLOCA events are described in the following sections in terms of the LER number and facility where the events happened.

ANO-1, 1/20/89, LER-89-002

During a complicated transient at Arkansas Nuclear One, Unit 1, a single check valve in a High-Pressure Injection (HPI) train failed to seat properly, resulting in a backflow of reactor coolant water to lines outside containment (which were not instrumented for high piping temperatures or pressures). Detection was accomplished when tape attached to the pipe began to smoke and set off a local area smoke detector causing an alarm to ring in the control room. The backflow occurred for approximately 10 to 15 minutes before the fire alarm was observed and investigated.

Because of several equipment failures, control room personnel were involved in an unusual post-trip condition that complicated their response to the initiating event. At the time when backflow was occurring, the reactor experienced a minor overcooling event caused by the overfeed of the once-through steam generators (OTSGs). Because their attention was focused on stabilizing the post-trip cooldown rate, the backflow condition was not observed. Since the backflow was not released outside of the HPI piping, no appreciable pressurizer level decrease would have been observed. However, overcooling transients do result in RCS shrinkage and an attendant decrease in pressurizer level. Thus, any leak that may have occurred might have been masked by the effects of overcooling, making detection and diagnosis difficult if other equipment did not direct the operators' attention to the condition.

Approximately 6 months later at the same unit, back leakage of reactor coolant through a faulted safety injection check valve occurred three times. The leak was detected promptly by control room personnel as a result of pressurizer level decreasing and the valve was reseated by injecting High-Pressure Safety Injection (HPSI) water through it. A second occurrence was also detected promptly and corrected in a similar fashion. The third occurrence of leakage could not be terminated by HPSI injection, and mechanical maintenance personnel were required to enter the containment building and physically reseat the valve. In all three instances, the leakage was promptly detected and monitoring was facilitated by pressure instrumentation on the low-pressure side of the valve, which causes an audible alarm in the control room.

ANO-2, 6/26/89, LER-89-012

During plant heatup, RCS backleaked three times through a safety injection system check valve. Each time, the valve was reseated by injecting water using HPI pump. The leakage rate was brought to within acceptable limits and the plant heatup continued.

During the next day, an unrelated problem forced the plant into a cooldown mode. While shutdown, the check valve was inspected and the valve disk was found to be disengaged from the disk shaft. Two rollpins, normally connecting the disc shaft to the valve disc, were found to be missing. Another check valve was inspected and both rollpins were found to be intact. However, one of the rollpins was loose and cracked. The cause of the rollpin failures was unknown.

Braidwood-1, 12/1/89, NRC IN No. 90-05

Braidwood Unit 1 experienced a discharge of 68,000 gallons of water through the inadvertent opening of an RHR suction relief valve to the hold-up tanks. The premature opening of the suction relief valve was attributed to the presence of foreign material lodged between the valve spindle and the valve guide. The response time by the crew was approximately two hours for locating the stuck open valve, terminating the discharge, and starting to refill the pressurizer. The crew performed well for this type of situation considering the absence of Emergency Operating Plans (EOP). To their credit, they were able to combine two abnormal operating procedures. Conclusions suggested that EOPs need to be available for other than at power modes and that relying on ad-hoc procedures places an unnecessary burden on crews.

Browns Ferry-1, 8/14/84, LER-84-032

Similar to the Hatch-2 event, incorrect installation or assembly of a pilot solenoid valve led to a check valve for the core spray system being held open. While designed for 500 psi, the core spray system was subsequently overpressurized (approximately 1000 psi) but was not damaged due to substantial design margins. A small relief valve on the core spray system did lift during the overpressurization.

BWR Testable Check Valves

A study by AEOD (1985) identified eight events that occurred at BWRs involving the failure of an isolation check valve. Five of these events also involved the inadvertent opening of another isolation check valve that represented the final isolation barrier between the high- and low-pressure portions of the system. Four of these events occurred during power operations and resulted in overpressurization of an ECCS system. The inadvertent opening of the final check valve in all five of the events was attributed to personnel errors during surveillance testing. The most serious of these events resulted in the contamination of thirteen workers who were sprayed by coolant from a relief valve after it was over-pressurized.

Catawba, 3/90

In March 1990, after a seven-week refueling outage, Catawba Unit 1 experienced a potential overpressurization of the RHR and RCS systems. Three RCS pressure transmitters had been isolated for welding of tube fittings during the outage and were still in an isolated state during the time of the event. Unaware of this, the operators tried to make use of inoperable instruments during filling and initial pressurization of the RCS. RCS charging flow was delivered to the pressurizer relief tank; rising tank level was used by the operators to detect that the RCS had pressurized. Errors in planning and scheduling allowed the error to occur and this worked in conjunction with the lack of procedure requiring the tagging of inoperable instruments in the control room (an error of omission on the part of management) to set the stage for the event. Part of the recovery phase was enhanced by a systems engineer who recalled an information notice on ISLOCA, and then assisted in a review of system diagrams for potential leakage paths.

Catawba, 6/11/90, EN No. 18679

After realignment of the RHR trains following check valve testing, an operator was supposed to close and lock the RHR recirculation valve before proceeding with the periodic test procedures. But, before the valve was locked closed, the operator performed the next two steps of the procedure. The two steps opened the RHR A & B trains cross connect. The RWST was then lined up with the RCS, which resulted in 5000 gallons of RCS inventory being dumped into the RWST. The RHR cross connect valve was closed within 30 seconds which stopped the loss of RCS inventory. The two operating reactor coolant pumps were manually tripped due to the loss of pressurizer level and the drop of RCS pressure (from 335 psig to 110 psig). The plant was in RHR core cooling mode at the start of this event.

Cooper, 1/21/77, LER-77-04

A testable check valve in high pressure coolant injection (HPCI) failed to fully close because of a broken sample probe (from the main feedwater line) wedged under the disk. The outboard isolation valve was opened, as required, for the HPCI System Turbine Trip and Initiation Logic Surveillance Test allowing feedwater backflow into HPCI system. Opening the valve allowed backflow of feedwater to the pump suction piping.

D. C. Cook-1&2, 10/25/88

One of two internal *Anchor Darling* swing check valve bolts (installed in the ECCS) was found broken during inspection. The other bolt was found to be cracked. The redundant ECCS train check valve was also check and was found to have one broken and one cracked bolt. This event is a generic problem related to the Diablo Canyon Event.

Diablo Canyon-2, 10/15/88

Retaining block studs found broken in RHR swing disk check valve (PIV); apparently a generic problem for *Anchor Darling* check valves (see NRC information notice 88-85 dated October 14, 1988).

Farley-2, 11/27/87

During a refueling outage at Farley Unit 2, test and maintenance personnel failed to refill a section of pipe that had been drained during testing. While stroke testing a valve on this line, this section of pipe refilled and overpressurized, causing a pressure relief valve to lift. The relief valve failed to reseat and approximately 2,400 gallons of reactor coolant discharged to the PRT, causing the rupture disk to blow.

In order to terminate the leak, an RHR train had to be isolated from the RCS. Although procedure inadequacy was cited as the cause of the initiating event, administrative controls governing these types of tests and inadequate communication during the operations-engineering planning interface also contributed to the event failure.

Hatch-2, 10/28/83, LER-83-112

At Hatch Unit 2, incorrect installation or assembly of a testable check valve that was a part of the pressure boundary between the high-pressure (RCS) and low-pressure (ECCS) systems led to the valve being held open for about 4 months. The event was thought to be due, in part, to a failure to use and follow approved maintenance and assembly procedures. The occurrence of errors was similar to those that occurred during the Browns Ferry event and are documented in Information Notice No. 84-74.

LaSalle-1, 10/5/82, LER-82-115

Testable check valve was cycled for a test while the plant was operating at 20% power, however when the air pressure was removed the valve failed to reclose. The check valve was found to be 5% open. The failure of the check valve was caused by dry actuator lubricant, degraded pre-load on the actuator spring, and the bypass valve stayed open causing the pressure to equalize across the check valve.

LaSalle-1, 6/17/83, LER-83-067

This event was similar to the LaSalle-1 10/5/82 event. The HPCS testable check valve failed to close after a quarterly test. This failure was caused by insufficient spring tension of the actuator assembly, a stuck open bypass valve, and possible thermal binding of the check valve disc. During subsequent plant shutdown, the bypass valve closed without help as the reactor pressure and temperature decreased. The check valve then closed after the bypass valve closed.

LaSalle-1, 9/14/83, LER-83-105

During RHR System Relay Logic Test, injection valves were opened (as per procedure) leaving the injection check valve as the only isolation between RHR and RCS. This valve leaked because of improper timing (misalignment of the interfacing gears) and the packing gland being too tight, both of which resulted from maintenance errors. Reactor coolant immediately began to leave the system after the check valve failed, but was secured by operators by closing the injection valve that was originally opened. During the time of the event, the reactor water level dropped from the +50" to the 0" mark. The majority of the lost water was dumped to the suppression pool while some went to the drywell.

Limerick-1, 6/1/89, LER-89-039

At Limerick Unit 1, the licensee determined, via a self-assessment, that the Shutdown Analysis was inadequately performed and that RHR overpressurization and an Interfacing Systems LOCA could occur as a result of a fire in certain areas. This was contrary to the previous Shutdown Analysis. The errors in the previous Shutdown Analysis occurred as a result of the following: (a) a lack of detailed procedures in performing the Safe Shutdown Analysis, and (b) a misunderstanding or misapplication of detailed regulatory requirements.

Three main finding of the assessment were: a fire in the main control room could spuriously open RHR suction valves, a fire in certain plant areas could open three reactor water cleanup valves, and a fire in the auxiliary equipment room could dump smoke into the shutdown room.

McGuire-2, 9/5/89, LER-89-010

While stroke timing a valve at McGuire Unit 2, operators inadvertently released, over a 30-second period, 200 gallons of primary coolant to the pressurizer relief tank (PRT) and 2000 gallons to the auxiliary building. Operators were alerted to the abnormal condition when they observed pressurizer level decreasing and pressurizer relief tank level increasing. While attempting to return the system to pretest status, they subsequently opened another valve that began draining the refueling water storage tank (RWST). Approximately 8,000 gallons of water from the RWST were also drained to the auxiliary building over a 30-minute period. Control room personnel were notified of the flooding in the auxiliary building by Radwaste Chemistry personnel.

A year prior to this flooding event, a valve stroke timing test resulted in the overpressurization of the chemical and volume control (CVC) system. Although procedural changes were made to preclude the recurrence of that event, the changes only addressed the operation of valves that were involved in that particular event. The valves involved in the 9/5/89 event were overlooked when implementing that procedural change. Operators' attention was focused on preventing the reoccurrence of the last event, therefore other overpressurization and backleakage pathways were ignored.

In addition, the procedure required a review of system conditions prior to initiation of the test, it did not adequately address all conditions that could exist. The operator(s) had a high degree of confidence in the technical adequacy of the procedure they were following and hence, did not recognize the existence of potential abnormal conditions that could arise as a result. Thus, a combination of procedural inadequacies, training that focused operator attention to prevent a specific event, operator's belief in the adequacy of procedures, and inattention to potential problems contributed to this flooding event.

Pilgrim-1, 9/29/83, LER-83-048

Overpressurization of the HPCI system pump suction occurred during a test of the HPCI logic system. At the time of the event, the reactor was at 96% rated power. The cause of the event was inadvertent opening of two HPCI discharge valves coupled with a partially open HPCI testable check valve. Rusted linkage on the check valve contributed to the check valve failure. Verbal miscommunication between operators resulted in the opening of the two HPCI valves.

Pilgrim-1, 4/12/89, LER-89-014

During preparation for the Reactor Core Isolation Cooling (RCIC) logic function system test, six circuit breakers to motor-operated valves were incorrectly positioned. At the time, the reactor was at 25% rated power and ascending. An Instrument & Control technician, a Control Room Operator, and an Equipment Operator divided the task of positioning the breakers at the local area and incorrectly positioned the breakers. During verification of the tagouts for the breakers, they did not detect the errors the others had made. In addition, local inspection and verification of the circuit breakers was not conducted by the supervisor as required.

Low-pressure RCIC suction piping was exposed to high-pressure reactor coolant due to the incorrect breaker positions and approximately 100 gallons of reactor coolant (at 1000 psig and 300°F) was discharged to an area quadrant in a mixture of steam and water. The RCIC was subsequently declared inoperable and a plant shutdown was completed 4 days into a 7 day Limiting Condition for Operation (LCO) for RCIC recovery.

No pre-evaluation briefing was conducted by the operating shift prior to preparation for the RCIC logic function system test, although required by Technical Specifications. Two of the personnel were performing this test for the first time. The two operators (the CRO and the EO) were unaware of reasons for the tagouts and said they were only following the instructions on the tagout sheet. Both operators had attended an on-watch training module for tagging prior to this event. In addition, the procedure did not include precautions to warn workers of the effect that incorrectly performing the steps would have on the safety system.

Pilgrim-1, 2/11/86, LER-86-003

During maintenance of electrical cable, a 480 V safety-related bus was inadvertently de-energized resulting in the disablement of some primary containment isolation capability. The event occurred during a scheduled maintenance activity. The electricians grounded a non-safety cable, which should have opened a supply breaker. The supply breaker failed closed which tripped the main feeder breaker thereby de-energizing the bus. The bus was without power for approximately eight minutes. The cause for the event was a temporary modification (performed in 1976) that was not reflected in the plant electrical prints and the electricians not verifying that the cables were de-energized before attempting to work on the cable.

River Bend, 10/7/89, LER-89-036

During normal full power operation, it was discovered that various motor operated valves (including thirteen in the RHR system and one in the reactor core isolation cooling system) in the plant were energized when the valves should be de-energized according to the design basis of the plant concerning the plant fire hazards analysis. Accordingly, fire watches were initiated on the valves or the valves were de-energized. Two of the valves were considered to be in a potential ISLOCA pathway, and could be activated during a fire if the valves were energized.

Salem-2, 10/27/89, EN No. 17242

While the reactor was in hot shutdown, ECCS check valves started to leak, resulting in pressurizing the RHR suction piping to 600 psig. The reactor was returning to service and the RCS was in the process of being pressurized. The design pressure for the RHR system was 450 psig. The check valves were repaired and returned to service.

Sequoyah-1, 5/23/88, LER-88-021

With Unit-1 in cold shutdown, the A train of the RHR was placed in service (to enhance the B train operability). An operator went to open two RHR valves, but wrote the valve numbers incorrectly. Consequently, a manual valve used to align the RHR discharge with the RWST was opened. After opening the manual valve, the operator noted unusual flow noises and called the operator in the control room. The control room operator stopped the B train RHR pump and entered procedures for a loss of RHR. The cause of the event was a miscommunication. Inventory loss to the RWST was estimated to be 6000 gallons.

Foreign Reactor Event, December 1987, Inside NRC December 5, 1988

During plant startup the motor operated check valves isolating the RHR system from the RCS were closed by actuating their motor operators. However, one of the check valves did not close fully. This condition was displayed on both the control room indicator and on a CRT alarm. The operators chose to ignore the position indication and alarm (believing them to be false) and continued startup activities. A 2-mm relief valve opened but its defective indicator did not show it as being open. Approximately 14-hours after startup a high temperature alarm in a CVCS filter actuated. It was at this point when it was first recognized that the check valve was not closed and the decision was made to shutdown the plant. At some point during the shutdown the operators attempted to close the check valve by opening a down-stream MOV in order to create a differential pressure across the valve. This did not work and the plant was shut down.

Shoreham, 7/24/85, LER-85-031

A low reactor water level signal (while in refuel mode) started the reactor protection system. The low reactor water level was caused by an operator inadvertently opening the RHR suction valve before the shutdown cooling valve had completely closed. The event resulted in a direct path from the reactor vessel to the suppression pool. A low reactor water trip caused the shutdown cooling suction valves to close isolating the leak path. The water level was restored within 15 minutes.

Susquehanna-2, 5/28/84, LER-84-006

Dual indication (both open and close) prompted control room operators to attempt to reseat a testable check valve by opening an injection valve (which was normally closed). Opening the injection valve allowed backleakage to the RHR heat exchanger. The injection valve was then closed. A loose diaphragm plate connector caused an improper contact which resulted in the dual indication. The injection valve failed to completely close after cycling.

Trojan, 4/9/89, LER-89-009

During cold shutdown at the Trojan plant, one of two residual heat removal (RHR) isolation valves was determined to be inoperable after it was discovered that the valve would not close automatically. The valve had been wired incorrectly: its placement was based on an inadequate as-built drawing. Post installation testing did not detect this problem because this particular failure mode was not considered. Thus, the valve would have opened at any pressure on an auto-open signal but would not have responded to the auto-close signal, rendering low-pressure RHR piping vulnerable to a failure of the other check valve. Although detected during the *1989* refueling outage, the error occurred during the *1988* refueling outage, indicating that the plant operated in this condition during the interval between outages.

Vermont Yankee, 12/12/75, LER-75-24

A testable check valve did not seat properly and allowed overpressurization even though indicator lights in the control room showed that the valve was completely closed. Before stroke testing of the normally closed injection MOV, a normally open MOV was closed. However, the MOV did not close completely (approximately 1" was left opened). When the injection MOV was opened, it allowed backleakage into the RHR system though the testable check valve. A steam water mixture was discharged from a tube sheet-to-shell flange and three RHR system relief valves.

Vogtle-2, 3/9/89, LER-89-003

To prepare for initial heatup at Vogtle Unit 2, control room personnel were preparing to perform a pressure isolation valve leakage test. In order to establish test conditions, the shift supervisor decided, without approved procedures, to de-pressurize the RHR system by momentarily opening two locked-closed valves (an error in intention). Accordingly, an equipment operator was dispatched by a reactor operator to open the two locked-closed valves but not to return them to a closed position (due to a misunderstanding between the SS and the RO). The reactor operator duplicated this error and subsequently dispatched a second equipment operator to verify that the valves were open. Both RHR valves were left locked open for 14 hours. Upon discovery, both RHR trains were declared inoperable.

This failure was attributed to the shift supervisor failing to follow approved procedures, and inadequate communication between control room personnel. The shift supervisor failed to ensure that the valves were returned to the closed position, as required by technical specifications, and other knowledgeable shift personnel failed to point out implications that this would have on the unit. During this event, RCS coolant passed from the RHR system to the refueling water storage tank, and from there to the atmosphere. However, because the unit had not achieved its initial criticality, no radiation was released.

A.2. Summary of Valve Event Rate Determination

N. T. O'Connor

Problem Definition

The purpose of this calculation is to determine the rate at which operators inappropriately open remotely operated valves. These rates were compared with human failure rates generated by other means in this ISLOCA study. In order to perform the calculation, it was necessary to collect data on the number of applicable events, and information on an appropriate valve population matched to the events. The event data was obtained from Licensee Event Reports (LERs), and the valve population data was obtained from the Nuclear Plant Reliability Data System (NPRDS). Although there are several limitations on the use of both of these data sources (expounded upon in the body of this document), it is believed that this failure rate determination will serve as a rough check on the primary calculation.

Results

Failure rates expressed in terms of events per valve-hour were calculated for nine different valve applications. The valve applications and their associated rates are listed below.

Valve Application	Events Per Valve-Hour	
Containment Spray Suction Valve from Containment Sump	1.9E-7	
Main Steam Isolation Valve (BWR only)	7.4E-8	(Lowest rate)
Residual Heat Removal (RHR) Containment Sump Suction Valve	2.6E-7	
Residual Heat Removal (RHR) Shutdown Cooling Suction Valve	5.9E-7	
Residual Heat Removal (RHR) Hot Leg Injection Isolation Valves	1.5E-7	
Residual Heat Removal (RHR) Suction Valve from Reactor Coolant System	1.1E-7	
Residual Heat Removal (RHR) Discharge Valve to Torus	3.3E-7	

Residual Heat Removal (RHR) Suction Valve from Suppression Pool	4.8E-7	
Reactor Core Isolation Cooling (RCIC) Steam Supply Isolation Valve	6.8E-7	(Highest rate)
	2.2E-7	(Aggregate)

The aggregate event rate (determined by summing all of the failures and all of the valve-year data) is 2.2E-7 events per valve-hour.

Detailed Description of Methodology

Problem Definition

The purpose of this calculation is to determine the rate at which operators inappropriately open remotely operated valves. The calculated rate will be used as an independent check of a rate determined via a different method. LERs and the NPRDS were selected as the data sources for this determination.

The original problem leading to this calculation was a need for a backup calculation showing the probability that an operator will open a RHR letdown motor-operated valve when it is not appropriate to do so. However, insufficient data are available in the LERs to answer this specific question. Therefore, the general problem of determining the rate at which operators open remotely operated valves at an inappropriate time was formulated. The rate calculated is expressed in terms of events per valve-hour.

In order to calculate such an event rate, it is necessary to determine the number of events occurring within a certain time frame for the numerator, and to determine valve-hours based on an appropriate population of valves in service during the same time period for the denominator. The manner in which the event data and the valve-hour data were obtained is discussed below, as is the final event rate determination.

Event Data

LIMITATIONS

The events composing the numerator of this event rate calculation were identified from existing LERs. The user must be cognizant of the limitations of using LERs to derive the data for this type of calculation.

Not every event is reported in LERs. Only certain types of events are required to be reported in LERs (10 CFR 50.73). Thus, when answering a specific question, such as the one in this event rate determination, it is probable that there are many more events that would be of interest than are actually reported in LERs. Therefore, the rate at which the particular failure mechanism actually occurs is very likely to be higher than the calculated rate.

On the other hand, it must also be realized that events with some operational or safety significance generally are reportable under 10 CFR 50.73. For this study, then, it is reasonable to determine the rate of occurrence of events in which inappropriately opening a valve results in some operational or safety concern.

SEARCH STRATEGY

The only available system for rapidly searching all LERs for events involving a particular type of problem is the Sequence Coding and Search System (SCSS). Information about the event described in the LER is captured by SCSS in coded fields; by searching these fields for particular codes, the events of interest are identified. The various commands and field codes used in SCSS are described in *Sequence Coding and Search System for Licensee Event Reports*, ORNL/NOAC-231/VI.

The user of the event rate information from this database must understand the limitations of the searching strategy. The accuracy of the LER search depends heavily upon the accuracy of the SCSS coding. It is assumed that the SCSS coding is accurate and consistent; if not, certain events may have been missed. (Additional events that are not applicable would not appear in the final event rate calculation because of an event screening step in the search strategy. No formal study has been done to verify the validity of the assumption concerning SCSS consistency.)

The search for this event rate determination was conducted in two steps. First, SCSS was searched to identify all of the events in which an operator action was linked to opening a valve. This search strategy, which produced nearly 2000 events, was as follows:

```
*
FIND
+
<ICOMP> 270 350 <EFF> (AK AI)
+
END
```

(Note: This part of the search identifies all of the LERs containing an operator action [ICOMP 270], a valve [ICOMP 350], and either an effect code of "transfer open" [EFF AK] or an effect code of "open" [EFF AI]. A total of 2183 LERs were identified in this part of the search.)

```
*
LINK
+
<ICOMP 270>
+
<EFF> (AK AI)
+
END
```

(Note: This part of the search looks through already identified LERs for those events in which the operator action [ICOMP 270] is linked to or causes a later "transfer open" code [EFF AK] or the "open" code [EFF AI]. There can be several intervening steps between the operator action and the "transfer open" or "open," therefore, the relationship is not necessarily a direct causal effect. Of the 2183 LERs previously identified, 1977 fulfilled the criteria of this part of the search.)

Because the valves in the event data must be matched to a representative population for the event rate determination, and because good valve population data is available from NPRDS for only certain systems, the various systems involved in the events were listed using the following SCSS command:

```
*
VALUES * <PSYS>
```

(Note: This command produces a table of the various systems involved in the LERs previously identified. The systems may or may not be associated with the valve failure - if the system is named in the LER, it will appear on this list.)

Table A-1 contains the resultant list of systems involved in the LERs. Certain systems containing components readily identifiable within NPRDS were selected for further evaluation. A second SCSS search was then conducted for each system to identify the LERs containing a valve "transfer open" event of interest. (The "open" code, [EFF AI], was eliminated because evaluation of a sample of the LERs identified with this code showed that events of this type were not of interest in this determination.) A typical example of the logic used in these searches is as follows:

```
*
FIND
+
<ICOMP> 270 350 <EFF> AK <PSYS> BH
+
END
*
LINK
+
<ICOMP> 270
+
<ICOMP> 350 <EFF> AK <PSYS> BH
+
END
*
DISPLAY
```

(Note: The only functional differences between this search strategy and the one used previously is the inclusion of the [PSYS BH] filter to identify only those LERs containing a valve "transfer open" in the "BH" system, and the exclusion of the [EFF AI] code. The search was repeated for each of the various systems selected.)

A list of LERs resulted from each system-oriented search. The LERs were then individually evaluated to determine whether or not the event should be counted when determining the event rate. Only those LERs with event dates between January 1, 1984 and June 30, 1990 were included. The 1984 cutoff was selected to minimize the impact of differences in LER reporting practices arising from a revision to the LER reporting rule. The 1990 cutoff was selected because more recent LER data may be incomplete, and because it is convenient to cut off the data at the end of the quarter when calculating the valve-hour data. Documentation of the LER evaluations is provided in Table A-2.

Table A-1. Systems involved in events with valve transfer open problems

Search Strategy Information (downloaded directly from SCSS)

VALUES * <PSYS>

A "VALUES" ANALYSIS WILL BE PERFORMED FOR THE PSYS FIELD

1977 OUT OF 1977 LERS HAVE BEEN PROCESSED
SECONDS: 37.76 (CPU) 363.53 (CLOCK)--RATIO:0.104

1977 OUT OF 1977 LERs CONTRIBUTED TO THE ANALYSIS

THE ACTIVE LIST OF 1977 LERs HAS 117 UNIQUE VALUES IN THE PSYS FIELD FOR THE STEPS INCLUDED IN THIS ANALYSIS

(Note: The system listed may not necessarily be the system linked to the valve problem.)

KEY VALUE	NUMBER OF STEPS	DESCRIPTION
PO	1169(18.1%)	OPERATION ACTIVITY
DB	877(13.3%)	CONTAINMENT ISOLATION
PT	682(10.4%)	TEST/CALIBRATION ACTIVITY
PM	620(9.4%)	MAINTENANCE/REPAIR ACTIVITY
PD	337(5.1%)	DESIGN ACTIVITY
PZ	208(3.2%)	UNKNOWN ACTIVITY
FI	184(2.8%)	CONDENSATE AND FEEDWATER
EC	178(2.7%)	LOW VOLTAGE ac (LESS THAN 600V)
PA	157(2.4%)	ADMINISTRATIVE ACTIVITY
KF	119(1.8%)	FIRE PROTECTION
PI	97(1.5%)	INSTALLATION ACTIVITY
EB	95(1.4%)	MEDIUM VOLTAGE ac (35KV TO 600V)
FF	95(1.4%)	TURBINE BYPASS
PF	87(1.3%)	FABRICATION ACTIVITY
IW	85(1.3%)	ENGINEERED SAFETY FEATURES ACTUATION
AF	81(1.2%)	PRESSURIZER (PWR)
BP	77(1.2%)	MAIN STEAM PRESSURE RELIEF (PWR)
BK	71(1.1%)	CHEMICAL AND VOLUME CONTROL (PWR)
ED	70(1.1%)	VITAL INSTRUMENT, CONTROL, AND COMPUTER ac
IU	66(1.0%)	REACTOR PROTECTION
FB	65(1.0%)	TURBOGENERATOR
EA	61(0.9%)	HIGH VOLTAGE ac (GREATER THAN 35KV)
IZ	59(0.9%)	NON NUCLEAR INSTRUMENTATION
WP	59(0.9%)	REACTOR WATER CLEANUP (BWR)
EE	53(0.8%)	dc
BA	52(0.8%)	AUXILIARY FEEDWATER (PWR)
PC	51(0.8%)	CONSTRUCTION ACTIVITY
BR	49(0.7%)	NUCLEAR BOILER OVERPRESSURE PROTECTION (BWR)
BF	42(0.6%)	RESIDUAL HEAT REMOVAL (PWR)
CB	39(0.6%)	ESSENTIAL RAW COOLING/SERVICE WATER
BH	35(0.5%)	RESIDUAL HEAT REMOVAL (BWR)
KB	27(0.4%)	SAMPLING

Table A-1. (continued)

AB	25(0.4%)	CONTROL ROD DRIVE
WA	25(0.4%)	LIQUID RADWASTE
CA	23(0.3%)	COMPONENT COOLING WATER
PR	21(0.3%)	RADIATION PROTECTION ACTIVITY
BS	20(0.3%)	CORE FLOODING ACCUMULATOR (PWR)
WC	19(0.3%)	GASEOUS RADWASTE
FA	19(0.3%)	MAIN STEAM
HA	18(0.3%)	REACTOR BUILDING HVAC
DE	17(0.3%)	CONTAINMENT SPRAY
IL	17(0.3%)	LEAK MONITORING
AD	17(0.3%)	REACTOR VESSEL
II	16(0.2%)	TURBOGENERATOR INSTRUMENTATION AND CONTROL
IT	16(0.2%)	FEEDWATER CONTROL
BN	16(0.2%)	HIGH-PRESSURE COOLANT INJECTION (BWR)
HH	13(0.2%)	CONTROL BUILDING HVAC
FL	13(0.2%)	VARIOUS THERMAL CYCLE DRAINS AND VENTS
IN	12(0.2%)	RADIATION MONITORING
HC	11(0.2%)	PRIMARY CONTAINMENT VACUUM RELIEF
SF	11(0.2%)	REACTOR AUXILIARY BUILDING
BC	11(0.2%)	REACTOR CORE ISOLATION COOLING (BWR)
CD	10(0.2%)	BORATED/REFUELING WATER STORAGE (PWR)
BL	10(0.2%)	INTERMEDIATE PRESSURE INJECTION (PWR)
FK	9(0.1%)	MOISTURE SEPARATORS, REHEATERS
FP	9(0.1%)	CONDENSATE DEMINERALIZER
KP	9(0.1%)	LUBE OIL
SE	9(0.1%)	SECONDARY REACTOR CONTAINMENT (BWR)
AI	8(0.1%)	RECIRCULATING WATER (BWR)
HF	8(0.1%)	REACTOR AUXILIARY BUILDING HVAC
KA	8(0.1%)	AUXILIARY STEAM
AE	8(0.1%)	PRIMARY COOLANT (PWR)
KH	7(0.1%)	COMPRESSED GAS
KC	7(0.1%)	CONTROL AND SERVICE AIR
FD	7(0.1%)	MAIN CONDENSER
WK	7(0.1%)	EQUIPMENT DRAINAGE (INCLUDING VENTS)
CC	7(0.1%)	ESSENTIAL COMPRESSED AIR
BE	7(0.1%)	STANDBY LIQUID CONTROL (BWR)
PX	6(0.1%)	OTHER ACTIVITY
AH	6(0.1%)	STEAM GENERATOR (PWR)
BD	6(0.1%)	EMERGENCY BORATION (PWR)
IJ	6(0.1%)	PLANT MONITORING
KD	5(0.1%)	DEMINERALIZED WATER
HD	5(0.1%)	SECONDARY CONTAINMENT HVAC-STANDBY GAS TREATMENT
BX	5(0.1%)	LOW-PRESSURE CORE SPRAY (BWR)
FR	5(0.1%)	CIRCULATING WATER
CK	5(0.1%)	EMERGENCY GENERATOR STARTING
DA	5(0.1%)	SPENT FUEL POOL/REFUELING POOL COOLING AND CLEANUP
IY	5(0.1%)	ATWS
SW	5(0.1%)	MISCELLANEOUS/UNKNOWN STRUCTURES
ZZ	4(0.1%)	UNKNOWN
HE	4(0.1%)	DRYWELL/TORUS HVAC AND PURGE (BWR)
IP	4(0.1%)	REACTOR POWER CONTROL

Table A-1. (continued)

CF	4(0.1%)	CONDENSATE STORAGE
FH	4(0.1%)	STEAM EXTRACTION
WF	4(0.1%)	STEAM GENERATOR BLOWDOWN (PWR)
BW	4(0.1%)	HIGH-PRESSURE CORE SPRAY (BWR)
IH	4(0.1%)	EMERGENCY GENERATOR INSTRUMENTATION AND CONTROLS
DH	3(0.0%)	CONTAINMENT COMBUSTIBLE GAS CONTROL
KR	3(0.0%)	BORON RECOVERY
IC	3(0.0%)	PANELS
FC	3(0.0%)	TURBOGENERATOR TURBINE STEAM SEALING
ZX	2(0.0%)	OTHER
KX	2(0.0%)	CHEMICAL ADDITIVE INJECTION
HI	2(0.0%)	EMERGENCY GENERATOR HVAC
FE	2(0.0%)	NON-CONDENSIBLE GASES EXTRACTION
HT	2(0.0%)	CHILLED WATER
BT	2(0.0%)	UPPER HEAD INJECTION (PWR)
SC	2(0.0%)	REACTOR DRYWELL (BWR)
CI	1(0.0%)	EMERGENCY GENERATOR FUEL
KW	1(0.0%)	RAW SERVICE WATER
HR	1(0.0%)	PUMPING STATIONS HVAC
DD	1(0.0%)	CONTAINMENT ISOLATION LEAKAGE CONTROL
SH	1(0.0%)	CONTROL BUILDING
WE	1(0.0%)	NONRADIOACTIVE WASTE (LIQUID, SOLID, & GASEOUS)
DI	1(0.0%)	CONTAINMENT ICE CONDENSER (PWR)
HS	1(0.0%)	MISCELLANEOUS STRUCTURES HVAC
SU	1(0.0%)	PANELS
CH	1(0.0%)	EMERGENCY GENERATOR LUBE OIL
IS	1(0.0%)	RECIRCULATION FLOW CONTROL (BWR)
SL	1(0.0%)	TURBINE BUILDING
KI	1(0.0%)	POTABLE AND SANITARY WATER
EN	1(0.0%)	CONDUIT AND CABLE TRAY
IX	1(0.0%)	SOLID STATE PROTECTION/CONTROL
CL	1(0.0%)	EMERGENCY GENERATOR COOLING
DF	1(0.0%)	CONTAINMENT PRESSURE SUPPRESSION MAKE-UP (BWR)
SK	1(0.0%)	FUEL BUILDING
	SECONDS:	38.47 (CPU) 366.63 (CLOCK)--RATIO:0.105

Table A-2. Licensee Event Report evaluations

PWR Residual Heat Removal System

LER Number	Applicable	NPRDS Code	Description
244/84-003	Yes	RHCTSCVA	RHR containment sump suction valves were opened before shutting the appropriate downstream valve; the procedure was not followed.
255/86-034	No		Design problem.
269/87-002	No		Appendix R problem.
327/88-021	No		Loss of RHR when incorrect manual valve (RHR pump discharge to RWST) was opened.
327/89-011	Yes	RHRXDSVA	Two different cases of opening the hot leg injection isolation valve per procedure, but the procedure was wrong.
328/84-012	Yes	(None available)	RHR to RWST recirculation valve was opened per procedure, but the procedure was wrong.
338/87-022	Yes	(None available)	RHR heat exchanger outlet valve was stroked open per procedure, but the procedure should not have been done under existing plant conditions.
346/87-011	No		Nitrogen bubble in RHR piping.
348/86-020	No		The operator increased plant pressure too rapidly, thereby lifting a RHR loop suction relief valve.
361/84-017	No		Shutdown cooling heat exchanger flow control valve was found locked open.
364/87-008	Yes	RHCTSCVA	RHR containment sump suction valve was stroke tested per procedure, but should not have been because piping was not filled. This resulted in a pressure pulse that opened a relief valve.

Table A-2. (continued)

370/89-010	(No)		Not included on this page because it is a <u>containment spray</u> system valve -- see containment spray system.
382/86-015	No		RHR valves were not closed when they should have been.
425/89-003	Yes	(None available)	Two RHR test return valves were opened outside of procedure.
456/88-008	No		RHR pump drain valve was found cracked open.
456/89-016	No		RHR pump suction relief lifted due to inadequate pressure control.
457/90-002	Yes	RHRXSCVA	RHR suction valve from reactor coolant system was opened inappropriately. Because another valve had been left in an unusual position, a flow path was created; the Senior Reactor Operator should have recognized this.
482/85-066	Yes	(None available)	RHR recirculation valve to RWST was opened on two occasions in accordance with a deficient surveillance procedure.
483/84-016	Yes	(None available)	RHR injection balance line isolation valve was opened per procedure. This should not have been done, as it created a path to pump the reactor coolant system to the RWST via the RHR system.

<u>BWR Residual Heat Removal System</u>

<u>LER Number</u>	<u>Applicable</u>	<u>NPRDS Code</u>	<u>Description</u>
324/84-011	Yes	(None available)	Operator opened the RHR discharge to the radioactive waste control system valves because he thought RHR was lined up to the suppression pool, not the reactor vessel.
324/84-014	No		The valve was travelling because its jack was in the wrong place.

Table A-2. (continued)

374/84-009	No		Improper valve lineup due to breaker miss-positioning. (This is considered to be a different problem than an operator miss positioning a valve during routine operations.)
263/85-004	Yes	(None available)	The RHR inter-tie line valve was opened per an improper test procedure.
277/85-020	Yes	(None available)	The full flow test return to the torus valve was opened; the operator did not recognize that a flow path was created.
322/85-031	Yes	RHSPSCVA	The RHR suction valve from the suppression pool was opened prematurely.
324/85-012	No		A reactor scram occurred while testing RHR valves; believed to have been caused by a pressure pulse created during valve testing.
352/85-064	Yes	RHRRSCVA	A shutdown cooling suction valve was opened prematurely.
388/85-016	Yes	(None available)	RHR injection line flow control valve was opened per a faulty procedure.
397/85-030	Yes	RHSPSCVA	RHR suction valve from the suppression pool was opened prematurely. (Same as LER 322/85-031)
458/85-008	Yes	RHSPSCVA	RHR suction valve from the suppression pool was opened prematurely. (Same as LER 322/85-031)
265/87-010	Yes	(None available)	RHR system test return valves were opened to perform an evolution not covered by a procedure.
321/87-010	Yes	RHSPDSVA	Two RHR discharge valves to torus were opened to perform an evolution not covered by a procedure.

Table A-2. (continued)

397/87-013	No		Valve wire problem.
298/88-010	No		Valve mechanical problem.
271/89-013	Yes	RHRHSCVA	Two RHR pump shutdown cooling suction valves were opened per procedure without realizing a drain path had been created.
387/89-003	No		A faulty procedure allowed opening a valve without having completely filled a line. (Although this is very similar to other events included, the circumstances in this event are such that it is considered to be a different problem than the one of interest.)
461/89-009	No		Automatic valve actuation, not operation action.
397/90-006	No		Appendix R problem.

Main Steam System

LER Number	Applicable	NPRDS Code	Description
301/88-001	No		MSIV opened as a result of an improperly operated breaker.
312/88-019	No		Auxiliary steam valve was used to control pressure after a regulating valve failed.
370/88-009	No		A main steam miscellaneous drain valve was failed open for maintenance. (This is a different type of problem than an operator miss positioning a valve.)
397/84-090	No		Main steam drain valves were opened per a startup check list without having appropriate plant conditions. (It is believed that these are not remotely operated valves, and are therefore not applicable.)

Table A-2. (continued)

400/87-019	No		No main steam valve problem in LER.
413/86-021	No		Personnel opened a sliding link which caused an auxiliary feedwater valve to open.
440/86-053	Yes	MSISBVA	The main steam isolation valves (MSIVs) were opened as per the startup procedure, but should not have been because this created a lineup to draw a vacuum in the reactor vessel from the main condenser.
440/86-054	No		No event of the desired type in this LER.
455/87-002	No		No valve miss positioning in this LER.
461/89-033	No		Main steam isolation valve bypass valves and main steam line warmup valve were opened after an extended length of time in Mode 3. This allowed condensate in the line to cause a pressure pulse that resulted in an ESF actuation.
528/85-063	No		Following a reactor trip, having the main steam drain valves open contributed to an unusually high cooldown rate.

Containment Spray System

LER Number	Applicable	NPRDS Code	Description
255/84-005	No		Engineered safety feature (ESF) actuation while replacing a fuse.
280/85-012	No		Equipment failure, not a personnel problem.
280/88-038	No		No operator error.
295-85-035	No		Maintenance personnel shorted two MOV leads together on containment spray adductor isolation valve and caused it to open.

Table A-2. (continued)

339/84-009	Yes	(None available)	A pump casing cooling line isolation valve was opened as per a faulty procedure.
370/89-010	Yes	CSCTSCVA	The containment spray suction valve from the containment sump was stroke tested per a procedure that had insufficient precautions.

Reactor Core Isolation Cooling System

LER Number	Applicable	NPRDS Code	Description
293/89-014	No		Valves were opened during a test when a relay was actuated.
325/88-020	No		No events of the desired type in this LER.
373/84-060	No		No operator errors linked to valve operations in this LER.
374/84-025	No		A valve opened automatically when an instrument was vented.
416/84-051	No		A high steam flow experienced when opening valves caused an automatic isolation. (This is more a problem with inadequately controlling the opening rate of a valve than it is with choosing a wrong valve to open or opening a valve when it should not have been.)
416/84-056	No		A high steam flow experienced when opening valves caused an automatic isolation. (Similar to LER 416/84-051)
416/89-009	Yes	RITUMSVAT	The shift supervisor (SS) directed a tag clearance and opening of RCIC isolation valves without warming the lines. (It is assumed that this means a steam supply isolation valve.)

Table A-2. (continued)

Intermediate Pressure Injection System

LER Number	Applicable	NPRDS Code	Description
255/84-005	No		Inadvertent actuation of safety injection while replacing fuses.
306/84-001	Yes	(None available)	Boric acid storage tank suction was automatically swapped when an operator followed a faulty test procedure.
362/84-035	Yes	(None available)	Safety injection minimum flow bypass valves were opened for a surveillance test procedure that should not have been performed because the other train was already inoperable.
370/89-011	No		Opening a valve caused an entry into Technical Specification 3.0.3; opening the valve was a necessary and appropriate action.
413/90-003	No		Safety injection system valves were opened to relieve pressure in a cold leg injection line caused by leaking check valves. This placed the plant in an un analyzed condition. (Although this event might be of the right type for this investigation, further evaluation will not be done because there are no NPRDS codes to cover these valves.)

SPECIAL LERs and GENERAL OBSERVATIONS

While reviewing the LERs to find applicable events, certain LERs were found to be especially pertinent because they are typical of certain kinds of events. Copies of these LERs are provided as a supplement to this report.

In general, the largest number of events in which valves were opened inappropriately involved inadequate procedures. The problems with procedures include (a) situations or plant conditions not covered by procedures, (b) errors in the sequence of valve manipulations in the procedures, and (c) inadequate precautions in the procedures. Another apparent problem is the difficulty in visualizing alternate flow paths that could be created by opening a valve.

One particular type of event is particularly noteworthy because it occurred at three different plants over a relatively short time. In LERs 32285031, 39785030, and 45885008, RHR suction valve from the suppression pool was opened prematurely, creating a drain path from the reactor vessel to the suppression pool. The root cause in all of these events was the failure to recognize that additional time was required to allow a certain valve in the path to completely close prior to opening the RHR suction valve. (It appears that this problem has been resolved across the industry; no other events have been reported since 1985.)

Valve-Hour Data

The various valves identified in the event data were cross-referenced to NPRDS application codes where possible. The NPRDS application codes correspond to specific valves in the plant. For example, a main steam isolation valve in a BWR plant is identified with the application code MSISBVA. NPRDS contains information for each commercial nuclear power plant, specifically, the number of components with a particular application code present in the plant, and the dates on which the component was put into and taken out of service. Thus, valve-hours can be readily calculated for those valves with application codes in NPRDS.

The computer program that derives the valve-hours from NPRDS data first performs a quality check on the data to ensure that only reasonable dates are used. Then it calculates the number of hours a particular component has been in service at a particular plant on a quarterly basis, and sums the hours for the individual components across all plants for each quarter. Finally, it sums all quarterly values to derive a total number of in-service hours.

It is important to note that the information on valve-hours represents *in-service hours*, and not necessarily *operating hours*. Essentially, the number of hours calculated is equivalent to the number of calendar hours in the quarter multiplied by the number of components; adjusting as necessary for in-service or out-of-service dates that fall during the middle of a quarter. Therefore, the final event rate more closely resembles a rate based on a calendar time period than one based on actual system operating time.

The calculated valve-hours are presented in Table A-3.

Event Rate Determination

The event rate is calculated by simply dividing the total number of in-service hours for an application code into the number of applicable events involving that application code.

$$\textit{Event Rate} = \frac{\textit{Number of Events}}{\textit{Total Hours}}$$

An aggregate failure rate for all of the component groups (application codes) was determined by summing all of the events, summing all of the hours, and dividing the hours into the events.

As discussed above, the final event rates can most accurately be thought of as "the number of safety significant or operationally significant events in which an operator inappropriately opened a valve per valve in-service hour." The calculated rates are presented in Table A-4. Because of the inherent bias in the searching strategy toward missing some applicable events rather than toward including nonapplicable events, the event rates are more accurately thought of as minimum rates (i.e., the events happen at least this often) rather than maximum rates.

Table A-3. Valve hour data

Trend Analysis of NPRDS Key Component Data--84Q1 Through 90Q2
Number of Million Hours by Application Code and Quarter
Index 1 is 1984, Q1; Index 26 is 1990, Q2

APPL	DENOM1	DENOM2	DENOM3	DENOM4	DENOM5	DENOM6	DENOM7	DENOM8	DENOM9
CSCTSCVA	0.159	0.166	0.168	0.168	0.168	0.170	0.179	0.190	0.186
MSISBVA	0.402	0.404	0.424	0.428	0.441	0.454	0.477	0.477	0.478
MSISPVA	0.304	0.314	0.318	0.319	0.320	0.329	0.349	0.367	0.367
RHCTSCVA	0.244	0.251	0.254	0.255	0.253	0.258	0.270	0.280	0.278
RHRRSCVA	0.100	0.101	0.106	0.107	0.110	0.114	0.119	0.119	0.119
RHRXDSVA	0.392	0.408	0.413	0.414	0.415	0.426	0.447	0.470	0.469
RHRXSCVA	0.240	0.251	0.254	0.255	0.257	0.263	0.281	0.302	0.303
RHSPDSVA	0.170	0.170	0.172	0.172	0.168	0.170	0.172	0.172	0.168
RHSPSCVA	0.190	0.191	0.199	0.200	0.205	0.212	0.221	0.221	0.222
RITUMSVA	0.041	0.042	0.044	0.045	0.047	0.048	0.051	0.051	0.051

APPL	DENOM10	DENOM11	DENOM12	DENOM13	DENOM14	DENOM15	DENOM16	DENOM17	DENOM18
CSCTSCVA	0.194	0.201	0.203	0.199	0.205	0.212	0.221	0.223	0.223
MSISBVA	0.492	0.512	0.515	0.518	0.524	0.530	0.546	0.572	0.593
MSISPVA	0.386	0.398	0.411	0.402	0.416	0.439	0.447	0.453	0.454
RHCTSCVA	0.266	0.292	0.298	0.292	0.302	0.313	0.316	0.316	0.317
RHRRSCVA	0.123	0.128	0.129	0.130	0.131	0.132	0.136	0.143	0.148
RHRXDSVA	0.491	0.503	0.514	0.503	0.516	0.533	0.545	0.553	0.555
RHRXSCVA	0.319	0.330	0.342	0.335	0.347	0.364	0.373	0.381	0.382
RHSPDSVA	0.170	0.172	0.173	0.177	0.179	0.181	0.181	0.186	0.188
RHSPSCVA	0.228	0.236	0.237	0.240	0.242	0.245	0.251	0.262	0.271
RITUMSVA	0.053	0.055	0.055	0.056	0.057	0.057	0.059	0.063	0.065

APPL	DENOM19	DENOM20	DENOM21	DENOM22	DENOM23	DENOM24	DENOM25	DENOM26	TOTDENOM
CSCTSCVA	0.234	0.248	0.244	0.248	0.252	0.252	0.246	0.249	5.408
MSISBVA	0.599	0.589	0.583	0.594	0.601	0.601	0.588	0.594	13.536
MSISPVA	0.469	[illegible]	0.475	0.488	0.501	0.501	0.490	0.496	10.697
RHCTSCVA	[illegible]	[illegible]	0.328	0.334	0.340	0.340	0.333	0.336	7.746
RHRRSCVA	[illegible]	[illegible]	0.147	0.149	0.150	0.150	0.147	0.149	3.388
RHRXDSVA	[illegible]	[illegible]	0.583	0.592	0.601	0.601	0.588	0.594	13.295
RHRXSCVA	[illegible]	0.416	0.408	0.417	0.426	0.426	0.417	0.422	8.908
RHSPDSVA	[illegible]	0.190	0.186	0.188	0.190	0.190	0.186	0.188	4.651
RHSPSCVA	[illegible]	0.274	0.268	0.271	0.274	0.274	0.268	0.271	6.246
RITUMSVA	[illegible]	0.066	0.065	0.066	0.066	0.066	0.065	0.066	1.466

Table A-4. Event rate calculation worksheet (events per valve-hour)

NPRDS Code	# of Events	Valve-Hours	Event Rate
CSCTSCVA	1	5.408E6	1.9E-7
MSISBVA	1	13.536E6	7.4E-8
RHCTSCVA	2	7.746E6	2.6E-7
RHRRSCVA	2	3.388E6	5.9E-7
RHRXDSVA	2	13.295E6	1.5E-7
RHRXSCVA	1	8.908E6	1.1E-7
RHSPDSVA	1	4.651E6	3.3E-7
RHSPSCVA	3	6.246E6	4.8E-7
RITUMSVA	1	1.466E6	6.8E-7
Aggregate	14	64.644E6	2.2E-7

Appendix B

Component Failure Rates

CONTENTS

APPENDIX B
COMPONENT FAILURE RATES

This Appendix contains the failu.e rate data used in the B&W reference plant ISLOCA assessments. Two sets of component failure rate information were collected and evaluated for use in the assessments.

The first data set is the result of the sorting and aggregation of LERs (Section B.1.). These data were sorted and counted to estimate the hourly failure rates associated with different components and failure modes. Not all the information obtained from this source is usable (e.g., the hourly failure rate for a valve to close). The deficiencies with respect to generating failure rates with the LER data are well known and understood. Specifically, not all the equipment failures are reported and it is very difficult to obtain the components exposure history. Also it may appear that the LER data base was used to develop demand failures based on hourly information. These LER calculated demand rates were not used in the analysis and are provided only to highlight the make up of the failure modes observed in the data base. However, most of the LER data provide interesting insight into the causes and severity of the hardware failures.

The second set of information (Section B.2.) is a list of applicable failure probability data collected from a variety of published sources. It is used to provided a comparison to the LER generated rates and for furnishing demand-based failure probabilities when needed.

The second set of data provided the preferred source of information used in this assessment. Wherever possible, the data identified as NUCLARR catagory 1 was used since it is the most complete and has the highest statistical confidence of any other available data. Further, this data is based on actual operation experience and generated from plant records.

A review of the data base will show that both BWR and PWR data was used to develop the failure rates. The selection of both sets of data was based on information that both BWRs and PWRs use similar motor operated valves. Although service histories vary, the increased confidence produced by using a larger population of valves prompted the use of the combined LWR failure rates in the analysis.

The remainder of this Appendix is a listing of the data used in the assessment of the ISLOCA frequency of occurrence in the B&W reference plant.

B.1. Aggregated LER Data

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve

FAILURE MODE: Fail to Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	164	338008080	.485195E-6
Pre-1984:	152	174459552	.871262E-6
Post-1983:	12	163548528	.733727E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	7	4
Design Error	3	2
Fabrication/Construction/QC	1	1
Procedural Inadequacy	1	1
Excessive Wear	1	1
Corrosion	3	2
Lack of Lubrication	2	1
Packing Problem	5	3
Mechanical/Control Part Problem	5	3
Seat/Disc Problem	1	1
Bearing/Bushing Problem	1	1
Limit Switch Problem	14	9
Torque Switch Problem	27	16
Electric Motor Operator Problem	41	25
Electrical Input Problem	35	21
Unknown	17	10

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	16	45215424	.353861E-6
CE	High Pressure Safety Injection	7	31556448	.221824E-6
GE	High Pressure Coolant Injection	22	16449552	.133742E-5
GE	Low Pressure Core Spray	15	11885098	.126208E-5
GE	Reactor Core Isolation Cooling	16	14408352	.111046E-5
GE	Reactor Water Cleanup	5	11885088	.420695E-6
GE	Residual Heat Removal	54	35655264	.151450E-5
WE	Emergency Core Cooling Systems	29	134046768	.216342E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve

FAILURE MODE: Fail to Close

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	156	338008080	.461527E-6
Pre-1984:	112	174459552	.641982E-6
Post-1983:	44	163548528	.269033E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
ersonnel (Operations)	6	4
Personnel (Maintenance)	5	3
Personnel (Testing)	1	1
Design Error	4	3
Fabrication/Construction/QC	6	4
Procedural Inadequacy	2	1
Normal Wear	1	1
Excessive Wear	1	1
Corrosion	3	2
Lack of Lubrication	2	1
Foreign Material Contamination	3	2
Packing Problem	4	3
Mechanical/Control Part Problem	9	6
Limit Switch Problem	13	8
Torque Switch Problem	31	20
Electric Motor Operator Problem	22	14
Electrical Input Problem	31	20
Unknown	12	8

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	83	88
Large Le[illegible] (external)	0	0
Small Lea[illegible] (internal)	11	12
Small Leak (external)	0	0

[illegible] ATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	9	45215424	.199047E-6
CE	High Pressure Safety Injection	3	31556448	.950677E-7
CE	Low Pressure Safety Injection	5	29302416	.170634E-6
GE	High Pressure Coolant Injection	21	16449552	.127663E-5
GE	Isolation Condenser	3	1139760	.263213E-5
GE	Low Pressure Core Spray	14	11885088	.117794E-5
GE	Reactor Core Isolation Cooling	25	14408352	.173510E-5
GE	Reactor Water Cleanup	24	11885088	.201933E-5
GE	Residual Heat Removal	37	356552[illegible]4	.103771E-5
WE	Emergency Core Cooling Systems	15	134046768	.111901E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve FAILURE MODE: Internal Leakage

AGGREGATE FAILURE RATE

	# Events	# Hours	Failure Rate
Total:	57	338008080	.168635E-6
Pre-1984:	24	174459552	.137567E-6
Post-1983:	33	163548528	.201774E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	2	4
Personnel (Maintenance)	3	5
Design Error	1	2
Fabrication/Construction/QC	2	4
Corrosion	1	2
Excessive Vibration	1	2
Seat/Disc Problem	24	42
Limit Switch Problem	2	4
Torque Switch Problem	12	21
Electric Motor Operator Problem	2	4
Unknown	7	12

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	1	2
Large Leak (external)	0	0
Small Leak (internal)	48	98
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	45215424	.221163E-7
GE	High Pressure Core Spray	1	1955856	.511285E-6
GE	High Pressure Coolant Injection	5	16449552	.303959E-6
GE	Isolation Condenser	1	1139760	.877377E-6
GE	Low Pressure Core Spray	5	11885088	.420695E-6
GE	Reactor Core Isolation Cooling	5	14408352	.347020E-6
GE	Reactor Water Cleanup	6	11885088	.504834E-6
GE	Residual Heat Removal	30	35655264	.841390E-6
WE	Emergency Core Cooling Systems	3	134046768	.223802E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve FAILURE MODE: External Leakage/Rupture

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	37	338008080	.109464E-6
Pre-1984:	26	174459552	.149031E-6
Post-1983:	11	163548528	.672583E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Corrosion	1	3
Seal/Gasket Problem	4	11
Packing Problem	29	78
Mechanical/Control Part Problem	1	3
Unknown	2	5

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	1	4
Small Leak (internal)	0	0
Small Leak (external)	23	96

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	45215424	.221163E-7
CE	High Pressure Safety Injection	7	31556448	.221824E-6
CE	Low Pressure Safety Injection	2	29302416	.682537E-7
GE	High Pressure Coolant Injection	4	16449552	.243167E-6
GE	Reactor Core Isolation Cooling	4	14408352	.277616E-6
GE	Residual Heat Removal	8	35655264	.224370E-6
WE	Emergency Core Cooling Systems	11	134046768	.820609E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve (Control Valves Only)

FAILURE MODE: Fail to Operate as Required (e.g. fail to control around set point)

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	115	338008080	.340228E-6
Pre-1984:	93	174459552	.533074E-6
Post-1983:	22	163548528	.134516E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	6	5
Personnel (Maintenance)	6	5
Design Error	7	6
Fabrication/Construction/QC	6	5
Procedural Inadequacy	2	2
Mechanical/Control Part Problem	14	12
Limit Switch Problem	8	7
Torque Switch Problem	5	4
Electric Motor Operator Problem	24	21
Electrical Input Problem	26	23
Failure of Component Supply Sys	1	1
Unknown	10	9

LEAK RATE ESTIMATE (1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	19	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	7	45215424	.154814E-6
CE	High Pressure Safety Injection	6	31556448	.190135E-6
CE	Low Pressure Safety Injection	2	29302416	.682537E-7
GE	High Pressure Coolant Injection	15	16449552	.911878E-6
GE	Isolation Condenser	2	1139760	.175475E-5
GE	Low Pressure Core Spray	11	11885088	.925529E-6
GE	Reactor Core Isolation Cooling	14	14408352	.971658E-6
GE	Reactor Water Cleanup	4	11885088	.336556E-6
GE	Residual Heat Removal	38	35655264	.106576E-5
WE	Emergency Core Cooling Systems	16	134046768	.119361E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve

FAILURE MODE: Plugged/Transfer Closed

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	5	338008080	.147925E-7
Pre-1984:	4	174459552	.229279E-7
Post-1983:	1	163548528	.611439E-8

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Procedural Inadequacy	1	20
Electrical Input Problem	4	80

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	1	31556448	.316892E-7
WE	Emergency Core Cooling Systems	4	134046768	.298403E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve

FAILURE MODE: Maintenance/Replacement

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	13	338008080	.384606E-7
Pre-1984:	11	174459552	.630518E-7
Post-1983:	2	163548528	.122287E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	8
Design Error	5	38
Fabrication/Construction/QC	2	15
Procedural Inadequacy	1	8
Excessive Vibration	1	8
Torque Switch Problem	1	8
Electric Motor Operator Problem	1	8
Electrical Input Problem	1	8

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	45215424	.442326E-7
GE	High Pressure Coolant Injection	3	16449552	.182375E-6
GE	Low Pressure Core Spray	1	11885088	.841390E-7
GE	Reactor Core Isolation Cooling	4	14408352	.277616E-6
GE	Reactor Water Cleanup	1	11885088	.841390E-7
GE	Residual Heat Removal	2	35655264	.560926E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve

FAILURE MODE: Test Not Performed

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	2	338008080	.591701E-8
Pre-1984:	2	174459552	.114639E-7
Post-1983:	0	163548528	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	50
Personnel (Maintenance)	1	50

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
WE	Emergency Core Cooling Systems	2	134046768	.149201E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve

FAILURE MODE: Improper Valve Configuration

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	19	338008080	.562116E-7
Pre-1984:	10	174459552	.573198E-7
Post-1983:	9	163548528	.550295E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	12	63
Personnel (Maintenance)	1	5
Design Erro.	1	5
Procedural Inadequacy	5	26

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	6	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	45215424	.221163E-7
CE	High Pressure Safety Injection	1	31556446	.316892E-7
CE	Low Pressure Safety Injection	1	29302416	.341268E-7
GE	Reactor Core Isolation Cooling	4	14408352	.277618E-6
GE	Reactor Water Cleanup	4	11885088	.336558E-6
GE	Residual Heat Removal	2	35655264	.560926E-7
WE	Emergency Core Cooling Systems	6	134046768	.447604E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Motor-Operated Valve FAILURE MODE: Transfer Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	15	338008080	.443776E-7
Pre-1984:	0	174459552	.000000E+0
Post-1983:	15	163548528	.917158E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	8	53
Personnel (Testing)	3	20
Electrical Input Problem	1	7
Unknown	3	20

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	11	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	45215424	.221163E-7
GE	High Pressure Coolant Injection	3	16449552	.182375E-6
GE	Reactor Core Isolation Cooling	1	14408352	.694041E-7
WE	Emergency Core Cooling Systems	10	134046768	.746008E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve FAILURE MODE: Fail to Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	14	174900288	.800456E-7
Pre-1984:	9	88671312	.101498E-6
Post-1983:	5	86228976	.579851E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	7
Leaking/Ruptured Diaphragm	1	7
Mechanical/Control Part Problem	2	14
Limit Switch Problem	1	7
Failure of Component Supply Sys	7	50
Unknown	2	14

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	7863552	.254337E-6
CE	High Pressure Safety Injection	1	13524192	.739415E-7
CE	Low Pressure Safety Injection	1	2254032	.443649E-6
GE	Reactor Water Cleanup	8	59425440	.134622E-6
WE	Emergency Core Cooling Systems	2	38350176	.521509E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve   FAILURE MODE: Fail to Close

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	38	174900288	.217266E-6
Pre-1984:	30	88671312	.338328E-6
Post-1983:	8	86228976	.927762E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Fabrication/Construction/QC	1	3
Lack of Lubrication	4	11
Foreign Material Contamination	5	13
Packing Problem	1	3
Leaking/Ruptured Diaphragm	1	3
Mechanical/Control Part Problem	6	16
Solenoid Problem	1	3
Electrical Input Problem	2	5
Failure of Component Supply Sys	15	39
Unknown	2	5

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	16	84
Large Leak (external)	0	0
Small Leak (internal)	3	16
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	4	13524192	.295766E-6
GE	Reactor Water Cleanup	16	59425440	.269244E-6
WE	Emergency Core Cooling Systems	15	38350176	.391132E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve

FAILURE MODE: Internal Leakage

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	46	174900288	.263006E-6
Pre-1984:	33	88671312	.372160E-6
Post-1983:	13	86228976	.150761E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Fabrication/Construction/QC	2	4
Procedural Inadequacy	1	2
Corrosion	2	4
Foreign Material Contamination	3	7
Mechanical/Control Part Problem	3	7
Seat/Disc Problem	27	59
Unknown	8	17

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	1	6
Large Leak (external)	0	0
Small Leak (internal)	15	94
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	7863552	.254337E-6
CE	High Pressure Safety Injection	3	13524192	.221824E-6
GE	Reactor Water Cleanup	8	59425440	.134622E-6
WE	Emergency Core Cooling Systems	28	38350176	.730113E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve FAILURE MODE: External Leakage/Rupture

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	13	174900288	.743280E-7
Pre-1984:	7	88671312	.789432E-7
Post-1983:	6	86228976	.695821E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Procedural Inadequacy	1	8
Seal/Gasket Problem	1	8
Packing Problem	9	69
Leaking/Ruptured Diaphragm	2	15

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	9	100

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Residual Heat Removal	2	53482893	.373951E-7
WE	Emergency Core Cooling Systems	6	38350176	.156452E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve
(Control Valves Only)

FAILURE MODE: Fail to Operate as Required
(e.g. fail to control around set point)

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	19	174900288	.108633E-6
Pre-1984:	14	88671312	.157886E-6
Post-1983:	5	86228976	.579851E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	5
Design Error	1	5
Foreign Material Contamination	4	21
Mechanical/Control Part Problem	4	21
Pilot Valve Problem	1	5
Electrical Input Problem	1	5
Failure of Component Supply Sys	3	16
Unknown	4	21

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	4	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	7863552	.127168E-6
CE	High Pressure Safety Injection	5	13524192	.369707E-6
GE	Reactor Water Cleanup	9	59425440	.151450E-6
WE	Emergency Core Cooling Systems	4	38350176	.104301E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve

FAILURE MODE: Plugged/Transfer Closed

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	4	174900288	.228701E-7
Pre-1984:	3	88671312	.338328E-7
Post-1983:	1	86228976	.115970E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Leaking/Ruptured Diaphragm	2	50
Failure of Component Supply Sys	2	50

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	7863552	.254337E-6
GE	Reactor Water Cleanup	1	59425440	.168278E-7
WE	Emergency Core Cooling Systems	1	38350176	.260754E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve

FAILURE MODE: Maintenance/Replacement

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	5	174900288	.285877E-7
Pre-1984:	5	88671312	.563880E-7
Post-1983:	0	86228976	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	2	40
Design Error	3	60

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	7863552	.254337E-6
CE	High Pressure Safety Injection	3	13524192	.221824E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve

FAILURE MODE: Improper Valve Configuration

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	2	174900288	.114350E-7
Pre-1984:	1	88671312	.112776E-7
Post-1983:	1	86228976	.115970E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	50
Unknown	1	50

LEAK RATE ESTIMATE
(1980 - 1989)

Leak Size	# Events	% of Total
Large Leak (internal)	[illegible]	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	1	13524192	.739415E-7
GE	Reactor Water Cleanup	1	59425440	.168278E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Air Operated Valve

FAILURE MODE: Transfer Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	1	174900288	.571754E-8
Pre-1984:	1	88671312	.112776E-7
Post-1983:	0	86228976	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	1	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
WE	Emergency Core Cooling Systems	1	38350176	.260754E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve FAILURE MODE: Fail to Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	9	14139120	.636531E-6
Pre-1984:	9	7273680	.123733E-5
Post-1983:	0	6865440	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	11
Mechanical/Control Part Problem	1	11
Solenoid Problem	6	67
Unknown	1	11

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Reactor Water Cleanup	8	11885088	.673112E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve

FAILURE MODE: Fail to Close

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	19	14139120	.134378E-5
Pre-1984:	18	7273680	.247467E-5
Post-1983:	1	6865440	.145657E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Corrosion	1	5
Foreign Material Contamination	4	21
Excessive Vibration	4	21
Mechanical/Control Part Problem	4	21
Limit Switch Problem	5	26
Solenoid Problem	1	5

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	11	92
Large Leak (external)	0	0
Small Leak (internal)	1	8
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	1	2254032	.443649E-6
GE	Reactor Water Cleanup	5	11885088	.420695E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve

FAILURE MODE: Internal Leakage

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	6	14139120	.424354E-6
Pre-1984:	3	7273680	.412445E-6
Post-1983:	3	6865440	.436971E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	2	33
Foreign Material Contamination	1	17
Seat/Disc Problem	1	17
Unknown	2	33

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	3	100
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	1	2254032	.443645E-6
GE	Reactor Water Cleanup	3	11885088	.252417E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve

FAILURE MODE: Fail to Operate as Required

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	9	14139120	.636531E-6
Pre-1984:	9	7273680	.123733E-5
Post-1983:	0	6865440	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Foreign Material Contamination	3	33
Leaking/Ruptured Diaphragm	1	11
Solenoid Problem	1	11
Electrical Input Problem	4	44

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	4	2254032	.177459E-5
GE	Reactor Water Cleanup	4	11885088	.336556E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve FAILURE MODE: Plugged/Transfer Closed

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	16	14139120	.113161E-5
Pre-1984:	16	7273680	.219971E-5
Post-1983:	0	6865440	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Solenoid Problem	4	25
Electrical Input Problem	12	75

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Reactor Water Cleanup	16	11885088	.134622E-5

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve

FAILURE MODE: Maintenance/Replacement

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	1	14139120	.707257E-7
Pre-1984:	1	7273680	.137481E-6
Post-1983:	0	6865440	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Fabrication/Construction/QC	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Reactor Water Cleanup	1	11885088	.841390E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve

FAILURE MODE: Test Not Performed

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	1	14139120	.70/257E-7
Pre-1984:	1	7273680	.137481E-6
Post-1983:	0	6865440	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Solenoid Operated Valve

FAILURE MODE: Improper Valve Configuration

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	1	14139120	.707257E-7
Pre-1984:	1	7273680	.137481E-6
Post-1983:	0	6865440	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Reactor Water Cleanup	1	11885088	.841390E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve

FAILURE MODE: Fail to Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	9	276433056	.325576E-7
Pre-1984:	8	142218144	.562516E-7
Post-1983:	1	134214912	.745073E-8

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	11
Foreign Material Contamination	4	44
Mechanical/Control Part Problem	1	11
Unknown	3	33

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	43249536	.462432E-7
CE	High Pressure Safety Injection	1	39445560	.253513E-7
GE	High Pressure Coolant Injection	1	7049808	.141847E-6
GE	Reactor Core Isolation Cooling	2	12006960	.166570E-6
GE	Reactor Water Cleanup	2	23770176	.841390E-7
WE	Emergency Core Cooling Systems	1	112223856	.891076E-8

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve　　　　FAILURE MODE: Fail to Close

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	30	276433056	.108525E-6
Pre-1984:	25	142218144	.175786E-6
Post-1983:	5	134214912	.372536E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	2	7
Design Error	4	13
Fabrication/Construction/QC	4	13
Foreign Material Contamination	2	7
Mechanical/Control Part Problem	5	17
Seat/Disc Problem	3	10
Limit Switch Problem	2	7
Unknown	8	27

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	17	89
Large Leak (external)	0	0
Small Leak (internal)	2	11
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	3	43249536	.693649E-7
CE	High Pressure Safety Injection	3	39445560	.760541E-7
GE	High Pressure Core Spray	2	1117632	.178949E-5
GE	High Pressure Coolant Injection	2	7049808	.283695E-6
GE	Reactor Core Isolation Cooling	2	12006960	.166570E-6
GE	Reactor Water Cleanup	2	23770176	.841390E-7
GE	Residual Heat Removal	7	14856360	.471178E-6
WE	Emergency Core Cooling Systems	9	112223856	.801968E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve　　　　FAILURE MODE: Internal Leakage

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	4	276433056	.144700E-7
Pre-1984:	2	142218144	.140629E-7
Post-1983:	2	134214912	.149014E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Packing Problem	1	25
Mechanical/Control Part Problem	1	25
Seat/Disc Problem	1	25
Unknown	1	25

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	4	100
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	1	39445560	.253513E-7
GE	Reactor Water Cleanup	1	23770176	.420695E-7
GE	Residual Heat Removal	1	14856360	.673112E-7
WE	Emergency Core Cooling Systems	1	112223856	.891076E-8

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve

FAILURE MODE: External Leakage/Rupture

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	17	276433056	.614977E-7
Pre-1984:	9	142218144	.632830E-7
Post-1983:	8	134214912	.596058E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	6
Seal/Gasket Problem	8	47
Packing Problem	5	29
Weld Failure	1	6
Unknown	2	12

LEAK RATE ESTIMATE
(1980 - 1988)

k Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	13	100

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	43249536	.231216E-7
CE	High Pressure Safety Injection	4	39445560	.101405E-6
GE	Reactor Core Isolation Cooling	1	12006960	.832850E-7
GE	Reactor Water Cleanup	1	23770176	.420695E-7
GE	Residual Heat Removal	2	14856360	.134622E-6
WE	Emergency Core Cooling Systems	8	112223856	.712860E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve

FAILURE MODE: Reverse Leakage (Check Valves)

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	161	276433056	.582419E-6
Pre-1984:	111	142218144	.780491E-6
Post-1983:	50	134214912	.372536E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	1
Design Error	10	6
Fabrication/Construction/QC	1	1
Procedural Inadequacy	1	1
Excessive Wear	4	2
Corrosion	3	2
Foreign Material Contamination	14	9
Seal/Gasket Problem	3	2
Mechanical/Control Part Problem	3	2
Seat/Disc Problem	75	47
Unknown	46	29

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	3	3
Large Leak (external)	0	0
Small Leak (internal)	110	96
Small Leak (external)	1	1

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	9	43249536	.208094E-6
CE	High Pressure Safety Injection	19	39445560	.481676E-6
CE	Core Flood Accumulators	47	4508064	.104257E-4
CE	Low Pressure Safety Injection	4	13524192	.295766E-6
GE	High Pressure Core Spray	6	1117632	.536849E-5
GE	High Pressure Coolant Injection	10	7049808	.141847E-5
GE	Low Pressure Core Spray	2	2971272	.673112E-6
GE	Reactor Core Isolation Cooling	5	12006960	.416425E-6
GE	Reactor Water Cleanup	6	23770176	.252417E-6
GE	Residual Heat Removal	22	14856360	.148084E-5
WE	Emergency Core Cooling Systems	31	112223856	.276233E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve

FAILURE MODE: Fail to Operate as Required

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	5	276433056	.180875E-7
Pre-1984:	5	142218144	.351572E-7
Post-1983:	0	134214912	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Design Error	1	20
Corrosion	2	40
Mechanical/Control Part Problem	1	20
Seat/Disc Problem	1	20

LEAK RATE ESTIMATE
(1980 - 1980)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	43249536	.462432E-7
GE	High Pressure Coolant Injection	1	7049808	.141847E-6
GE	Reactor Core Isolation Cooling	1	12006960	.832850E-7
GE	Residual Heat Removal	1	14856360	.673112E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve

FAILURE MODE: Plugged/Transfer Closed

AGGREGATE FAILURE "'TES

	# Events	# Hours	Failure Rate
Total:	1	276433056	.361751F-8
Pre-1984:	1	142218144	.703145E-8
Post-1983:	0	134214912	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Foreign Material Contaminat' n	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
WE	Emergency Core Cooling Systems	1	112223856	.891076E-8

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve

FAILURE MODE: Test Not Performed

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	3	276433056	.108525E-7
Pre-1984:	3	142218144	.210943E-7
Post-1983:	0	134214912	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	33
Procedural Inadequacy	2	67

LEAK RATE E [illegible]
(1980 - [illegible])

Leak Size	[illegible]	[illegible]
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	High Pressure Coolant Injection	2	7049808	.283695E-6
WE	Emergency Core Cooling Systems	1	112223856	.891076E-8

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Check Valve FAILURE MODE: Improper Valve Configuration

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	2	276433056	.723502E-8
Pre-1984:	2	142218144	.140629E-7
Post-1983:	0	134214912	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	2	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	43249536	.231216E-7
GE	Reactor Core Isolation Cooling	1	12006960	.832850E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Testable Check Valve | FAILURE MODE: Fail to Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	1	17195232	.581556E-7
Pre-1984:	0	8738880	.000000E+0
Post-1983:	1	8456352	.118254E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Limit Switch Problem	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Residual Heat Removal	1	6221952	.160721E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Testable Check Valve | FAILURE MODE: Fail to Close

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	3	17195232	.174466E-6
Pre-1984:	2	8738880	.228862E-6
Post-1983:	1	8456352	.118254E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	2	67
Mechanical/Control Part Problem	1	33

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	3	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	High Pressure Coolant Injection	1	2349936	.425543E-6
GE	Low Pressure Core Spray	1	2971272	.336556E-6
GE	Residual Heat Removal	1	6221952	.160721E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Testable Check Valve

FAILURE MODE: Internal Leakage

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	2	17195232	.116311E-6
Pre-1984:	0	8738880	.000000E+0
Post-1983:	2	8456352	.236508E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Seat/Disc Problem	2	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	2	100
Small Leak (external)	0	0

FAIL'RE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Residual Heat Removal	2	6221952	.321442E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Testable Check Valve

FAILURE MODE: External Leakage/Rupture

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	2	17195232	.116311E-6
Pre-1984:	1	8738880	.114431E-6
Post-1983:	1	8456352	.118254E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Seal/Gasket Problem	1	50
Packing Problem	1	50

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	2	100

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Reactor Core Isolation Cooling	1	2401392	.416425E-6
GE	Residual Heat Removal	1	6221952	.160721E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Testable Check Valve

FAILURE MODE: Improper Valve Configuration

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	2	17195232	.116311E-6
Pre-1984:	2	8738880	.228862E-6
Post-1983:	0	8456352	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	2	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	High Pressure Coolant Injection	1	2349936	.425543E-6
GE	Reactor Core Isolation Cooling	1	2401392	.416425E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Testable Check Valve　　　　FAILURE MODE: Transfer Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	1	17195232	.581556E-7
Pre-1984:	0	8738880	.000000E+0
Post-1983:	1	8456352	.118254E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Operations)	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	1	100
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Low Pressure Core Spray	1	2971272	.336556E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Relief/Safety Valve FAILURE MODE: Fail to Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	2	76599936	.261096E-7
Pre-1984:	2	39126864	.511157E-7
Post-1983:	0	37473072	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Foreign Material Contamination	1	50
Unknown	1	50

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
WE	Emergency Core Cooling Systems	2	10276704	.194614E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Relief/Safety Valve

FAILURE MODE: Fail to Close

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	4	76599936	.522193E-7
Pre-1984:	1	39126864	.255578E-7
Post-1983:	3	37473072	.800574E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Mechanical/Control Part Problem	1	25
Seat/Disc Problem	2	50
Unknown	1	25

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	2	67
Large Leak (external)	0	0
Small Leak (internal)	1	33
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
GE	Reactor Water Cleanup	4	17827632	.224370E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Relief/Safety Valve FAILURE MODE: Internal Leakage

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	25	76599936	.326371E-6
Pre-1984:	10	39126864	.255578E-6
Post-1983:	15	37473072	.400287E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Fabrication/Construction/QC	1	4
Lack of Lubrication	2	8
Foreign Material Contamination	2	8
Mechanical/Control Part Problem	1	4
Seat/Disc Problem	10	40
Unknown	9	36

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	17	100
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	7	11270160	.621109E-6
GE	Reactor Core Isolation Cooling	1	2401392	.416425E-6
GE	Reactor Water Cleanup	2	17827632	.112185E-6
GE	Residual Heat Removal	9	17827632	.504834E-6
WE	Emergency Core Cooling Systems	3	10276704	.291922E-6

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Relief/Safety Valve

FAILURE MODE: External Leakage/Rupture

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	10	76599936	.130548E-6
Pre-1984:	7	39126864	.178905E-6
Post-1983:	3	37473072	.800574E-7

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	10
Design Error	1	10
Excessive Vibration	3	30
Seal/Gasket Problem	1	10
Bellows/Boot Problem	4	40

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	4	100

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	6	11270160	.532379E-6
GE	High Pressure Core Spray	3	558816	.536849E-5
WE	Emergency Core Cooling Systems	1	10276704	.973074E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Relief/Safety Valve

FAILURE MODE: Fail to Operate as Required

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	4	76599936	.522193E-7
Pre-1984:	4	39126864	.102231E-6
Post-1983:	0	37473072	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Unknown	4	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	1	11270160	.887298E-7
WE	Emergency Core Cooling Systems	1	10276704	.973074E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Relief/Safety Valve   AILURE MODE: Premature Open

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	20	76599936	.261093E-6
Pre-1984:	13	39126864	.332257E-6
Post-1983:	7	37473072	.186800E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	5
Design Error	1	5
Fabrication/Construction/QC	2	10
Procedural Inadequacy	1	5
Foreign Material Contamination	1	5
Mechanical/Control Part Problem	1	5
Seat/Disc Problem	2	10
Unknown	11	55

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	5	56
Large Leak (external)	0	0
Small Leak (internal)	4	44
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	2	3931776	.508675E-6
CE	High Pressure Safety Injection	5	11270160	.443649E-6
CE	Low Pressure Safety Injection	3	9016128	.332737E-6
GE	Reactor Water Cleanup	3	17827632	.168278E-6
GE	Residual Heat Removal	1	17827632	.560926E-7
WE	Emergency Core Cooling Systems	5	10270704	.486537E-6

ISLOCA VALVE FAILURE ANALYSIS
============================

COMPONENT: Relief/Safety Valve — FAILURE MODE: Maintenance/Replacement

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	1	76599936	.130548E-7
Pre-1984:	1	39126864	.255578E-7
Post-1983:	0	37473072	.000000E+0

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Design Error	1	100

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	0	0
Large Leak (external)	0	0
Small Leak (internal)	0	0
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
CE	High Pressure Safety Injection	1	11270160	.887298E-7

ISLOCA VALVE FAILURE ANALYSIS

COMPONENT: Relief/Safety Valve

FAILURE MODE: Fail to Reseat (Relief Valve)

AGGREGATE FAILURE RATES

	# Events	# Hours	Failure Rate
Total:	22	76599936	.287206E-6
Pre-1984:	9	39126864	.230020E-6
Post-1983:	13	37473072	.346915E-6

FAILURE CAUSE BREAKDOWN

Description	# Events	% of Total
Personnel (Maintenance)	1	5
Excessive Wear	3	14
Mechanical/Control Part Problem	1	5
Seat/Disc Problem	3	14
Unknown	14	64

LEAK RATE ESTIMATE
(1980 - 1988)

Leak Size	# Events	% of Total
Large Leak (internal)	12	55
Large Leak (external)	0	0
Small Leak (internal)	10	45
Small Leak (external)	0	0

FAILURE RATES BY SYSTEM

Vendor	System	# Events	# Hours	Failure Rate
BW	Emergency Core Cooling Systems	1	3931776	.254337E-6
CE	High Pressure Safety Injection	8	11270160	.709839E-6
GE	Reactor Water Cleanup	5	17827632	.280463E-6
WE	Emergency Core Cooling Systems	8	10276704	.778459E-6

B.2. Data From Generic Sources

The failure rates listed herein were gleaned from a report by Eide, et. al.[B-1] and IAEA-TECDOC-508.[B-2] The failure rates reported in both documents were extracted from other well documented sources. Also included herein, is any significant information reportedly used in deriving the failure rates.

Codes used in determining and reporting the failure rate information include:

NUCLARR CATEGORY 1 (N1) = 19 PRA data sources
NUCLARR CATEGORY 2 (N2) = LER and IPRDS data sources
NUCLARR CATEGORY 3 (N3) = IEEE and NPRD-3 data sources
* = estimated values (see equation attached)
\# = only the median value was reported; in all other cases either the mean or mean and median were reported
$ = calculated value (see equation attached)
/D = per demand
/H = per hour
/CY = per cycle
W/O = without
CKV = check valve
EF = error factor
RF = range factor
LB = lower/minimum bound
SBO = station blackout
UB = upper/maximum bound.

Equations used:

Where the mean, median, or error factor was not reported in the referenced documents, they were calculated using the following:

Error Factor = (95% Upper Bound)/Median
Mean/Median = exp[(ln(Error Factor)/1.6449)^2)/2]
1.6449 = constant for 90% confidence interval (95 & 5% bounds)
Estimate of Error Factor = square root of Upper Bound/Lower Bound

B.2.1. Failure Rates for Mechanical Components

CHECK VALVE FAILURE RATES - SELF OPERATING CHECK VALVE
FAIL TO CLOSE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	1.0E-03 /D (CATEGORY 1)	$6.2E-04 /D	EF = 5	B-1, P.12
	2.8E-05 /D	$2.2E-05 /D	UB = 7.1E-05 /D EF = 3.2	B-4
ASEP	1.0E-03 /D	$8.0E-04 /D	EF = 3	B-1, P.12
IREP/NREP	2.0E-06 /H	$1.4E-06 /H	EF = 4.1	B-1, P.12
IREP	1.0E-03 /D	1.0E-03 /D	$UB(95%) = 3.0E-03 /D EF = 3	B-3, P.126
	3.0E-06 /H	1.0E-6 /H (BASED ON 1 ACTUATION PER MONTH)	EF = 10 $UB = 1.0E-05 /H	B-3, P.126
NUREG-2728 (IREP)	1.0E-03 /D	$8.0E-04 /D	EF = 3	B-2, P.119
	3.0E-06 /H (1 ACTUATION PER MONTH)	$1.1E-06 /H	EF = 10	B-2, P.119
SEABROOK PRA	2.7E-04 /D	$1.8E-04 /D	EF = 4.4	B-1, P.12
NUREG-4550	1.0E-03 /D	$8.0E-04 /D	EF = 3	B-2, P.119
SHOREHAM PRA BWR (TESTABLE CKV; ASSUMES MONTHLY TESTING)	7.9E-04 /D	--	--	B-2, P.119
		5.8E-04 /D (ASSUMES MONTHLY TESTING)	--	B-2, P.120
NUREG-2815	7.2E-04 /D (ASSUMES MONTHLY TESTING)	*5.0E-04 /D	UB = 3.6E-03 /D LB = 2.2E-04 /D *EF = 4	B-2, P. 120
OCONEE NPP PRA TILTING DISK TYPE VALVE	1.3E-04 /D (UPDATED)	*1.0E-04 /D	UB(95%) = 2.7E-04 /D LB(5%) = 3.0E-05 /D *EF = 3	B-2, P.123
SWING TYPE VALVE	9.8E-05 /D	*7.6E-05 /D	UB(95%) = 2.1E-04 /D LB(5%) = 2.0E-05 /D *EF = 3.2	B-2, P.123
ZION NPP PRA	8.4E-07 /D (UPDATED; INCLUDES INTERNAL/REVERSE LEAKAGE)	--	--	B-2, P.124
ISLOCA - PWR	2.8E-04 /D (BASED ON LPI AND HPI VALVE EXPERIENCE)	$1.7E-04 /D	RF = 5	B-6, P.A-8

CHECK VALVE FAILURE RATES - SELF OPERATING CHECK VALVE
FAIL TO OPEN

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	5.0E-05 /D (CATEGORY 1)	$3.1E-05 /D	EF = 5	B-1, P.12
ASEP	1.0E-04 /D	$8.0E-05 /D	EF = 3	B-1, P.12
IREP/NREP	2.0E-07 /H	$1.5E-07 /H	EF = 3.5	B-1, P.12
IEEE-500	6.0E-05 /D	--	--	B-1, P.12
SEABROOK PRA	2.7E-04 /D	$1.8E-04 /D	EF = 4.4	B-1, P.12
WASH-1400	1.2E-04 /D	$9.6E-05 /D	EF = 3	B-1, P.12
	1.2E-04 /D	#1.0E-04 /D	EF = 3 UB(95%) = 3.0E-04 /D LB(5%) = 3.0E-05 /D	B-2, P.113
NUREG-4550	1.0E-04 /D (DEVELOPED IN SBO STUDY)	$8.0E-05 /D	EF = 3	B-2, P.113
NUREG-2815	7.2E-05 /D (ASSUMES MONTHLY TESTING)	*5.4E-05 /D	UB = 3.6E-04 /D LB = 2.9E-05 /D *EF = 3.5	B-2, P.114
NUREG-2729	1.0E-04 /D	$8.0E-05 /D	EF = 3	B-2, P.113
	3.0E-07 /H	$1.1E-07 /H (1 ACTUATION PER MONTH)	EF = 10	B-2, P.113
SHOREHAM PRA BWR	7.9E-05 /D (TESTABLE CK; ASSUMES MONTHLY TESTING)	--	--	B-2, P.114
	5.4E-05 /D (ASSUMES MONTHLY TESTING)	--	--	B-2, P.114
NUREG-1363	6.4E-05 /D	*5.0E-05 /D	[illegible]95%) = 1.7E-04 /D [illegible]5%) = 1.7E-05 /D *E[illegible] = 3.2	B-2, P.114
	3.0E-08 /H	*2.3E-08 /H		B-2, P.114
OCONEE NPP PRA SWING TYPE VALVE	9.8E-05 /D (UPDA[illegible]	*7.6E-05 /D	UB(95%) = 2.1E-04 /D LB(5%) = 2.0E-05 /D *EF = 3.2	[illegible]-2, P.117
TILTING DISK TYPE VALVE	8.7E-05 /D (UPDATED)	*7.0E-05 /D	UB(95%) = 1.7E-04 /D LB(5%) = 1.9E-05 /D *EF = 3	B-2, P.117
ZION NPP PRA	4.3E-05 /D	--	--	B-2, P.117
IREP	1.0E-04 /D	1.0E-04 /D	EF = 3 $UB(95%) = 3.0E-04 /D	B-3, P.126
	3.0E-07 /H	1.0E-07 /H (BASED ON 1 ACTUATION PER MONTH)	EF = 10 $UB(95%) = 1.0E-06 /H	B-3, P.126

CHECK VALVE FAILURE RATES - SELF OPERATING CHECK VALVE
INTERNAL LEAKAGE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	3.0E-06 /H (CATEGORY 1)	$1.1E-06 /H	EF = 10	B-1, P.12
	2.2E-06 /H	$9.3E-07 /H	UB = 7.9E-06 /H EF = 8.5	B-4
IREP/NREP	3.0E-06 /H (CATASTROPHIC)	$6.3E-07 /H	EF = 18.3	B-1, P.12
	1.0E-07 /H (MINOR LEAK)	$2.7E-09 /H	EF = 83.7	B-1, P.12
IREP	5.0E-07 /H (CATASTROPHIC)	1.0E-08 /H	$UB = 1.0E-06 /H EF = 100	B-3, P.126
	3.0E-05 /H (MINOR)	1.0E-06 /H	EF = 10 $UB(95%) = 1.0E-05 /H	B-3, P.126
IEEE-500	5.0E-07 /H	--	--	B-1, P.12
SEABROOK PRA	5.4E-07 /H	$3.8E-07 /H	EF = 4	B-1, P.12
WASH-1400	3.8E-07 /H	$3.0E-07 /H	EF = 3 UB = 9.0E-07 /H	B-1, P.12
WASH-1400	2.7E-08 /H	1.0E-08 /H	UB = 1.0E-07 /H EF = 10	B-7
ISLOCA - BWR	3.4E-07 /H	--	--	B-5, P.C-26
ISLOCA - PWR	8.7E-08 /H (LEAK RATE = 200 GPM - ACCUMULATORS & LPI SYSTEMS)	--	--	B-6, P.A-20
LER DATA	1.4E-08 /H	--	--	B-8
	5.6E-07 /H (REVERSE LEAK)	--	--	

CHECK VALVE FAILURE RATES - SELF OPERATING CHECK VALVE
EXTERNAL LEAKAGE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N2)	5.0E-08 /H	$1.9E-08 /H	EF = 10	B-1, P.12
IEEE-500	5.0E-08 /H	--	--	B-1, P.12
ISLOCA - BWR	1.0E-07 /H	--	--	B-5, P.C-26
LER DATA	6.1E-08 /H	--	--	B-8

CHECK VALVE FAILURE RATES - SELF OPERATING CHECK VALVE
EXTERNAL RUPTURE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
WASH-1400	2.7E-08 /H	$1.0E-08 /H	EF = 10 UB = 1.0E-07 /D	B-1, P.12

SAFETY/RELIEF VALVE FAILURE RATES
FAIL TO OPEN

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
E[illegible]/RECOMMENDED (NUCLARR (N1)				
RELIEF VALVE (CATEGORY 1)	3.0E-03 /D	$1.9E-03 /D	EF = 5	B-1, P.12
ASEP				
SAFETY VALVE	1.0E-05 /D	$8.0E-06 /D	EF = 3	B-1, P.12
RELIEF VALVE (PORV)	1.0E-05 /D	$8.0E-06 /D	EF = 3	B-1, P.12
IREP/NREP				
SAFETY VALVE (PRIMARY SAFETY VALVE)	2.0E-05 /H	$1.2E-05 /H	EF = 5	B-1, P.12
SAFETY VALVE (CODE SAFETY VALVE)	6.0E-07 /H	$4.4E-07 /H	EF = 3.7	B-1, P.12
IEEE-500				
SAFETY VALVE	4.0E-03 /D	--	--	B-1, P.12
RELIEF VALVE	1.0E-02 /D	--	--	B-1, P.12
IEEE-500				
SAFETY VALVE	4.0E-03 /CY	--	--	B-2, P.137
EIDE/RECOMMENDED (SEABROOK PRA)				
SAFETY VALVE	3.0E-04 /D	$1.9E-04 /D	EF = 5	B-1, P.12
RELIEF VALVE	3.4E-05 /D	$1.3E-05 /D	EF = 9.5	B-1, P.12
RELIEF VALVE (PORV)	4.3E-03 /D	$3.4E-03 /D	EF = 3	B-1, P.12
WASH-1400				
RELIEF VALVE	1.2E-05 /D	$9.6E-06 /D	EF = 3	B-1, P.12
WASH-1400				
RELIEF VALVE	1.2E-05 /D	#1.0E-05 /D	EF = 3 UB(95%) = 3.0E-05 /D LB(5%) = 3.0E-06 /D	B-2, P.129
NUREG-1363				
RELIEF VALVE	8.9E-03 /D (BWR ONLY)	*8.8E-03 /D	UB(95%) = 1.1E-02 /D LB(5%) = 6.8E-03 /D *EF = 1.3	B-2, P.129
RELIEF VALVE	8.7E-06 /H (BWR ONLY)	*8.6E-06 /H		B-2, P.129
RELIEF VALVE	4.9E-03 /D (BWR ONLY) (UPDATED) (PORV)	*2.4E-03 /D	UB(95%) = 1.1E-02 /D LB(5%) = 2.1E-04 /D *EF = 7.2	B-2, P.129
RELIEF VALVE	3.1E-03 /D (W/O COMMAND)	*3.0E-03 /D	UB(95%) = 4.7E-03 /D LB(5%) = 2.1E-03 /D *EF = 1.5	B-2, P.133
RELIEF VALVE	3.0E-06 /H (W/O COMMAND)	*2.9E-06 /H		B-2, P.133
RELIEF VALVE	3.2E-03 /D (WITH COMMAND)	*3.1E-03 /D		B-2, P.133
RELIEF VALVE	3.2E-06 /H (WITH COMMAND)	*3.1E-06 /H		B-2, P.133
SAFETY VALVE	3.9E-03 /D	*3.6E-03 /D	UB(95%) = 7.4E-03 /D LB(5%) = 1.8E-03 /D *EF = 2	B-2, P.137
SAFETY VALVE	1.7E-06 /H	*1.6E-06 /H		B-2, P.137

NUREG-2728				
RELIEF VALVE	3.0E-04 /D	$1.1E-04 /D	EF = 10	B-2, P.129
RELIEF VALVE	2.0E-02 /D	$1.6E-02 /D	EF = 3	B-2, P.133
SAFETY VALVE	1.0E-05 /D (BWR ONLY) (PRIMARY SAFETY VALVE)	$8.0E-06 /D	EF = 3	B-2, P.138
SAFETY VALVE	1.0E-05 /D (CODE SAFETY VALVE)	$8.0E-06 /D	EF = 3	B-2, P.138
NUREG-4550				
RELIEF VALVE	3.0E-02 /D (PORV)	$1.1E-02 /D	EF = 10	B-2, P.133
OCONEE NPP PRA				
RELIEF VALVE	4.9E-03 /D (PORV)	*2.4E-03 /D	UB(95%) = 1.1E-02 /D LB(5%) = 2.1E-04 /D *EF = 7.2	B-2, P.129
SAFETY VALVE	2.7E-04 /D (UPDATED) (PRESSURIZER (SAFETY VALVE)	*2.7E-04 /D	UB(95%) = 8.0E-04 /D LB(5%) = 7.4E-06 /D *EF = 1	B-2, P.138
NUREG-2815				
SAFETY VALVE	6.5E-03 /D (CODE SAFETY VALVE)	*2.9E-03 /D	UB = 8.6E-02 /D LB = 1.3E-03 /D *EF = 8.1	B-2, P.137
SAFETY VALVE	4.3E-02 /D (BWR: PRIMARY SAFETY VALVE)	*2.7E-02 /D	UB = 4.3E-01 /D LB = 1.7E-02 /D *EF = 5	B-2, P.137
IREP	3.0E-04 /D	1.0E-04 /D	EF = 10 $UB(95%) = 1.0E-03 /D	B-3, P.127

MANUAL VALVE FAILURE RATES
FAIL TO OPEN/CLOSE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1) (CATEGORY 1)	5.0E-04 /D	$1.9E-04 /D	EF = 10	B-1, P.12
IREP/NREP	2.0E-07 /H	$1.5E-07 /H	EF = 3.5	B-1, P.12
NUREG-2815	2.6E-02 /D (1 ACTUATION PER MONTH)	*1.9E-02 /D	UB = 1.3E-01 /D LB = 1.0E-02 /D *EF = 3.6	B-2, P.93
NUREG-2728	1.0E-04 /D 3.0E-07 /H (1 ACTUATION PER MONTH)	$8.0E-05 /D 2.4E-07 /H	EF = 3 --	B-2, P.93 B-2, P.93
IEEE-500	7.0E-05 /CY (PWR)	--	--	B-2, P.93
	6.0E-05 /CY (BWR)	--	--	B-2, P.94
NUREG-1363	6.3E-05 /D (ESF VALVES ONLY)	*5.2E-05 /D	UB(95%) = 1.6E-04 /D LB(5%) = 2.1E-05 /D *EF = 2.8	B-2, P.94
	2.4E-08 /H (ESF VALVES ONLY)	*2.0E-08 /H		B-2, P.94

IREP	1.0E-04 /D	1.0E-04 /D	EF = 3 $UB(95%) = 3.0E-04 /D	B-3, P.126

MANUAL VALVE FAILURE RATES
INTERNAL LEAKAGE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (SEABROOK PRA)	5.0E-08 /H	$1.9E-08 /H	EF = 10	B-1, P.12

MANUAL VALVE FAILURE RATES
EXTERNAL LEAKAGE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N2)	3.0E-08 /H (CATEGORY 2)	$1.1E-08 /H	EF = 10	B-1, P.12

MANUAL VALVE FAILURE RATES
EXTERNAL RUPTURE

REFERENCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
WASH-1400	2.7E-08 /H	$1.0E-08 /H	EF = 10	B-1, P.12

Motor-Operated Valve FAILURE RATES
FAIL TO OPEN/CLOSE/FAIL TO OPERATE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	3.0E-03 /D (CATEGORY 1)	$1.9E-03 /D	EF = 5	B-1, P.12
	4.4E-03 /D	$3.2E-03 /D	UB = 1.2E-02 /D EF = 3.8	F-1
ASEP	3.0E-03 /D	$2.4E-03 /D	EF = 3	B-1, P.12
IREP/NREP	1.0E-05 /H	$2.4E-06 /H	EF = 15.8	B-1, P.12
IREP	3.0E-03 /D	$1.0E-03 /D	$UB = 1.0E-02 /D EF = 10	B-3, P.126
IEEE-500	6.0E-03 /D	--	--	B-1, P.12
IEEE-500	4.0E-03 /CY (PWR ONLY)	--	--	B-2, P.102
	8.0E-03 /CY (BWR ONLY)	--	-	B-2, P.101
SEABROOK PRA	4.3E-03 /D	$3.[illegible]03 /D	EF = 3.7	B-1, P.12
WASH-1400	1.2E-03 /D	$9.6E-04 /D	EF = 3	B-1, P.12
WASH-1400	1.0E-03 /D	$8.0E-04 /D	EF = 3 UB(95%) = 3.0E-03 /D LB(5%) = 3.0E-04 /D	B-2, P.103
NUREG-1363	4.1E-03 /D (PWR ESF VALVES ONLY) (W/O COMMAND)	*4.1E-03 /D	UB(95%) = 4.9E-03 /D LB(5%) = 3.4E-03 /D *EF = 1.2	B-2, P.102

	6.2E-03 /D (PWR ESF VALVES ONLY) (WITH COMMAND)	*6.2E-03 /D		B-2, P.102
	1.9E-06 /H (PWR ESF VALVES ONLY)	*1.9E-06 /H		B-2, P.102
	6.8E-03 /D (BWR ESF VALVES ONLY) (W/O COMMAND)	*6.8E-03 /D	UB(95%) = 7.4E-03 /D LB(5%) = 6.2E-03 /D *EF = 1.1	B-2, P.101
	3.1E-06 /H (BWR ESF VALVES ONLY) (W/O COMMAND)	*3.1E-06 /H		B-2, P.101
	9.6E-03 /D (BWR ESF VALVES ONLY) (WITH COMMAND)	*9.6E-03 /D		B-2, P.101
	4.4E-06 /H (BWR ESF VALVES ONLY) (WITH COMMAND)	*4.4E-06 /H		B-2, P.101
NUREG-4550	3.0E-03 /D (FROM SBO STUDY) (INCLUDES HARDWARE FAULTS; 5.0E-04) (INCLUDES CIRCUIT FAULTS; 2.5E-04)	$1.1E-03 /D	EF = 10	B-2, P.103
OCONEE NPP PRA	1.0E-01 /D (UPDATED) (30 DEMANDS, 5 FAILURES)	*8.7E-02 /D	UB(95%) = 1.6E-01 /D LB(5%) = 2.7E-02 /D *EF = 2.4	B-2, P.105
	6.4E-03 /D (UPDATED) (6725 DEMANDS, 42 FAILURES)	*6.3E-03 /D	UB(95%) = 7.7E-03 /D LB(5%) = 4.5E-03 /D *EF = 1.3	B-2, P.105
ZION NPP PRA	5.7E-03 /D (1647 DEMANDS, 10 FAILURES)	--	--	B-2, P.105
	3.7E-03 /D (1720 DEMANDS, 7 FAILURES)	--	--	B-2, P.105
	1.6E-03 /D (UPDATED) (11310 DEMANDS, 14 FAILURES)	--	--	B-2, P.106

Motor-Operated Valve FAILURE RATES
FAIL TO CLOSE WHILE INDICATING CLOSED

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
ISLOCA - PWR	1.1E-04 /D	--	--	B-6, P.A-10
ISLOCA - BWR	1.1E-07 /D	--	--	B-5, P.C-26

Motor-Operated Valve FAILURE RATES
TRANSFERS OPEN

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
ISLOCA - PWR	9.2E-08 /H	--	--	B-6, P.A-10
ISLOCA - BWR	3.4E-07 /H	--	--	B-5, P.C-26

Motor-Operated Valve FAILURE RATES
INADVERTENTLY OPENED

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
ISLOCA - BWR	3.4E-07 /H	--	--	B-5, P.C-28

Motor-Operated Valve FAILURE RATES
FAIL TO REMAIN OPEN

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
IREP	1.0E-07 /H	1.0E-07 /H	EF = 3 $UB(95%) = 3.0E-07 /H	B-3, P.126

Motor-Operated Valve FAILURE RATES
SPURIOUS OPERATION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	5.0E-08 /H (CATEGORY 1)	$1.9E-08 /H	EF = 10	B-1, P.12
IREP/NREP	2.0E-07 /H (INCLUDES OTHER FAILURE MODES)	$1.5E-07 /H	EF = 3.5	B-1, P.12
SEABROOK PRA & ISLOCA - PWR	9.2E-08 /H (INCLUDES OTHER FAILURE MODES)	$6.0E-08 /H	EF = 4.6	B-1, P.12 B-6, P.A-10
WASH-1400	3.8E-07 /H (INCLUDES OTHER FAILURE MODES; EXCEPT COMMAND FAULTS)	$3.0E-07 /H	EF = 3	B-1, P.12

Motor-Operated Valve FAILURE RATES
INTERNAL LEAKAGE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
NUCLARR	7.7E-07 /K	$6.0E-07 /H (TRANSFERS OPEN)	UB = 1.9E-06 /H EF = 3.2	B-4
EIDE/RECOMMENDED (SEABROOK PRA)	1.0E-07 /H	$3.8E-08 /H	EF = 10	B-1, P.12
IREP/NREP	1.0E-07 /H (CATASTROPHIC)	$?.7E-09 /H	EF = 83.7	B-1, P.12
IREP	5.0E-07 /H (CATASTROPHIC) 9.3E-08 /H (BWR) (TRANSFERS OPEN)	$ .0E-08 /H --	$UB = 1.0E-06 /H EF = 100 --	B-3, P.126
ISLOCA - PWR	5.5E-07 /H 1.4E-07 /H (PWR) (DISK RUPTURE) 9.3E-08 /H (PWR) (TRANSFERS OPEN)	3.4E-07 /H -- --	RF = 5 -- --	B-6, P.A-10

LER DATA	1.9E-07 /H	--	--	B-8
	4.4E-08 /H (TRANSFERS OPEN)	--	--	
	5.6E-07 /H (IMPROPER CONFIG.)	--	--	
WASH-1400	2.7E-08 /H	$1.0E-08 /H	UB = 1.0E-07 /H EF = 10	B-7

Motor-Operated Valve FAILURE RATES
EXTERNAL LEAKAGE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N2)	1.0E-07 /H (CATEGORY 2)	$3.8E-08 /H	EF = 10	B-1, P.12
IEEE-500	1.0E-07 /H	--	--	B-1, P.12
ISLOCA -PWR	1.0E-07 /H	--	--	B-6, P.A-11
LER DATA	1.1E-07 /H	--	--	B-8

Motor-Operated Valve FAILURE RATES
EXTERNAL RUPTURE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
WASH-1400	2.7E-08 /H	$1.0E-08 /H	EF = 10	B-1, P.12
ISLOCA - BWR	1.4E-07 /H	--	--	B-5, P.C-26

Motor-Operated Valve FAILURE RATES
INTERNAL DISK RUPTURE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
ISLOCA - BWR	1.4E-07 /H	--	--	B-6, P.A-8

B.2.2. Failure Rates for Electrical Components

BISTABLE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
IEEE-500	3E-06 /H 1E-06 /D	$1E-06 $8E-07	EF = 10 EF 3	B-0, P.628 B-9, P.628

PRESSURE SWITCH
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
IREP/NUREG-2728	8.3E-06 /H (ASSUMES 1 DEMAND/DAY)	$5.0E-06	EF = 3	B-2, P.199
WASH-1400	1.0E-05 /H (ASSUMES 1 DEMAND/DAY)	#8.3E-06 /H	UB(95%) = 2.5E-05 /H LB(5%) = 2.5E-06 /H EF = 3	B-2, P.199
IEEE-500	4.0E-07 /H	*1.1E-07 /H	UB = 1.9E-06 /H LB = 1.0E-08 /H *EF = 13.8	B-2, P.201
NUREG-2815	2.0E-07 /H	*1.5E-07 /H	UB = 1.0E-06 /H LB = 8.0E-08 /H *EF = 3.5	B-2, P.201

PRESSURE SWITCH, PROCESS
FAIL TO OPEN/CLOSE/OPERATE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N3)	3.0E-07 /D	$1.1E-07 /D	EF = 10	B-1, P.24
IEEE-500	1.5E-07 /D	$1.2E-07 /D	EF = 3	B-1, P.24
SEABROOK PRA	2.7E-04 /D	--	--	B-1, P.24
IREP	1.0E-04 /D	1.0E-04 /D	EF = 3 $UB(95%) = 3.0E-04 /D	B-3, P.126

MANUAL SWITCH
FAIL TO OPEN/CLOSE/OPERATE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
IREP	3.0E-05 /D	1.0E-05 /D	EF = 10 $UB(95%) = 1.0E-04 /D	B-3, P.127

PRESSURE SWITCH, PROCESS
SPURIOUS OPERATION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED NUCLARR N1)	1.0E-06 /H		EF = 10	B-1, P.24
IEEE-500	1.0E-07 /H	$7.0E-08 /H	EF = 4	B-1, P.24

TEMPERATURE SWITCH
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
SHOREHAM PRA	2.3E-06 /H (BWR)	--	--	B-2, P.200
IEEE-500	2.0E-07 /H	*1.6E-07 /H	UB = 3.9E-07 /H LB = 5.0E-08 /H *EF = 2.8	B-2, P.202

TEMPERATURE SWITCH, PROCESS
FAIL TO OPEN/CLOSE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N3)	3.0E-07 /D	$1.1E-07 /D	EF = 10	B-1, P.23
IEEE-500	1.5E-07 /D	$1.2E-07 /D	EF = 3	B-1, P.23

TEMPERATURE SWITCH, PROCESS
SPURIOUS OPERATION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	1.0E-06 /H	$3.8E-07 /H	EF = 10	B-1, P.23
IEEE-500	2.9E-07 /H	$2.3E-07 /H	EF = 3	B-1, P.23

PRESSURE TRANSMITTER
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
NUREG-1740	1.9E-06 /H (WITH COMMAND)	*1.9E-06 /H	UB(95%) = 2.3E-06 /H LB(5%) = 1.6E-06 /H *EF = 1.2	B-2, P.205
	1.7E-06 /H (W/O COMMAND)	*1.7E-06 /H		
IEEE-500	8.8E-07 /H (RECOMMENDED; NOT REPORTED AS A MEAN)	*7.1E-07 /H	UB = 1.7E-06 /H LB = 2.0E-07 /H *EF = 2.9	B-2, P.206
IEEE-500	1.4E-06 /H	$1.1E-06 /H	EF = 3	B-1, P.23
EIDE/RECOMMENDED (NUCLARR N1)	3.0E-06 /H	$1.1E-06 /H	EF = 10	B-1, P.23
SEABROOK PRA	7.6E-06 /H	$5.2E-06 /H	EF = 4.2	B-1, P.23
WASH-1400	2.7E-06 /H (VALUE FOR PRESSURE SWITCH)	$1.0E-06 /H	EF = 10	B-1, P.23

PRESSURE ELEMENT
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED				

(NUCLARR N1)	1.0E-06 /H	$3.8E-07 /H	EF = 10	B-1, P.23
IEEE-500	1.9E-06 /H	$1.3E-06 /H	EF = 16	B-1, P.23

LEVEL TRANSMITTER
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
NUREG-[illegible]	1.9E-06 /H (WITH COMMAND)	*1.9E-06 /H	UB(95%) = 2.3E-06 /H LB(5%) = 1.6E-06 /H *EF = 1.2	B-2, P.205
	1.7E-06 /H (W/O COMMAND)	*1.7E-06 /H		
IEEE-500	1.4E-06 /H (RECOMMENDED; NOT REPORTED AS A MEAN)	*1.3E-06 /H	UB = 3.0E-06 /H LB = 7.1E-07 /H *EF = 2.1	B-2, P.206
IEEE-500	1.5E-06 /H	$1.4E-06 /H	EF = 2	B-1, P.24
EIDE/RECOMMENDED (NUCLARR N1)	3.0E-06 /H	$1.1E-06 /H	EF = 10	B-1, P.24
SEABROOK PRA	1.57E-05 /H	$1.3E-05 /H	EF = 2.7	B-1, P.24
WASH-1400	2.7E-06 /H	$1.0E-06 /H	EF = 10	B-1, P.24

LEVEL ELEMENT
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	1.0E-06 /H	$3.8E-07 /H	EF = 10	B-1, P.24

LEVEL SWITCH, PROCESS
FAIL TO OPEN/CLOSE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N3)	3.0E-07 /D	$1.1E-07 /D	EF = 10	B-1, P.24
IEEE-500	3.3E-08 /D	$3.0E-08 /D	EF = 2	B-1, P.24

LEVEL SWITCH, PROCESS
SPURIOUS OPERATION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	1.0E-06 /H	$3.8E-07 /H	EF = 10	B-1, P.24
IEEE-500	1.7E-06 /H	$1.6E-06 /H	EF = 2	B-1, P.24

TEMPERATURE ELEMENT
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (NUCLARR N1)	1.0E-06 /H	$3.8E-07 /H	EF = 10	B-1, P.23

IEEE-500	4.0E-06 /H	$1.8E-06 /H	EF = 8	B-1, P.23

TEMPERATURE TRANSMITTER
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
IEEE-500	3.7E-07 /H (INCLUDES: 1. ZERO OR MAX OUTPUT, 2. NO CHANGE OF OUTPUT)	*2.6E-07 /H	UB = 3.3E-06 /H LB = 1.9E-07 /H *EF = 4.1	B-2, P.206
IEEE-500	1.4E-06 /H	$9.8E-07 /H	EF = 4	B-1, P.23
EIDE/RECOMMENDED (NUCLARR N1)	3.0E-06 /H	$1.1E-06 /H	EF = 10	B-1, P.23
WASH-1400	2.7E-06 /H (VALUE FOR PRESSURE SWITCH)	$1.0E-06 /H	EF = 10	B-1, P.23

ALARMS
FAIL TO FUNCTION/OPERATE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
NUCLARR (N3)	1.0E-06 /H	$3.8E-07 /H	EF = 10	B-1, P.22
IEEE-500	2.5E-06 /H	$6.0E-07 /H	EF = 16	B-1, P.22

ALARMS
SPURIOUS OPERATION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
NUCLARR (N3)	3.0E-06 /H	$1.1E-06 /H	EF = 10	B-1, P.22
IEEE-500	1.7E-06 /H	$4.4E-07 /H	EF = 15	B-1, P.22

SWITCHES, GENERAL
FAIL TO OPEN

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (WASH-1400)	1.0E-05 /D	$6.2E-06 /D	EF = 5	B-1, P.22
IEEE-500	2.8E-07 /D	$5.0E-08 /D	EF = 21	B-1, P.22

SWITCHES, GENERAL
FAIL TO CLOSE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE/RECOMMENDED (WASH-1400)	1.0E-05 /D	$6.2E-06 /D	EF = 5	B-1, P.22
IEEE-500	3.0E-08 /D	$1.0E-08 /D	EF = 11	B-1, P.22

SWITCHES, GENERAL
SPURIOUS OPERATION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
WASH-1400	1.0E-06 /H (INCLUDES FAILURE TO PROVIDE PROPER OUTPUT)	$3.8E-07 /H	EF = 10	B-1, P.22
EIDE/RECOMMENDED (IEEE-500)	4.2E-06 /H	$2.5E-07 /H	EF = 50 (REDUCED BY ENGINEERING JUDGEMENT)	B-1, P.22

INDICATOR
FAIL TO FUNCTION

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
EIDE RECOMMENDED (IEEE-500)	1.0E-06 /H	$3.8E-07 /H	EF = 10	B-1, P.24
IEEE-500	1.4E-06 /H	$1.3E-06 /H	EF = 2	B-1, P.24

INSTRUMENTATION, GENERAL
FAIL TO OPERATE

SOURCE	MEAN	MEDIAN	VARIANCE DATA	REFERENCE
IREP	3.0E-06 /H	1.0E-06 /H	EF = 10 $UB(95%) = 1.0E-05 /H	B-3, P.129

B.3. REFERENCES

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B-4. B. G. Gilbert et al., *Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR)*, NUREG/CR-4639, Vol. 1-6, Revised August 1990.

B-5. T-L. Chu et al., *Interfacing Systems LOCA: Boiling Water Reactors*, NUREG/CR-5124, February 1989.

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Appendix C

Reference B&W Plant System Descriptions

CONTENTS

FIGURES

TABLES

APPENDIX C
REFERENCE B&W PLANT SYSTEM DESCRIPTIONS

The Reference B&W Plant began commercial operations 1976. The reactor is designed for a core power level of 2,772 MW(t) and a net electrical output of 906 MW(e). The reactor coolant system (RCS) comprises four reactor coolant pumps (RCP), two once-through steam generators, and has a total RCS fluid volume of 11,500 ft. A simplified schematic of the system is shown in Figure C-1. The relevant interfacing systems are described in subsequent sections.

C.1. High-Pressure Injection System

A simplified diagram of the High-Pressure Injection (HPI) system for the reference B&W plant is provided in Figure C-2. Table C-1 and C-2 list important operating characteristics of the HPI system. Two independent trains are shown and each train is capable of performing the system function. These trains are connected to two Low-Pressure Injection (LPI) suction lines commonly connected to the Borated Water Storage Tank (BWST). Also, additional valves are installed at the discharge side of the HPI pumps to allow cross connections between the two independent trains. The HPI pumps are rated at 2000 psig and can discharge coolant at 500 gpm. Once the HPI system is initiated, the HPI pumps will take suction from the BWST, and discharge borated water to two redundant flow paths leading to the cold legs penetrating the reactor vessel.

In the event of a Loss-of-Coolant Accident (LOCA), with the primary system pressure reaching 1650 psig, a Safety Feature Actuation Signal (SFAS, starts HPI pumps P58-1 and P58-2 and opens HPI isolation valves HP-2A, 2B, 2C, and 2D. The function of the High-Pressure Injection (HPI) system is to prevent core uncovery by injecting borated water into the core at high Reactor Coolant System (RCS) pressure. This provides the initial cooling needed to prevent fuel temperatures from reaching 2200°F. Fuel temperatures in excess of 2200 F can lead to a zirconium-water reaction with fuel and or cladding failure. If the pipe break is large enough to exceed the make up system capacity and small enough to maintain pressure above the Low-Pressure Injection (LPI) system initiation setpoint, the HPI system can be aligned to take suction from the Decay Heat Removal (DHR) pump. While the HPI system is providing make up, the water lost from the RCS is being collected in the Containment Emergency Sump. When the BWST is depleted, the DHR pumps provide suction to the HPI pumps from the Containment Emergency Sump. Long-term cooling for intermediate size breaks is also provided. Also, the HPI system provides borated water injection for large ruptures in the Main Steam Piping, which cause excessive contraction of the RCS.

The testing of the HPI system is typically performed when the reactor is shutdown for normal refueling. One train of the equipment that would be called upon to operate is tested. A safety actuation signal is applied

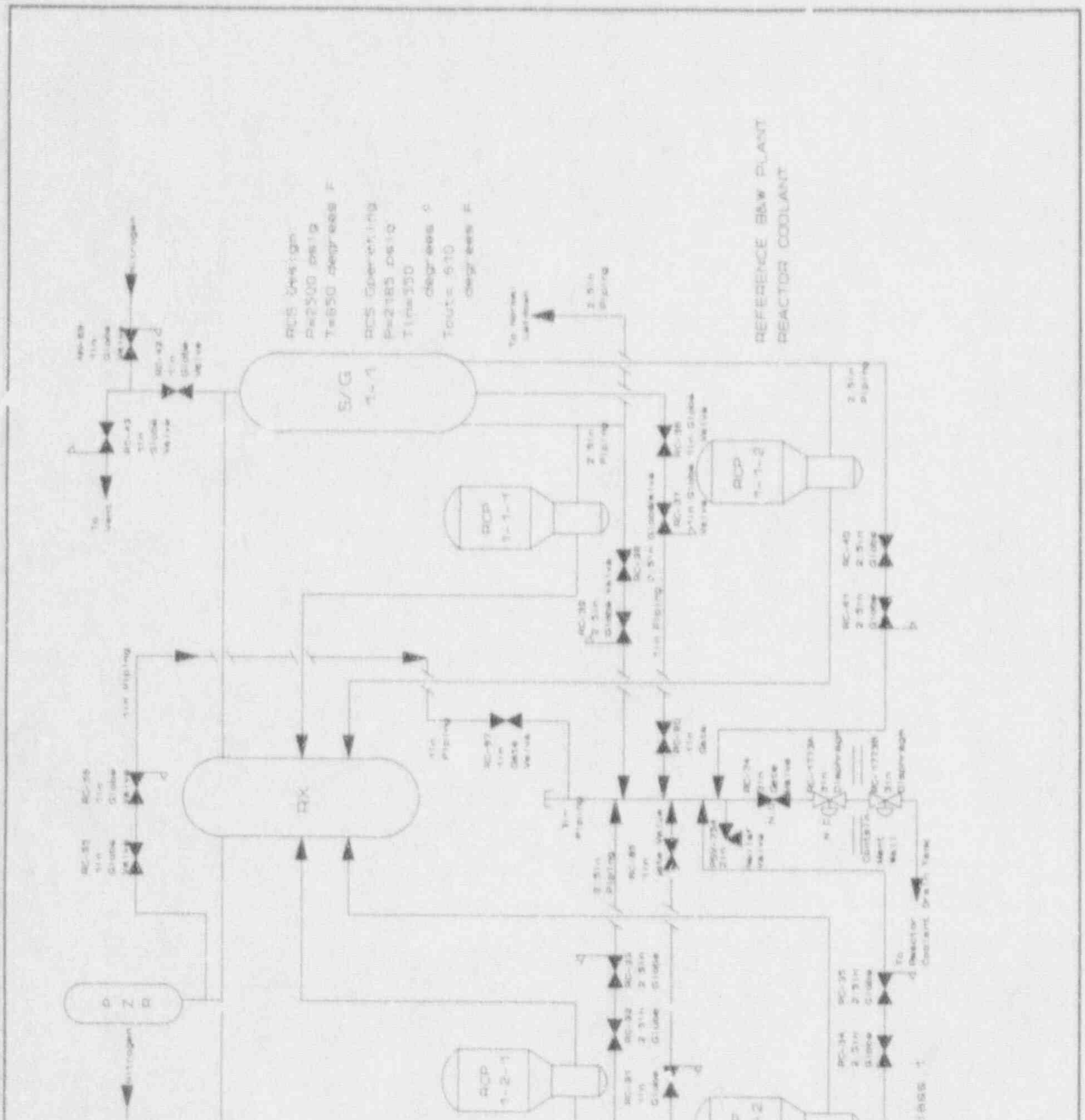

Figure C-1. Simplified Diagram of the Reference B&W plant Reactor Coolant System.

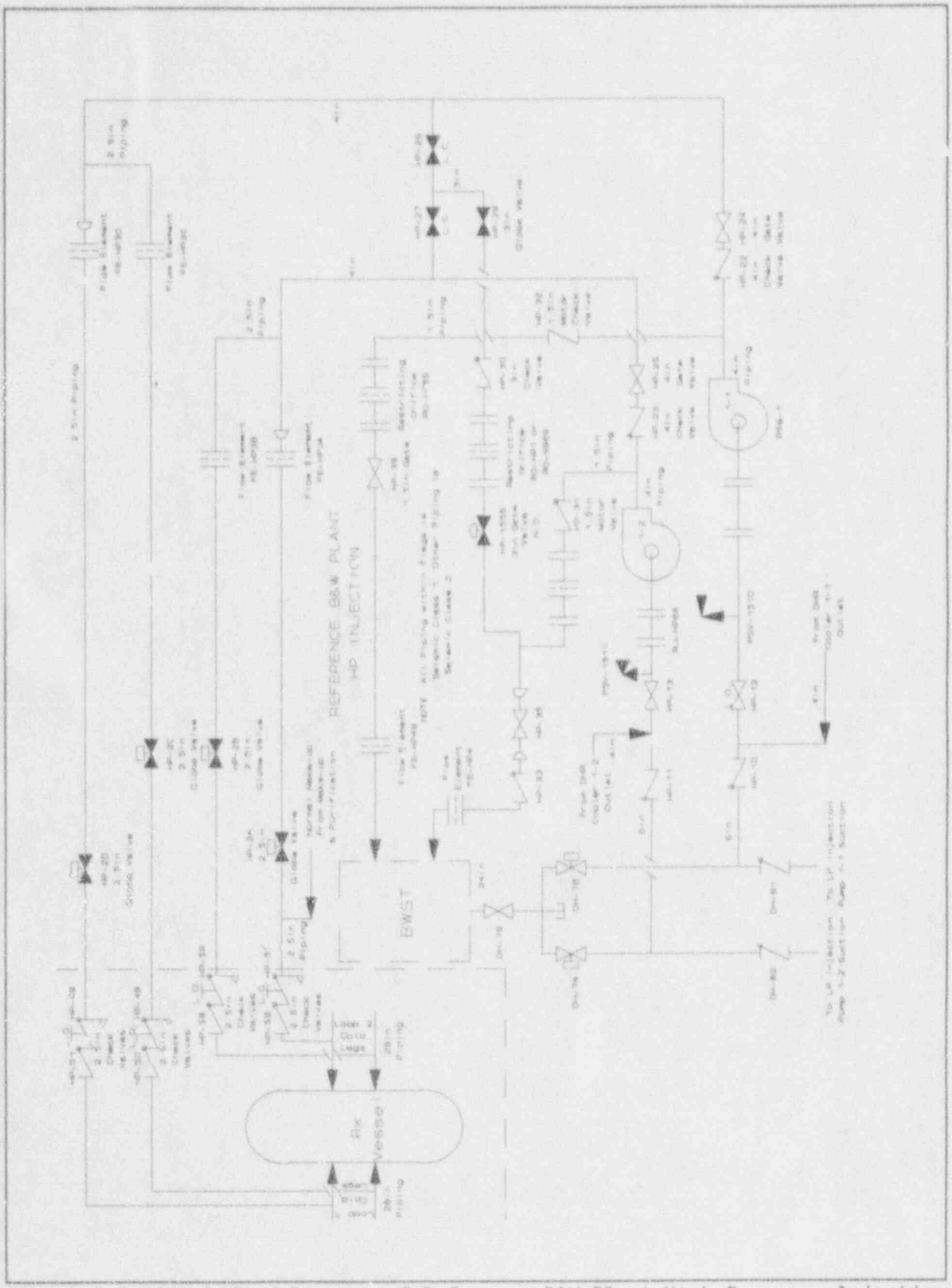

Figure C-2. Simplified Diagram of Reference B&W Plant High-Pressure Injection System.

separately to the HPI pump motor breaker and the HPI valves. The test is considered to be acceptable when the devices respond within a specified time frame. The valves that are required to stroke must be in their safety positions within 30 seconds. Provisions are also made to facilitate performance testing of components during operation of the plant. Quarterly, the applicable valves are stroked and the HPI pumps are tested in a recirculation mode to the BWST to ensure the capability of the pumps to perform their SFAS function. Once every 31 days, each valve in the flow path is verified to be in its correct position.

Table C-1. HPI Component Data

HPI Pump Motor	
Horsepower	600
Amperes	77
HPI Pump	
Type	11-stage centrifugal
Capacity	500 gpm
Head	2700 ft. (1200#)
Design Pressure	2000 psig
Design Temperature	300°F

Table C-2. HPI System Alarms and Setpoints.

Annunciator	
BWST Temp Low	50°F
BWST LVL Low to SFAS	8 ft
HP INJ 1(2) Flow High	475 gpm
HP INJ 1(2) Flow Low	75 gpm
Computer	
HPI Pump Recirc. Flow Low	37 gpm
HPI Discharge Header Press High	375 psig
HPI 1(2) dc oil pump on	dc oil pump on

C.2. Decay Heat Removal\Low-Pressure Injection System

A simplified diagram of the Reference B&W Plant Decay He⁻ˈ Removal DHR/LPI system is shown in Figure C-3. Two independent trai⁻ with suction line valves, pumps, and heat exchangers are shown. These trains are connected by common lines to the reactor hot leg outlet and BWST, but independently connected to the containment sump. Also, there are cross connections provided between the two trains at the discharge side of the DHR\LPI pumps. This system also interfaces with the HPI and Core Flood system. The DHR pumps are single stage, centrifugal pumps with a rated capacity of 3000 gpm.

The DHR and LPI systems are one and the same, but they serve different functions. During normal plant operations, the Steam Generators (SG) reduce the reactor coolant temperature to approximately 280°F. The function of the DHR system is to remove residual and sensible heat from the RCS by reducing the temperature from 280 to 140°F. Once reactor pressure reaches the appropriate set point (approximately 300 psig), DHR\LPI pumps P42-1 and P42-2 are started and valves DH-11 and DH-12 are opened. The DHR\LPI pumps take suction from the reactor outlet into redundant paths and discharge coolant through DHR coolers 1-1 and 1-2. The DHR coolers are designed to remove decay heat that is generated during normal shutdown. Finally, coolant passes through the Core Flood Injection nozzles to the reactor. The DHR\LPI system provides other functions such as providing auxiliary spray to the pressurizer for complete depressurization, maintaining temperature during refueling, filling, and partial draining of the refueling canal.

During a LOCA, if the primary system pressure drops and reaches 420.75 psig or the containment pressure increases to 18.4 psia, DHR\LPI pumps P42-1 and P42-2 will start. The DHR\LPI pumps take suction from the BWST and inject borated water through DHR coolers 1-1 and 1-2 and then to the reactor by the core flood injection nozzles. The system will remain in this alignment until the level in the BWST drops to approximately 8 ft. Then the DHR\LPI pumps are aligned to take suction from the Emergency Sump to recirculate the spilled water.

The system test of the DHR\LPI system is performed when the reactor is shut down for normal refueling. One train of the equipment that would be

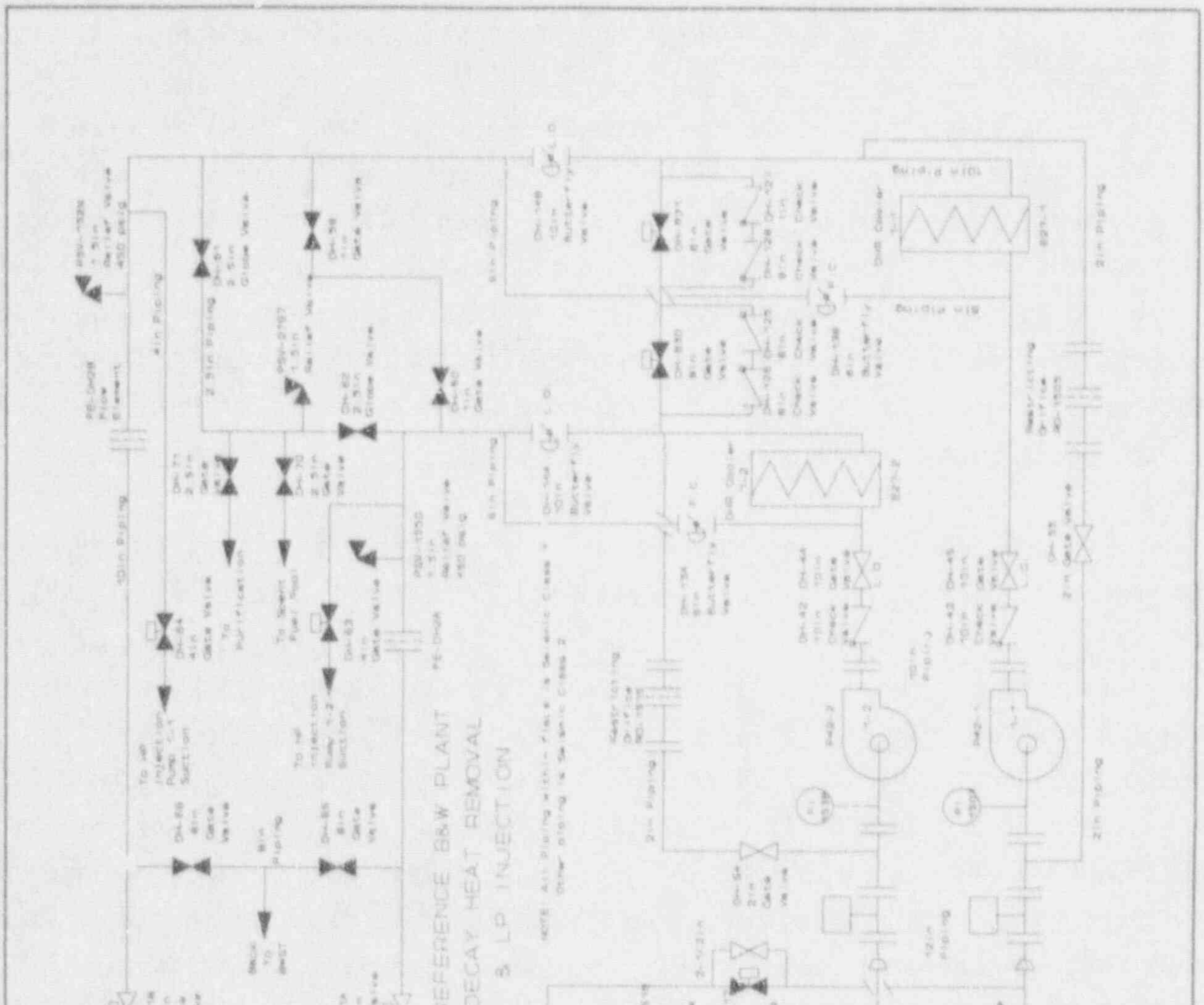

Figure C-3. Simplified Diagram of the Reference B&W Plant DHR/LPI System.

called upon to operate is tested. A safety actuation signal is applied separately to the DHR\LPI pump motor breaker and the DHR valves. The system test is considered successful if the devices respond within a specified time frame; valves that are required to stroke must be in their safety positions within 30 seconds, and provisions must be made to facilitate performance testing of components during operation of the facility. Quarterly, the DHR valves are stroked to verify their capability to function and the DHR\LPI pumps are tested in a recirculation mode to the BWST to ensure the pumps can perform their SFAS function. Also, every 31 days, each valve in the flow path is verified to be in its correct position.

Table C-3. DHR/LPI Component Data

DHR Pump Motor	
Horsepower	400
Amperes	50
DHR Pump	
Type	Single stage, centrifugal
Capacity	3000 gpm
Head	350 ft (150 psig)
Design Pressure	450 psig
Design Temperature	350°F
DHR Cooler	
Type	Shell and U-tube
RC flow (tube)	3000 gpm
CCW flow (shell)	6000 gpm
Design Pressure	
Tube	450 psig
Shell	150 psig
Design Temperature	
Tube	350°F
Shell	250°F
Heat Transfer Rate	105 million Btu/hr

Table C-4. DHR System Alarms and Setpoints

Annunciator	
DHR Cooler 1(2) Temp. High	280°F
DHR Pump 1(2) Suction Temp. High	315°F
LP Inj. 1(2) Flow High	3750 gpm
LP Inj. 1(2) Flow Low	2800 gpm

C.3. Core Flood System

A simplified diagram of the Reference B&W Plant Core Flood (CF) System is shown in Figure C-4. The CF System comprises core flood tanks (CFT) 1-1 and 1-2. Each tank has a volume of 1410 ft^3. Borated water occupies 1040 ft^3 and the remainder is filled with pressurized nitrogen gas. Each discharge line contains a motor-operated stop valve, and two check valves in series that are connected to one of the core flood nozzles. A DHR injection line interfaces with the two check valves. There are two lines connected to a common header that supplies makeup water or nitrogen to each tank.

The principal function for the Core Flood (CF) System is to provide emergency core injection at intermediate to low pressures and maintain core integrity during RCS leaks ranging from intermediate to large scale. The CF system is a passive system that requires no electrical power or operator intervention. During a LOCA when the primary system pressure decreases below the Core Flood Tank (CFT) pressure, the pressurized Nitrogen gas forces the borated water out of the CFTs and through the discharge lines allowing refilling of the reactor vessel. This is designed to prevent fuel clad temperatures from exceeding 2200°F.

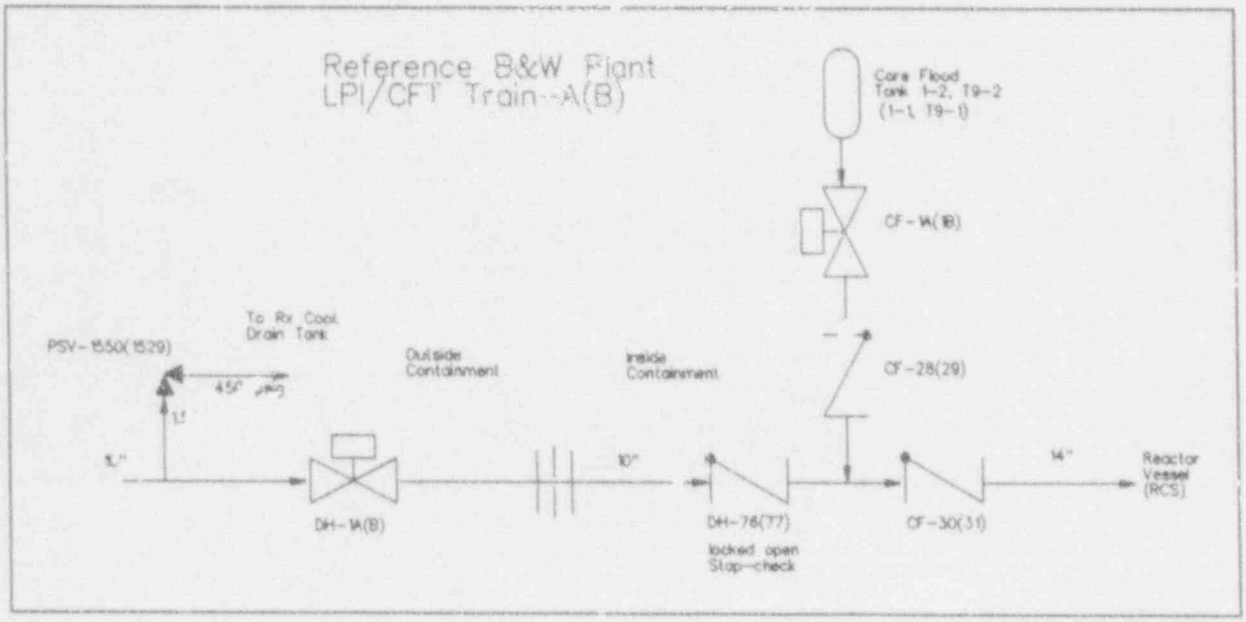

Figure C-4. Simplified Diagram of the Reference B&W Plant Core Flood System.

Testing of the CF system is performed when the Reactor Coolant System (RCS) is being depressurized. Testing consists of slowly lowering the RCS pressure below the CFT pressure and observing level changes in the CFT. The test is considered successful when the check valves open properly, and the level in the tanks decrease. When RCS pressure is increased, the check valves should seat with no significant level changes in the CFTs.

C.4. Makeup and Purification System

The Makeup and Purification (MUP) for the reference B&W plant is shown in Figures C-5 and C-6. Starting from the Makeup Tank, both trains are connected through a common line. These trains interface with redundant inlet HPI and Reactor Coolant Pump (RCP) injection lines. Return paths to the makeup tank include the RC letdown and RCP injection lines. These return paths include numerous types of valves, letdown coolers, seal return coolers, and filters. The makeup pumps are rated at 150 gpm at 2500 psig with runout at 350 gpm.

The Makeup and Purification (MU&P) system performs various functions during all phases of the Nuclear Steam Supply System (NSS) operation including startup, power operations, and shutdown. This system is also operated during refueling by employing purification equipment through interconnections to the DHR system. During normal NSS operation, one of the two Makeup pumps, P37-1 or P37-2, supplies injection water to RCS through a HPI line and to Reactor Coolant Pump (RCP) seals. The other makeup pump is on stand by. A control valve in the RCP seal injection line automatically maintains the desired flow rate to the seals. Needle valves in the injection lines manually throttle flow to the seals of the RCPs. However, part of the water supplied to the seals leaks into the RCS. To maintain the desired coolant inventory, a continuous letdown of coolant must occur. The reactor coolant is removed from the cold leg and passes through one of the two letdown coolers, E25-1 or E25-2. Pressure is reduced during flow through the letdown flow station. Impurities from the coolant are removed by flowing through a purification prefilter and a demineralizer. A three-way valve, MU-11, directs the coolant either through the makeup filter to the makeup tank or directly to the Clean Waste System. The level of the makeup tank is maintained with water from the Clean Waste System, the Boric Acid Addition Tank, or from the Demineralized Water Storage Tank. The makeup tank also receives chemicals for addition to the RCS. Chemicals in solution are injected into the letdown line upstream of the makeup filters and then passed into the makeup tank, which serves as a final mixing location. Coolant at the refueling boron concentration is supplied to the RCS for preoperational fill by using the boric acid pumps and the clean waste receiver transfer pumps or the demineralized water supply pumps. The fill line bypasses the makeup tank and makeup pumps and connects

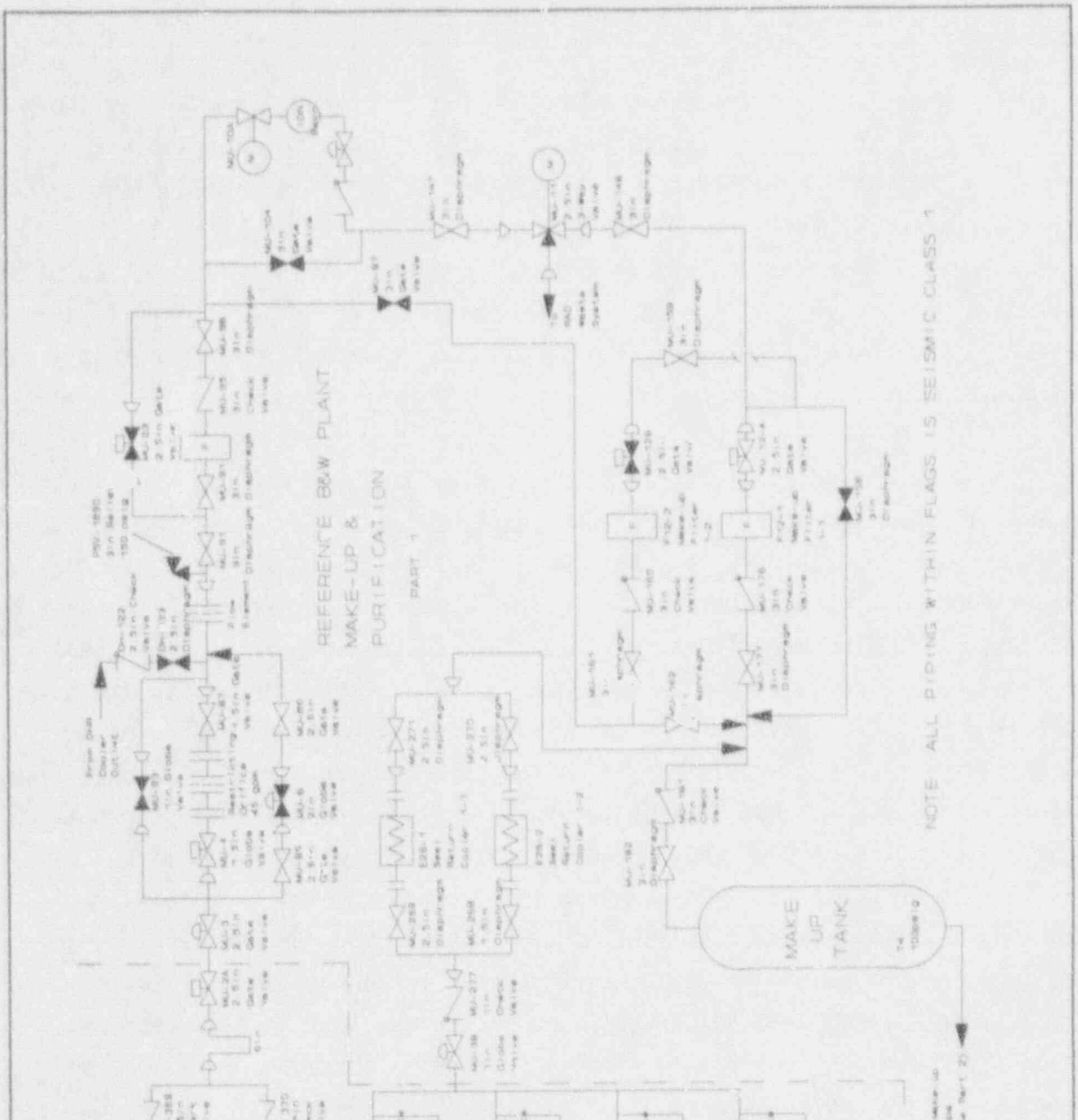

Figure C-5. Simplified P&ID of Reference B&W Plant MU&P System (1 of 2).

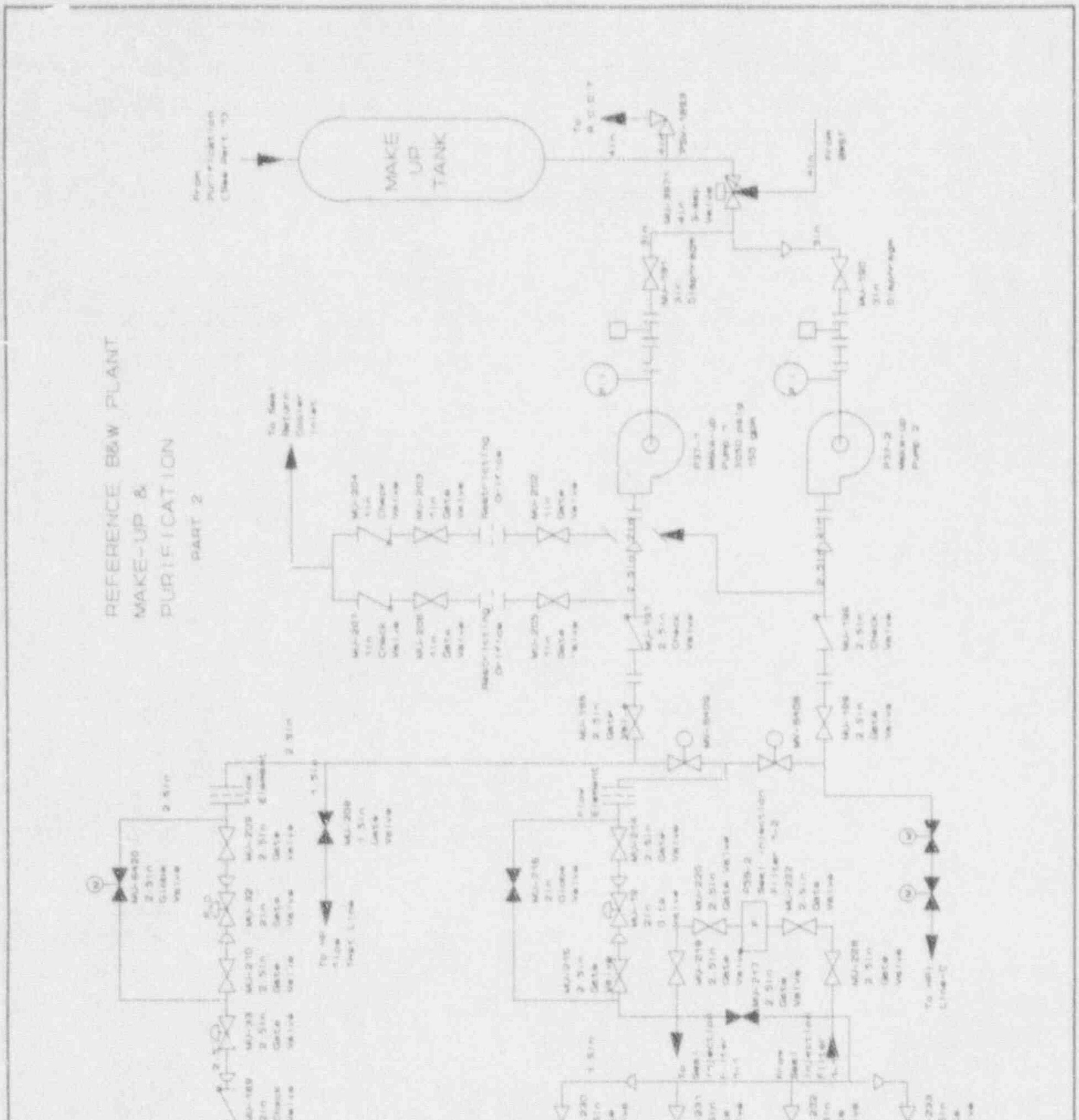

Figure C-6. Simplified P&ID of the Reference B&W Plant MU&P System (2 of 2).

into the RCS through the normal makeup control valve. When the fill operation is completed, the auxiliary line is secure and makeup and inventory control is then continued by operation through one of the makeup pumps. The MU&P system also provides makeup to the RCS by replenishing the inventory lost due to a small break in the RCS pressure boundary.

Components of the makeup and purification system are examined periodically to determine their operating condition. Periodic visual inspections, testing and preventive maintenance are conducted and practiced.

C.5. Pressure Isolation Valve Operability

Interfacing System LOCA analysis at the reference B&W plant has focused on three system isolation valves. These systems and their respective valves are as follows:

1. DHR system letdown isolation valves, 12 in. gate valves
2. LPI system injection valves, 10 in. gate valves
3. HPI system injection valves, 2.5 in. globe valves.

The gate valves are high recovery positive shutoff valves typically used in systems where minimal pressure drop is desired when the valve is fully open. They are used in many applications where they must open as well as close, because their primary function is one of isolation. The globe valves are typically used where flow control or leakage is of more concern.

A number of equations are currently used by the industry to estimate the torque and thrust requirements of a given valve and operator for a given application. One of these equations, the stem force equation, is used to estimate the stem force required to open or close a valve. This equation is considered the heart of the total operator sizing effort and is based on a static force balance of the internal parts of a valve. The unknown frictional coefficients can be estimated with small-scale test valves, then scaled as necessary to estimate the thrust requirements of larger valves.

The following sections discuss valve operator control and thrust potential, the two valve designs being evaluated at the B&W facility (the gate valve and the globe valve), and the estimated limiting pressure and differential pressure at which the valves can successfully operate, in both the opening and closing directions.

Valve Operator Control

The gate and globe valve applications being assessed involve inter system connections wherein an event in one system, such as a LOCA, can directly affect equipment, an isolation valve for instance, in another system. The valves in question are located in the DHR (gate valves), LPI (gate valves), and HPI (globe valves) systems. Operators for such valves can be controlled in one of two ways. The first is to monitor the torque output of the motor and shut the operator off when the load (and hence the thrust requirement of the valve) becomes too large. This technique has the imbedded assumption that the torque, at which the motor is tripped, is in excess of that required to fully close the valve. Under this condition, the motor will trip when the valve has seated and is fully wedged and has stopped moving, thereby offering infinite resistance to further motion.

The second technique of controlling the operator is through position limit switches. This technique allows the valve operator to develop its full torque and thrust potential until the valve is in either the fully open or the fully closed position. This method of control relies heavily on correctly positioned limit switches. A misaligned or nonfunctional limit switch can result in continual torque and thrust being applied to the valve until either the motor burns out, a power breaker trips or, if so equipped, a safety torque limit switch trips. If the loads on the valve are excessive and exceed the thrust capabilities of the operator, the valve will stop moving although full torque and thrust will continue to be applied, subject to the limitations of a torque limit switch if one is installed.

The valve operators at the reference B&W plant are controlled via the second method, the valve position limit switches, however they do not have a torque limit switch.

Operator Thrust Potential

The thrust potential of an operator will be the lesser of:

1. The maximum rated thrust of the operator
2. The maximum thrust deliverable to the stem by the motor.

The maximum rated thrust of an operator is dependent on its size where as the maximum thrust deliverable to the stem by the motor depends on the operating characteristics of the motor and the overall gear ratio between the motor and the stem. The resultant usable thrust will be the lesser of these two values. The maximum thrust deliverable to the stem can be estimated as follows:

$$F_t = O_t / SF$$

$$O_t = M_t \, M_{po} \, M_{af} \, OAR$$

$$OAR = M_s \, S_l / S_s$$

$$SF = \frac{d\,(0.96815 \text{ Tan } a + SN_f)}{24\,(0.96815 - SN_f \text{ Tan } a)}$$

$$\text{Tan } a = S_l / (3.14159\, d)$$

$$d = S_d - (S_p / 2)$$

where

F_t = maximum stem thrust the motor can develop
O_t = maximum operator torque the motor can develop
SF = stem factor
OAR = overall operator gearing ratio
M_s = motor rated speed
M_t = motor rated torque
M_{po} = pull out efficiency
M_{af} = application factor
S_d = stem diameter
S_p = stem pitch
S_l = stem lead
S_s = stem speed
SN_f = stem nut friction.

The above can be evaluated using the operator dependent parameters in Table C-5.

Gate Valve

The gate valve stem force equation used to estimate the thrust requirements of a larger valve based on the testing of a smaller valve is as follows:

$$F_t = f_d A_d DP \pm A_s P + F_p$$

where

F_t = total stem force
f_d = disc factor
A_d = disc area exposed to the flow
DP = differential pressure
A_s = stem cross-sectional area
P = pressure acting on the stem
F_p = packing drag load (a constant).

The first term, the disk load, represents the frictional resistance of the disk as it moves against a differential pressure loading. The industry typically assumes that full system pressure will act across the valve unless a system specific application justifies a lower differential pressure. The industry also assumes a 0.3 disk friction factor, although factors up to 0.5 are occasionally used when additional conservatism is desired. This force will always oppose valve motion.

The second term, the stem reaction load, represents the internal pressure trying to push the valve stem out of the valve. This force always acts outwards of the valve and will thus resist valve closure but assist valve opening. The industry typically assumes the pressure upstream of the valve is acting on the valve stem.

Table C-5. Operator Dependant Parameters

Parameter	12 in. valve Operator	10 in. valve Operator	2.5 in. valve Operator
Operator			
manufacturer	Limitorque	Limitorque	Limitorque
number	SMB-3-150	SMB-3-100	SMB-00-15
rated thrust, lb_f	140,000	140,000	14,000
Motor rated			
torque, ft lb_f	150	100	15
speed, rpm	1700	3600	1700
pull out efficiency[a]	0.400	0.400	0.400
application factor[a]	0.900	0.900	0.900
Stem			
diameter, in.	2.750	2.500	1.125
pitch, thd./in.	0.333	0.333	0.200
lead, thd./in.	0.333	1.000	0.400
speed, in./min.	10.5	51.6	13.0
Stem nut friction[b]	0.200	0.200	0.200
Stem factor	0.02688	0.03431	0.01449
overall gear ratio	53.835	69.750	52.135
max. operator torque, ft lb_f	2,907	2,511	282
max. stem thrust, lb_f	108,151	73,186	19,429
Usable stem thrust[c], lb_f	108,151	73,186	14,000

a. Values are typical of those observed by the INEL during the Motor-Operated Valve testing program discussed under the gate valve section.

b. For operators operating within a normal range of frictions, a 0.2 friction factor is conservative and bounds the rate of loading concerns currently being explored by the INEL.

c. The usable stem thrust is the lesser of the operator rated thrust and the maximum stem thrust that the motor can produce.

The third term, the packing drag load, varies from valve to valve and is primarily the result of maintenance to control leakage through the stem region of the valve. The industry recognizes the variability of this parameter and assigns a conservative packing drag load to reflect extreme packing compression. The packing drag load will always oppose valve motion.

In order to assess the operability limits of a valve, the thrust limit must be known. With this information, and assuming that the valve develops its maximum loading near closure, the variables in the above equation can be evaluated using the following:

F_t = 73,186 lb_f (10 in. valve)
108,151 lb_f (12 in. valve)

1_d = 0.40 for flow orifice blockage, minimum flow exists
0.55 for complete closure and wedging of the disc
0.70 for opening

A_d = 50.240 in^2 (10 in. valve)
86.542 in^2 (12 in. valve)

A_s = 4.906 in^2 (10 in. valve)
5.936 in^2 (12 in. valve)

F_p = 2500.0 lb_f (10 in. valve)
4000.0 lb_f (12 in. valve).

The above values for the disc factor are based on testing performed by the Idaho National Engineering Laboratory (INEL). The U.S. Nuclear Regulatory Commission is sponsoring this valve and motor-operator functionality research in support of Generic Issue (GI)-87, "Failure of HPCI Steamline Without Isolation." Among the objectives of this research program is a task to determine what factors affect the performance of motor-operated gate valves and to determine how well industry's analytic tools predict that performance. This research program also supports the implementation of Generic Letter (GL) 89-10, "Safety-related Motor-Operated Valve Testing and Surveillance," which is applicable to all light water reactor safety-related motor-operated valves (MOVs) as well as selected position changeable MOVs in safety-related systems.

Three boiling water reactor (BWR) process lines were investigated. These include the HPCI turbine steam supply line, the reactor core isolation cooling (RCIC) turbine steam supply line, and the reactor water cleanup (RWCU) process line. All three of the BWR process lines communicate with the primary system, pass through containment, and normally have open isolation valves. The concern with the isolation valves is whether they will close in the event of a pipe break outside of the containment. A high energy steam or hot water release in the auxiliary building could result in the common cause failure of other components necessary to mitigate the accident.

One major area of the research program was the evaluation of two full-scale flexible wedge gate valve qualification and flow interruption test programs. In 1989, these tests were performed, in part, at the Kraftwerk Union (KWU) facilities near Frankfurt, Germany. Six valves were tested: three 6-in. valves typical of those used in RWCU applications and three 10-in. valves typical of those used in HPCI applications. One of the 6-in. valves was also tested at RCIC conditions. In all, seven design basis flow interruption tests were performed.

The test results clearly show that for the GI-87 concerns, all valves subjected to design basis flow interruption tests required more torque and subsequently more thrust to close than would be predicted using the standard industry motor-operator sizing equation for disc load calculations at common coefficients of friction. The highest loads recorded were the result of internal valve damage caused from the high differential pressure loads across the valve disc as it attempted to isolate flow. The analysis of the results also shows that the industry's disc load calculation equation is incomplete. It appears that the pressure distributions across the disc have obscured the true disc friction factor, which is probably much closer to the 0.6 to 0.7 that Westinghouse found after the EPRI Marshall PORV block valve tests than the 0.3 that industry has been using for the last 30 or more years.

The equation used to estimate the stem thrust requirements of a gate valve also assumes that the maximum stem force loading occurs when the valve is near full closure. At this time, the disk area is maximized as is the differential pressure across the valve, the dominant terms in the equation. However, based on the testing performed by the INEL, this is not always the case. This observation further supports the above statement that the industry's disc load calculation equation is incomplete. This issue is currently being addressed by the INEL, although the disc factors used in this assessment should bound this phenomena.

The remaining terms in the above equation (the valve pressure and differential pressure) can be estimated with the aid of one additional assumption, that the postulated pipe break occurs in the vicinity of but downstream of the isolation valve in question. With this assumption, the differential pressure would be equal to the upstream pressure of the valves.

The results thus represent the pressure at which the valve reaches its threshold limit of operability. This threshold will differ depending on whether the valve is being opened or closed and whether complete closing and wedging of the disc in the seat or flow orifice blockage with minimum flow is desired.

Globe Valve

The globe valve stem force equation used to estimate the thrust requirements of a larger valve from testing of a smaller valve is as follows:

$$F_t = A_d DP + F_p$$

where

F_t = total stem force
A_d = disc area exposed to the flow
DP = differential pressure
F_p = packing drag load (a constant).

The first term, the disk load, represents the frictional resistance of the disk as it moves against a differential pressure loading. The industry typically assumes that full system pressure will act across the valve unless a system specific application justifies a lower differential pressure. This force will always oppose valve motion when the flow is from under the disc.

The second term, the packing drag load, varies from valve to valve and is primarily the result of maintenance to control leakage through the stem region of the valve. The industry recognizes the variability of this parameter and assigns a conservative packing drag load to reflect extreme packing compression. The packing drag load will always oppose valve motion.

In order to assess the operability limits of a valve, the thrust limit must be known. With this information, and assuming that the valve develops its maximum loading near closure, the variables in the above equation can be evaluated using the following:

F_t = 14,000 lb_f

A_d = 3.454 in^2

F_p = 1500.0 lb_f.

The remaining term in the above equation (the valve differential pressure) can be estimated with the aid of one additional assumption: that the postulated pipe break occurs in the vicinity of, but downstream of the isolation valve in question. With this assumption, the differential pressure would be equal to the upstream pressure of the valves. The results thus represent the pressure at which the valve reaches its threshold limit of operability. Because the globe valve at the reference B&W plant is orientated such that flow is from under the disc, the pressure will tend to open the valve and will require little if any stem force. However, the pressure will tend to oppose closure of the valve and the stem thrust will be determined by the above relationship.

Valve/Operator Sizing Results

Table C-6 presents the threshold pressure and/or differential pressure at which the valves will successfully operate based on the above assumptions. The three systems evaluated include the DHR system letdown isolation valves, LPI system injection valves, and the HPI system injection valves. Note that degraded voltage conditions were not considered and that these valves are assumed to be operating at a normal system voltage of 460 vac.

Table C-6. Valve Data and Pressure Limit Results

				Limiting Operating Pressure (psig) and/or differential pressure (psid)		
					Closure	
System	Valve Number	Size	Type	Opening	Flow Orifice Blockage	Complete Closure/ Wedging
DHR	DH-11,-12	12 in.	Gate	1906	2568	1946
LPI	DH-1A,B	10 in.	Gate	2336	2827	2172
HPI	HP-2A,B,C,D	2.5 in.	Globe	N/A	N/A	3619

Appendix D

Reference B&W Plant ISLOCA Event Trees

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FIGURES

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APPENDIX D
REFERENCE B&W PLANT ISLOCA EVENT TREES

D.1. Introduction

Detailed descriptions of the ISLOCA event trees for the B&W reference plant are contained in this appendix. These event trees were developed based on an understanding of the capabilities of the plant's hardware and procedures. The detail of the event trees is necessary to accurately describe and analyze the ISLOCA challenge. The ISLOCA sequence events can be divided into three main groups: 1) initiation events, 2) rupture event, and 3) recovery events. The initiation events comprise all events that contribute to the violation of the pressure isolation barrier. This can include hardware faults, human errors of commission, and latent faults. The rupture event is a single event describing the probability that, given the pressure isolation barrier is opened, the interfacing system ruptures. It is the result of a series of calculations estimating the local internal pressure experienced by the interfacing system and the expected rupture pressure for each component in the system. The last phase of the sequence considers the potential for the operators identifying the occurrence of an ISLOCA, diagnosing the cause, isolating the rupture, and mitigating any possible radioactive releases (if the sequence was not recovered).

Given an ISLOCA has occurred, a high priority item for the control room operations crew should be isolating the break and terminating the ISLOCA leak. This action should be taken because the supply of water for cooling the core is limited. The BWST coolant inventory is maintained at about 480,000 gallons. A small ISLOCA break (equivalent to a 2-in. line) will result in an initial leak rate of about 1,000 gpm. At this leak rate the BWST would be depleted in about 8 hours. The BWST makeup system would not significantly affect this scenario at these 1000 gpm leak flow rates. Other postulated ruptures, particularly those associated with the DHR system, can result in much larger leakage rates. When the breaks are isolated in a timely manner and the leak terminated, the plant can be safely cooled down using the auxiliary feedwater system (AFW) and steam generators (SG). This is particularly significant for those sequences where the likely break location would result in disabling one or both trains of the DHR system.

The following sections describe the event trees developed for the five ISLOCA sequences. The quantification of the event trees is based on a yearly time frame. This time frame is reflected in the frequency of the initial event-tree events. The initiating event simply postulates a particular operating mode or status of the plant and includes consideration of multiple interface lines. The plant operating status modeled in the initial event is only slightly conservative. The event trees are based on the plant operating all four quarters per year. The event trees also includes one outage (during which manual valves DH-21 and DH-23 are opened to allow MOVATS testing of DH-11 and DH-12) with a single startup and shutdown. The event trees are constructed such that the downward branch depicts the failure event listed at the top of the event tree and the upward branch denotes the complement of the event (typically, success). The top events are a combination of individual component failures, human errors, and functional failures that were deemed most appropriate for describing the individual ISLOCA scenario progression.

The event frequencies described in this Appendix are mean values and are presented as point estimates. A separate uncertainty analysis has been performed and is presented in Appendix L.

Each event tree end-state described in this Appendix was assigned to one of the release categories listed below.

OK - No overpressurization of the low-pressure system occurred.

OK-op - This scenario that results in overpressurization of the interfacing system. The system does not rupture or leak.

LK-ncd - This scenario results in a rupture in, and RCS leakage from, the interfacing system, but no core damage occurs. The leak is either isolated before core uncovery or the leak is too small to interfere with core cooling.

LOCA-ic - Identifies scenarios that produce a loss-of-coolant-accident inside containment. The ECCS is functional and as a result this scenario is not considered a core damage event.

REL-mit - An ISLOCA in which core damage occurs. The radioactive release is mitigated through an accident management strategy.

REL-1g - An ISLOCA with core damage occurs and results in a large unmitigated radioactive release.

The REL-mit and REL-1g categories are sometimes subdivided according to failure location, with the event-tree end-states identified as RL1, RL2, etc. These are described further in the appropriate sequence description.

D.2. Makeup and Purification System Interface Event Tree - MU&P

A schematic diagram of the interface between the makeup and purification system (MU&P) and the reactor coolant system (RCS) is shown in Figure D-1. The base case ISLOCA event tree for this system is shown in Figure D-2. The MU&P system supplies high-pressure purified makeup to the RCS and seal injection to the reactor coolant pumps. The normal RCS makeup flows from the MU&P system through the HPI A-header via check valves HP-57 and HP-59. MU&P/HPI system features include:

(a) The HPI pressure isolation check valves (PIVs HP-57/59, HP-56/58, HP-48/50, and HP-49/51) are welded together. This prevents leak testing of individual check valves. Therefore, upon completion of a successful leak test, only one of the two check valves can be assured of being properly seated;

(b) The normally closed HPI MOVs (HP-2A, B, C, and D) are stroke tested quarterly. While the A-header valve (HP-2A) is being stroke tested, the MU&P system continues to provide RCS makeup through that line. When HP-2A is opened during the test, high-pressure makeup water backflows to the HP-pump discharge check valve (HP-23). Once the test is completed, the MOV is closed, and the HP line is vented by opening HP-27 and HP-29 to a HPI-pump test recirculation line. This same recirculation line is opened to the BWST for the quarterly HPI-pump flow test. This process presents an opportunity for mis-aligning the recirc line after the pump test, and/or HP-2A after the stroke test, possibly allowing RCS water to backflow to the BWST.

The MU&P event tree events are defined as follows. Point estimates of the base case branch probabilities are also listed.

M1-MU - Plant Operating in Mode 1. 4.0

The event tree is quantified on a yearly basis. In order to account for the quarterly stroke tests of the high-pressure injection valves, the

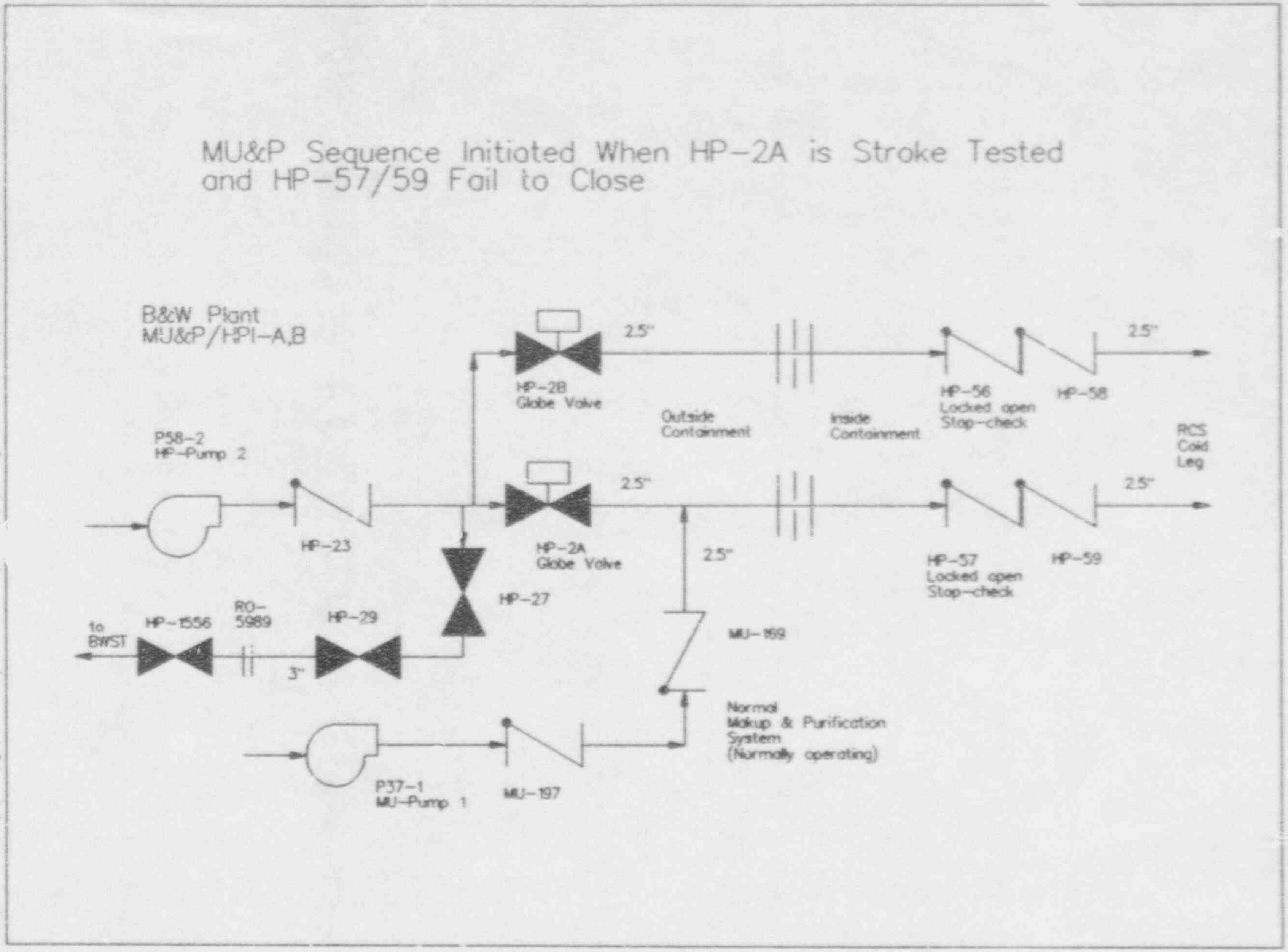


Figure D-1. Schematic diagram of the makeup and purification system interface.

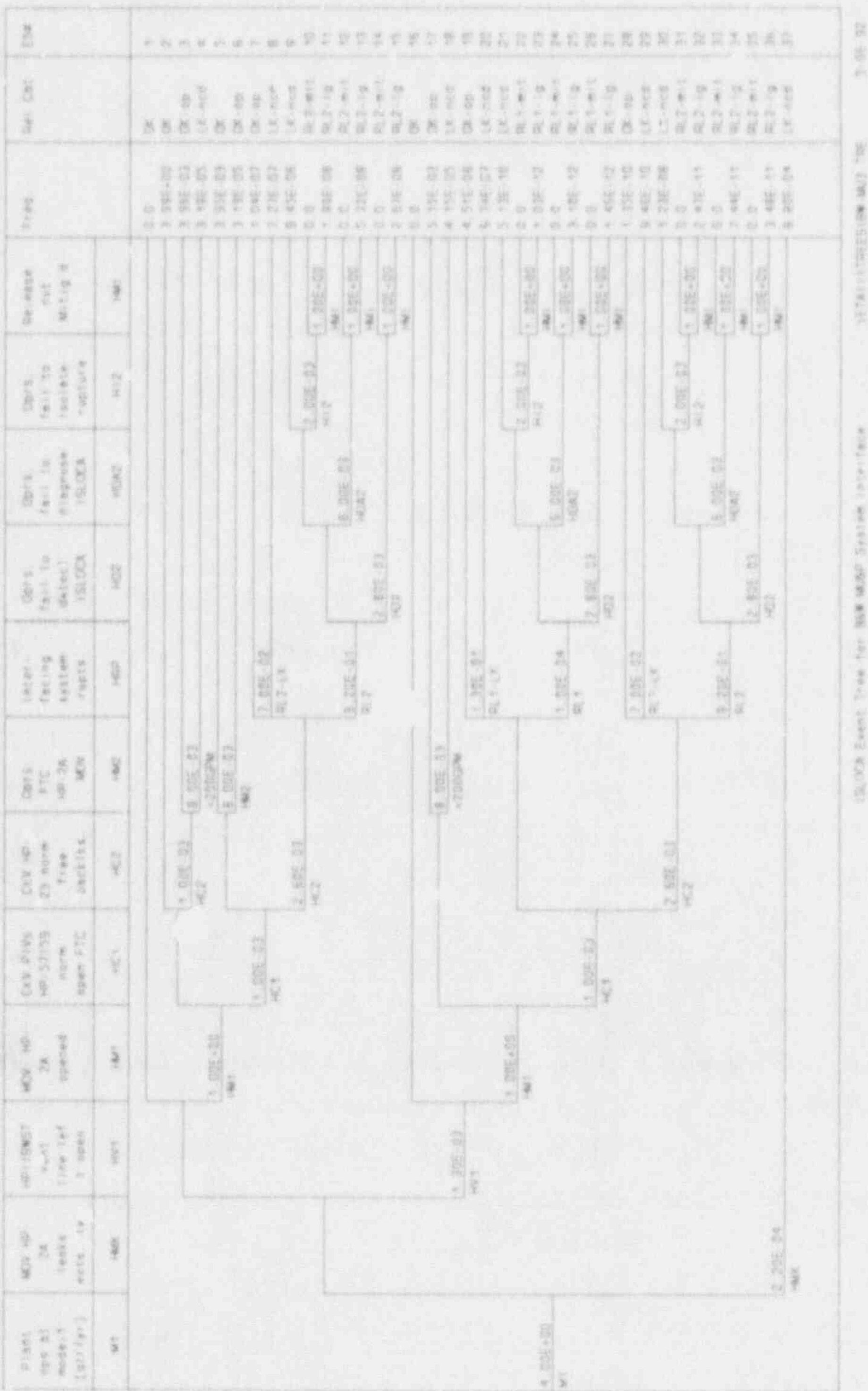


Figure D-2. Reference B&W Plant makeup & purification system interface event tree.

initiating event is quantified based on four quarters per year to obtain a yearly estimate of the accident frequency. The subsequent events are quantified under the assumption that the MU&P system is operating through leg A.

HMX-MU - MOV HP-2A Leaks Externally (Makeup Water). 2.2E-4

This event's probability is calculated by taking the product of the hourly failure rate of 1.0E-7 (see Appendix B). The hourly failure rate is calculated from the LER aggregations and the number of hours per quarter (2,190). This event results in a makeup water leak outside containment. The leak rate is expected to be small.

HV1-MU - HPI to BWST Vent Line Left Open. 0.0013

This event addresses the possibility that the operators inadvertently leave the vent line open after the previous HPI pump test. The normal procedure for executing the pump test includes opening a recirculation line from the pump to the BWST. This is the same line used to vent the pressure in the HPI line between the HPI pump discharge check valve HP-23 and HP-2A after the HP-2A stroke test. However, in the venting procedure leaves HP-1556 closed. This event considers the chance that the recirculation line (specifically manual valves HP-27, HP-29, and HP-1556) is open at the time the stroke test is conducted.

The HRA task analysis determined that the RO must check the vent path. The valves in question are locally operated and there is no direct procedural warning for the possibility that this line-up could contribute to an ISLOCA. Also considered in the HRA was the lack of a valve status board in the control room to indicate valve status for the crew, and no control room instrumentation indicating valve positions. Modeling of the HV1 event includes: 1-the potential for the shift supervisor to inquire about the status of these valves, 2-failure to send an EO to close the valves, and 3-errors of omission and commission for the operators to correctly close HP-27 and HP-29. The human error failure probabilities for this event were obtained from THERP and NUCLARR. (See Appendix E; Figure 4, 5, and 6; Table E8)

HM1-MU - MOV HP-2A Normally Closed is Opened. 1.0

The probability for this event is based on the routine quarterly stroke tests of MOV HP-2A. The valve is opened during this event.

HC1-MU - Pressure Isolation Check Valves HP-57 and HP-59 Normally Open, Fail to Close. 1.0E-3

This is a demand failure rate for one valve. The failure rate is based on data in the NUCLARR database (see Appendix B).

These valves are welded together and can only be leak tested as a pair. The failure rate data for one valve was used as a result. If failure of one valve were to occur it would not be detected during leak testing. This is because the leak test can only verify that one of the two valves is positively seated. The selection of the demand failure rate for one valve then provides a bounding case for this event.

Success of this event (valve closes) gives rise to a situation in which the potential coolant loss from the RCS is limited to the MU&P letdown flow rate (typically about 75 gpm). The MU&P flow will be diverted from the RCS and the control room operators may increase the makeup flow rate in response to the resulting decrease in pressurizer level. With the valves closed, the net leakage rate out is limited to the diversion of make-up flow. The RCS loss consists of the letdown rate via the MU&P system.

HC2-MU - Check Valve HP-23 Normally Free, Backleaks. 1.0E-3/2.6E-3

Because the HP pump is tested quarterly, this valve is required to close if a leak occurs in the PIV. For most scenarios on this event tree, the failure probability used is the conditional probability of a second check valve failing to close, given the failure of another check valve. Both HP-56/58 (treated as a single check valve) and HP-23 are modeled as having identical failure rates. The failure of the two are then correlated and the probability of both valves failing is higher than the combination of two independent failures. The value of 1.0E-3/demand (EF=5) for one valve is from the NUCLARR database. The aggregated failure probability of two valves (i.e., HP-57/58 and HP-23) is calculated as 2.6E-6 (using Monte Carlo sampling).

HM2-MU - Operators Fail to Close HP-2A MOV. 0.008

During the quarterly stroke test of valve HP-2A, the valve is opened and the stroke time is measured. The valve is then returned to its normal closed state. The possibility exists for the operators to fail to reclose the valve. The probability is based on the combination of both hardware failure (from Appendix B) and human error (from Appendix E) probabilities (3.8E-3 plus 3.8E-3, respectively). The HEP was determined from THERP for properly implementing a procedure and includes omission as well as inadvertent selection of any similar switches on the panel (see Appendix E; Figure 7 and 8, Table E9).

HRP-MU - Interfacing System Ruptures. 0.92/0.07 - 1.0E-4/0.13

This event is evaluated in a separate analysis that uses a series of RELAP5 computer runs (Appendix F) to estimate the pressures generated in the low-pressure piping and components. These estimated system pressures are then compared to the estimated failure pressures. These failure pressures were obtained from a structural analysis performed by ABB IMPELL Corporation (NUREG/CR-5603). The rupture probabilities for both the system and individual components are obtained from a Monte Carlo simulation that compares system pressure to the estimated component failure pressure (see Appendix H). Rupture is assumed to occur if the system pressure exceeds the estimated failure pressure in the simulation. The rupture probability of a component is simply the fraction of the Monte Carlo sample observations in which the system pressure exceeds the failure pressure. The rupture probability estimate for a given location in a system is obtained by combining the rupture probabilities of components located in the area of interest. This composite probability is the one used in the event tree.

A review and walkdown of the system, in combination with the analysis described above, revealed two likely failure locations. The first location is in the recirculation line to the BWST, downstream from manual valve HP-35. At this point the pipe schedule changes from 1500 psi rated to 150 psi rated.

Overpressurization of the BWST is not a credible scenario since this tank contains both an overflow and vent line. The second failure location is in the suction piping of the HPI pump. For a rupture to occur in this location, the HPI pump discharge check valve (HP-23) would have to fail to close when demanded (see event HC2, above). The BWST recirculation line and the HPI pump suction line are identified as failure locations RL1 and RL2, respectively. Because the ECCS pumps share a common room (e.g., all train-A pumps in one room) and the recirculation line passes through one of the rooms, a failure in either location would likely disable one train of each ECC system. This failure would include the HPI, LPI, and CSS, but would not include the MU&P system.

HD2-MU - Operators Fail to Detect ISLOCA. 0.0028

The HEP value reflects recognition on the crew's part that a rupture in an interfacing system has occurred. The detection of the ISLOCA was modeled so that is not necessary to include the identification of the cause or the corrective actions that need to be taken to isolate or mitigate the accident.

The detection of the ISLOCA event may require that the operations crew recognize that the following information indicates that an ISLOCA has occurred: 1) observation of 2 out of three computer-based alarms (high temperature alarm for the HP pump 1-2, RAD-FA alarm, or Auxiliary building sump) and 2) recognition of 1 of 2 available annunciators (decreasing makeup tank level or local annunciation of relief valve 1511 open). Other items not taken into consideration are indications of low seal injection for the reactor coolant pumps, low makeup flow, decreasing pressurizer level, and decreasing level in the makeup tank. These other indicators were not considered since they are typical signatures of a design basis LOCA.

Plant interviews indicated that during an ordinary stroke test of HP-2A, the high-pressure alarm would sound. Therefore, credit was not given for that alarm being part of a unique ISLOCA signature. HEP values from THERP and a 2 out of 3 failure logic based on plant interviews were used to model of the amount of information necessary for the operators to conclude that an ISLOCA has occurred (see Appendix E, Figure 9, Table E11).

HDA2-MU - Operators fail to diagnose ISLOCA. 0.006

Because no ISLOCA procedure exists for this plant, part of the process of diagnosis depends on the crew's ability to conclude that the existing fault must lie outside of the Small Leak Procedures. Failure to implement the procedure properly was modeled in HRA event trees with quantification values determined from THERP. This modeling took into account RO-EO communication and potential recovery factors. The instrumentation available is described as part of HD2-MU, "Operators Fail to Detect ISLOCA." The HEP value for HDA2 reflects performance shaping factors such as time available to the crew to diagnose, stress, resources such as procedures or instrumentation, and training (see Appendix E, Figure 11, Tables E13 and E14).

HI2-MU - Operators Fail to Isolate ISLOCA. 0.002

After the operators become aware of an abnormal situation, they must select the appropriate procedure and begin corrective action(s). HI2-MU models the probability that the crew gets caught up in trying to diagnose the situation, forgets that HP-2A is still open, and has no ISLOCA procedure to direct them to the right actions. In this case, the HEP value represents the crew's realization that there is a connection between the test procedure and the ISLOCA and takes into account the appropriate control actions (i.e., closes the valve). The HEP was determined from THERP and represents the potential for the crew to view symptoms properly and conclude that an ISLOCA exists, but due to the moderately stressful situation, select an inappropriate response (see Appendix E, Figure 12).

HM1-MU - Operators Fail to Mitigate the Release. 1.0

Many things determine the potential for mitigating a possible radioactive release from an ISLOCA. These include: location of rupture, submergence of the break, presence and operation of fire suppression sprays, design of the auxiliary building (water tight doors, flow paths to the environment, etc), conditions at the time of core degradation (i.e., temperature of the pipes, water and surfaces inside the aux-bldg), and effects of severe accident phenomena such as possible hydrogen generation and burning.

The results of the aux-bldg environmental analysis (see discussion of Break Sequence 5 in Appendix M) indicate that for this ISLOCA sequence, the break will likely be submerged. However, the calculations performed for this study only consider the time up to failure of all ECCS, and do not examine conditions at the time of core degradation. Therefore, given the uncertainties associated with the parameters mentioned above, the initial assumption used the bounding situation that all releases are unmitigated. Once risk was calculated this assumption was reexamined. Based on the relatively low ISLOCA risk results (i.e. 6 person-rem/Rx-year), it was deemed prudent to not expend additional effort on this issue.

D.3. High-Pressure Injection System Interface Event Tree - HPI

Figure D-3 shows a schematic diagram of the interface between the HPI system and the RCS. The ISLOCA event tree for this system is shown in Figure D-4. Each of the two HPI pump trains branch into two injection legs. Each injection leg then discharges into one of the RCS cold legs. The pressure isolation boundary is maintained by:

1. two check valves that are welded together,
2. a normally closed MOV (stroke tested quarterly) and,
3. the HPI pump discharge check valve.

Because the MU&P system provides normal makeup to the RCS through a connection in HPI leg A, that line is analyzed separately. The other three injection legs are modeled together in the HPI event tree.

M1-HP - Plant Operating at Mode-1. **12.0**

The event tree is quantified using four quarters per year multiplied by three injection lines. This produces a yearly estimate of accident frequency. This is done to account for the quarterly stroke tests of the high-pressure injection valves. The event tree models the three injection lines that do not normally have makeup flow through them. The key implication is that the pressure boundary check valves are normally closed with a 2200 psi differential pressure across them.

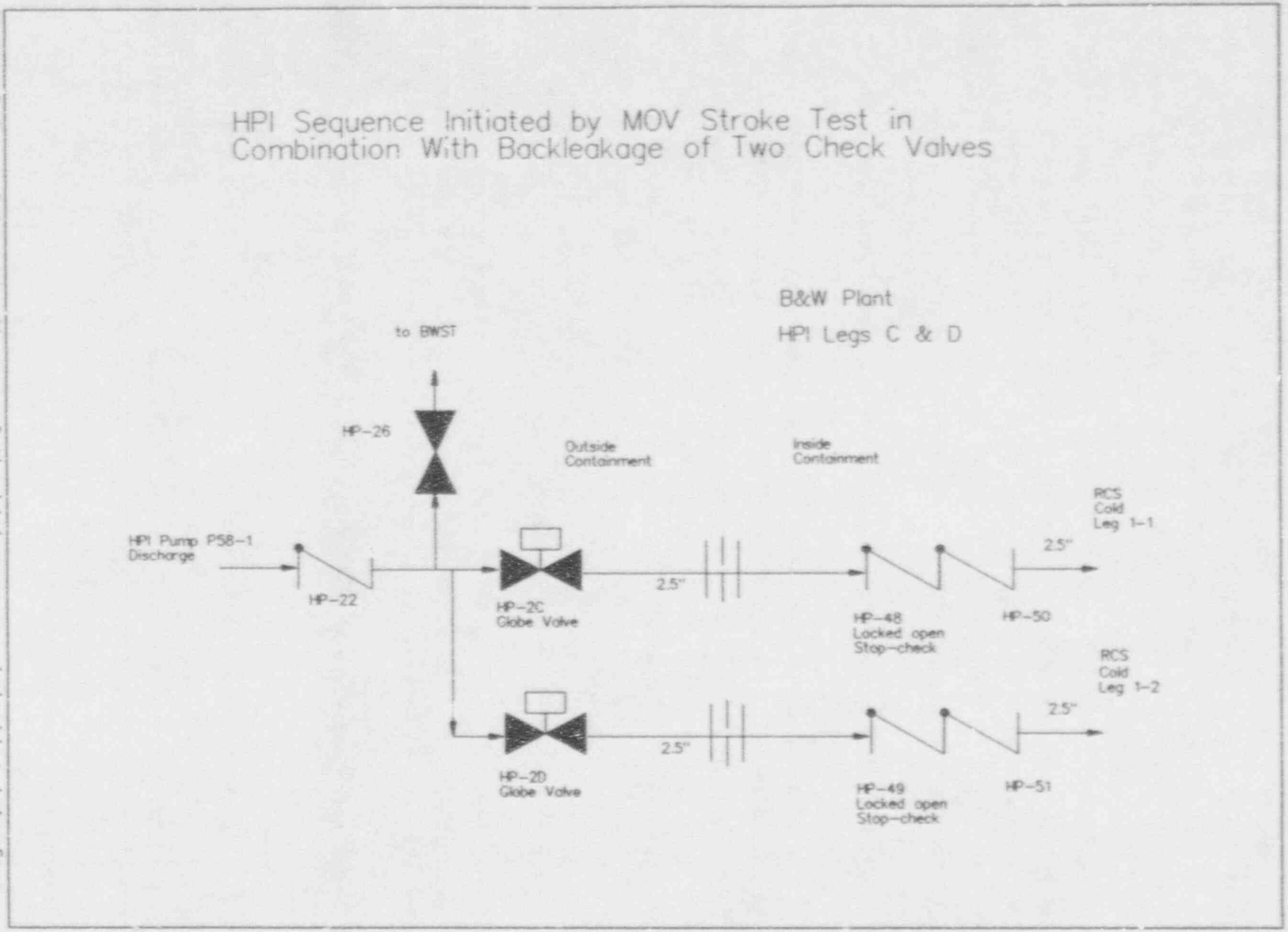


Figure D-3. Schematic diagram of the high pressure injection interface.

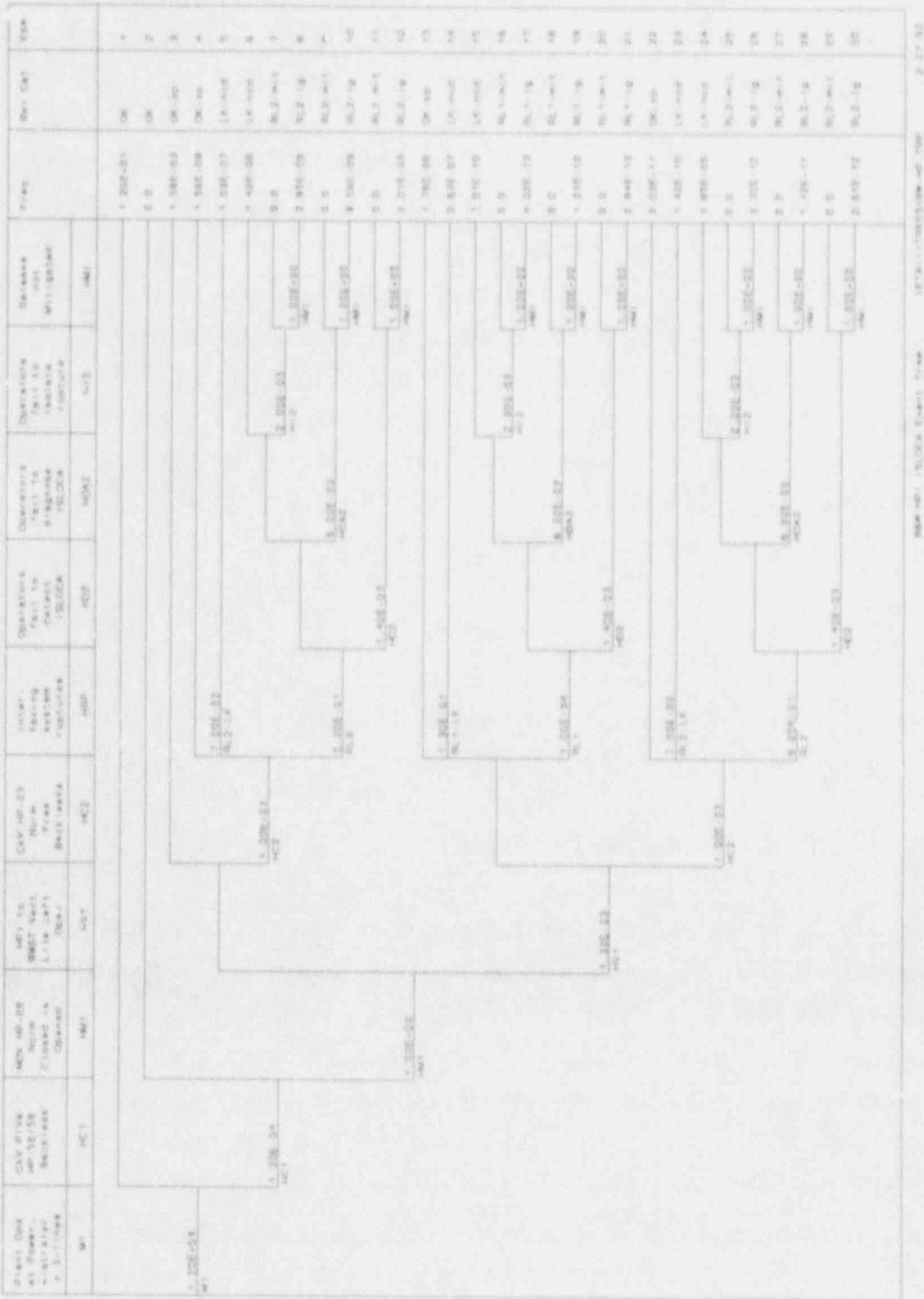


Figure D-4. High pressure injection ISLOCA sequence event tree.

HC1-HP - Pressure Isolation Check Valves HP-56/58 Backleak. 1.3E-4

Although there are two check valves inside containment in each injection line, these valves are welded together and physically coupled such that they cannot be individually leak tested. As stated in the description of the MU&P event tree, each check valve pair is treated as a single valve in the calculation of the backleakage probability. The reverse leakage probability is taken from the LER summaries and is estimated at 5.8E-7/hour (see Appendix B). Where possible, the LER valve failures were qualified as either a large leak or a small leak, with only 3% classified as large leaks (50 gpm was typically used to define the threshold between large and small leaks). However, given the ambiguous nature of the qualification and the uncertainty as to whether the LERs comprise a complete set of data, a conservative large-leak fraction of 10% is used here. The large-leak failure rate of 5.8E-8/hour is then multiplied by 2190 hours/quarter to generate a quarterly reverse leakage failure probability of 1.3E-4.

HM1-HP - MOV HP-2B(C,D) Normally Closed is Opened. 1.0

The HEP value of 1.0 is based on the routine quarterly stroke tests of MOVs HP-2B, C, and D as directed by procedure.

HV1-HP - HPI to BWST Vent Line Left Open. 0.0013

This event models the possibility that the 3-in. recirculation line (MOVs HP-26 or HP-27, and HP-29) is open at the beginning of the stroke test. This line is used for the quarterly flow tests of the HPI pumps. It is therefore possible that this line is left open after the pump test and, along with a preexisting failure of the PIV check valves (HP-58 and HP-59), allows RCS water to flow back to the BWST when the HPI MOV (HP-2B) is stroke tested. An HRA event tree was used to model the series of events that could lead to the EO leaving these two valves open after a pump test. Included is the communication between the RO and EO and the potential for recovery factors such as the SS (or other control room personnel) verifying the position of these valves after test completion. THERP values were used to quantify the event (see Appendix E, Figure 13).

HC2-HP - Check Valve HP-23, Normally Free, Backleaks. 1.0E-3

If the PIV check valves fail open, and the HPI MOV HP-2B is stroke tested, the HPI pump discharge check valve, HP-23 (22), must close in order to prevent overpressurizing vulnerable portions of the system. Because the HPI pump is flow tested quarterly the check valve periodically sees flow through it, but is normally in the "free" state. That is, most of time there is no flow and no differential pressure across the valve. Therefore, in a situation that exposes the valve to reverse flow, it is demanded to close and isolate the HPI pump from the RCS. The failure probability is simply the estimated probability that a check valve fails to close on demand (from Appendix B).

HRP-HP - Interfacing System Ruptures. 0.92/0.07 - 1.0E-4/0.13

This event models the conditional probability that, if given portions of the system are overpressurized, they will rupture. The two sets of values are for the HPI pump suction piping and the recirculation line to the BWST, respectively. Similarly, each value of the pair represents the probability that the rupture will be large or small, respectively. These numbers were obtained by first performing RELAP5 analyses of the HPI system to identify the pressures seen by the different portions of the system upon ingress of RCS water (Appendix F). These local system pressures are then compared to the estimated failure pressures of the system components (from Appendix G) in a Monte Carlo simulation using the EVNTRE computer code. The branch probabilities are taken as the fraction of Monte Carlo observations that resulted in large, small, or no ruptures in the HPI system (see Appendix H for the details of this calculation).

HD2-HP - Operators fail to detect ISLOCA. 0.0014

A number of indicators are available that provide status information on the interfacing systems to the control room operators. The operator's ability to detect this ISLOCA sequence is based on the successful recognition of 2 of 4 computer alarms (flow indication P-465, high temperature T-464, RAD-FA, and Auxiliary building sump) and 1 of 2 annunciators (falling pressurizer level or opening of relief valve 1510 or 1511). No credit was given for flow indicators (HP-3-C-1) registering reverse flow in the analysis. All failure probabilities were derived from THERP. Note that this event does not include the process by which the operators diagnose the situation (see the next event description). All that is included in the HEP for HD2-HP is detection of overpressurization of an interfacing system, not identification of the cause or the corrective actions (see Appendix E, Figure 14, Table E14).

HDA2-HP - Operators fail to diagnose ISLOCA. 0.006

This event has the same description as that for HDA2-MU. THERP values were used to quantify the implementation of the RCS small break procedure, and to quantify the ability of the crew to identify the signature of events as an ISLOCA (see Appendix E, Figure 11, Tables E12 and E13).

HI2-HP - Operators Fail to Isolate ISLOCA. 0.002

After the operators become aware of an abnormal situation, they must diagnose the cause and initiate corrective actions. This event models the probability that they will fail to isolate the break. The HEP estimation includes consideration of the time available for the operators to take the appropriate corrective action (i.e., the time to core uncovery, see Appendices G and H). The probabilities used were derived from THERP and were determined in the same manner as that for HI2-MU (see Appendix E, Figure 16).

HMI-HP - Operators fail to mitigate release. 1.0

Many things determine the potential for mitigating a possible radioactive release from an ISLOCA. These include: location of rupture, submergence of the break, presence and operation of fire suppression sprays, design of the auxiliary building (water tight doors, flow paths to the environment, etc), conditions at the time of core degradation (i.e., temperature of the pipes, water and surfaces inside the aux-bldg), and effects of severe accident phenomena such as possible hydrogen generation and burning.

The results of the aux-bldg environmental analysis (see discussion of Break Sequence 5 in Appendix M) indicate that for this ISLOCA sequence, the break will likely be submerged. However, the calculations performed for this study only consider the time up to failure of all ECCS, and do not examine conditions at the time of core degradation. Therefore, given the uncertainties associated with the parameters mentioned above, the initial assumption used the bounding situation that all releases are unmitigated. Once risk was calculated this assumption was reexamined. Based on the relatively low ISLOCA risk results (i.e. 6 person-rem/Rx-year), it was deemed prudent to not expend additional effort on this issue.

D.4. DHR Letdown Interface (Shutdown) Event Tree - DHR-SD

Once plant shutdown has been initiated, the control room operators monitor the primary system pressure and temperature in order to ensure adherence to the limits and requirements governing shutdown (e.g., at the Reference B&W plant the cooldown rate is limited to 100°/hr for temperatures above 270°F and 50°F for temperatures below 270°F). When the RCS temperature and pressure are reduced to approximately 280°F and 266 psig respectively, DHR operation is initiated. Figure D-5 shows a schematic diagram of the interface between the DHR letdown and the RCS. The ISLOCA event tree for this interface is shown in Figure D-6. The scenario of concern here begins with the premature opening of the DHR letdown line (MOVs DH-11 and DH-12). This action is based on the unlikely premise that shutdown has begun and that the control room operators misjudge the need for DHR, misread the cooldown curve, misinterpret the system indicators, misunderstand the procedures and instructions, etc. The pressure and temperature of the RCS will be anywhere from 2200 psi and 600°F to 266 psi and 280°F. The lower end of the pressure range would seem more likely in those cases where plant shutdown proceeds expeditiously, while the high end of the range might be possible if the plant has spent an unusually long amount of time in hot standby or there was some external constraint that necessitated a quick shutdown.

A second area of interest relates to the plant procedures for initiating DHR operations. The two DHR letdown MOVs (DH-11 and DH-12) are interlocked with RCS pressure such that they cannot be opened if the RCS pressure is above 301 psi for DH-11 and 266 psi for DH-12. If DH-12 will not open, the procedure allows the operators to jumper-out the relays in order to bypass the interlock. Because this action is procedurally sanctioned, the potential exists that the operations crew could jumper-out these relays when such an action is not warranted.

M3-SD - Plant Cooldown Mode-3 (Shutdown). 1.0

An orderly and controlled plant shutdown that requires operation of the DHR system is assumed to occur, on average, once a year. This presents the opportunity for the DHR shutdown interfacing system LOCA sequence. This sequence is based on the premise that the control room operators are

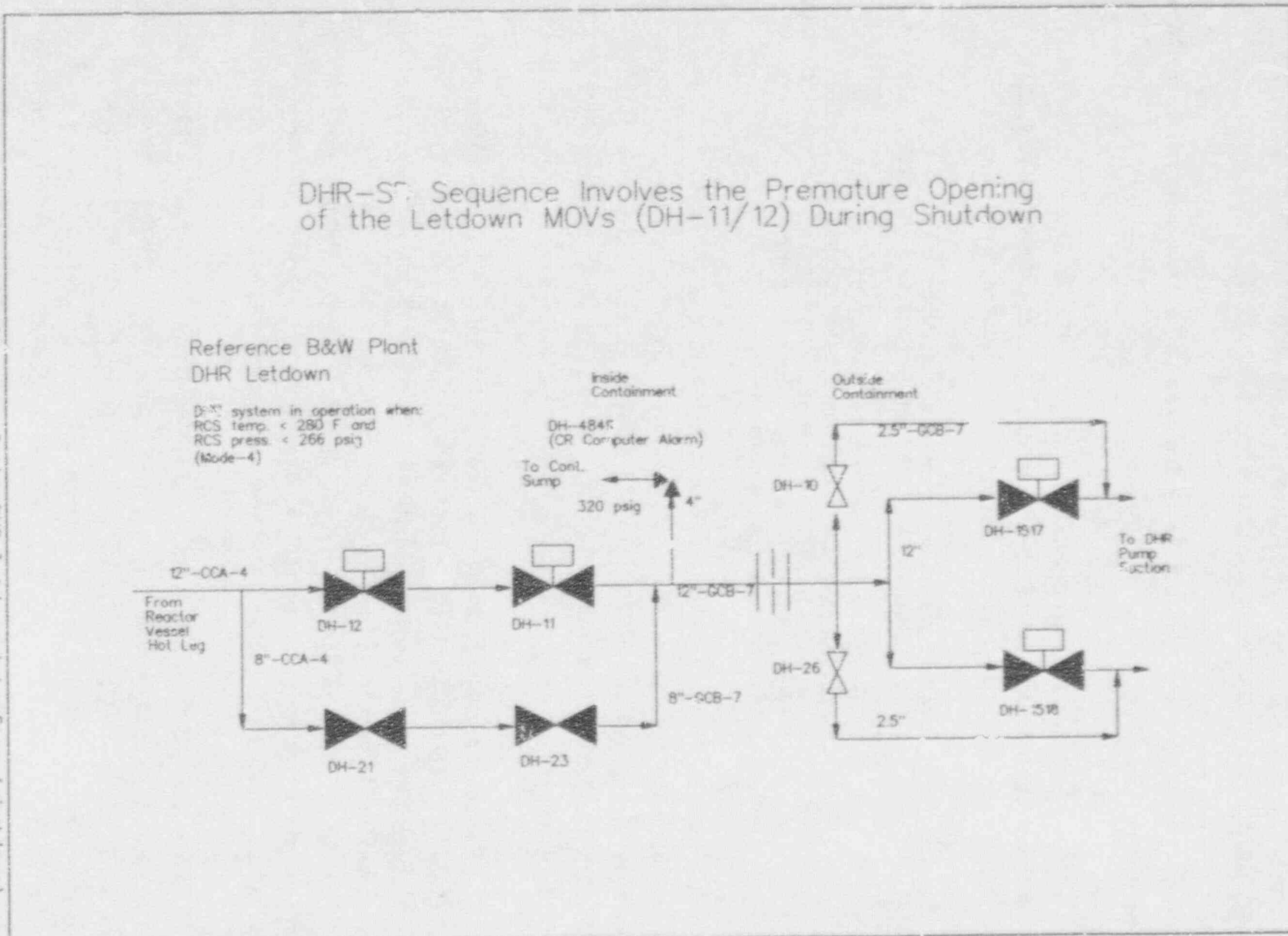


Figure D-5. Schematic diagram of the DHR letdown interface (shutdown).

susceptible to the human error of commission of entering DHR cooling prematurely (i.e., when RCS pressure is still above 300 psi).

DM1-SD - Operators Open DH-11 & DH-12 Too Soon While Transitioning to Shutdown Conditions. 0.00065

The first failure event provided in Figure 18 (in Appendix E), Fault Tree for Premature Opening of DH-11 and DH-12, "Operators Misread or Fail to Verify," represents the combined HEPs for the operating crew for:

1. incorrectly reading the RCS pressure indicator,
2. failing to verify that the DH permissive trip switch light is *not* lit,
3. recording information from the wrong instrumentation in the control room, and erring when comparing this information to the core cooling tables. Likewise, they obtain incorrect readings of system pressure and err in comparing correct information against the core cooling tables.

The HEP obtained for this failure event is negligible (i.e., < 1E-4) and so does not contribute appreciably to prematurely opening DH-11 and DH-12 and initiating an ISLOCA. The HEPs for this event were obtained from THERP Chapter 20, Tables 7, 9 and 10, which address selection and commission errors in using control room displays [see Appendix E, Figure(s) 17, 18, Table E15].

The second block in the fault tree (see Figure 17) models the operators decision to enter decay heat removal before temperature and pressure limits are acceptable. The cognitive action HEP for this block was determined by engineering judgement and reflects the possibility of a joint decision by the SS and RO. The basis for the HEP estimate includes sanctioned jumpering of interlocks which exist in current SD procedures. Allowance has been made for a refusal by the I&C technician during the execution of this procedure.

The basis for this estimation utilized the industry operating of zero occurrences in 1510 reactor-years (Rx-yr) experience. Using a Bayesian update of a noninformative prior yields a mean occurrence rate of 3.3E-4/Rx-yr (95% upper bound of 1.3E-3/Rx-yr). After modifying this rate for the specific context of the B&W reference plant (as described above), an adjusted rate of 6.6E-4/Rx-yr was estimated.

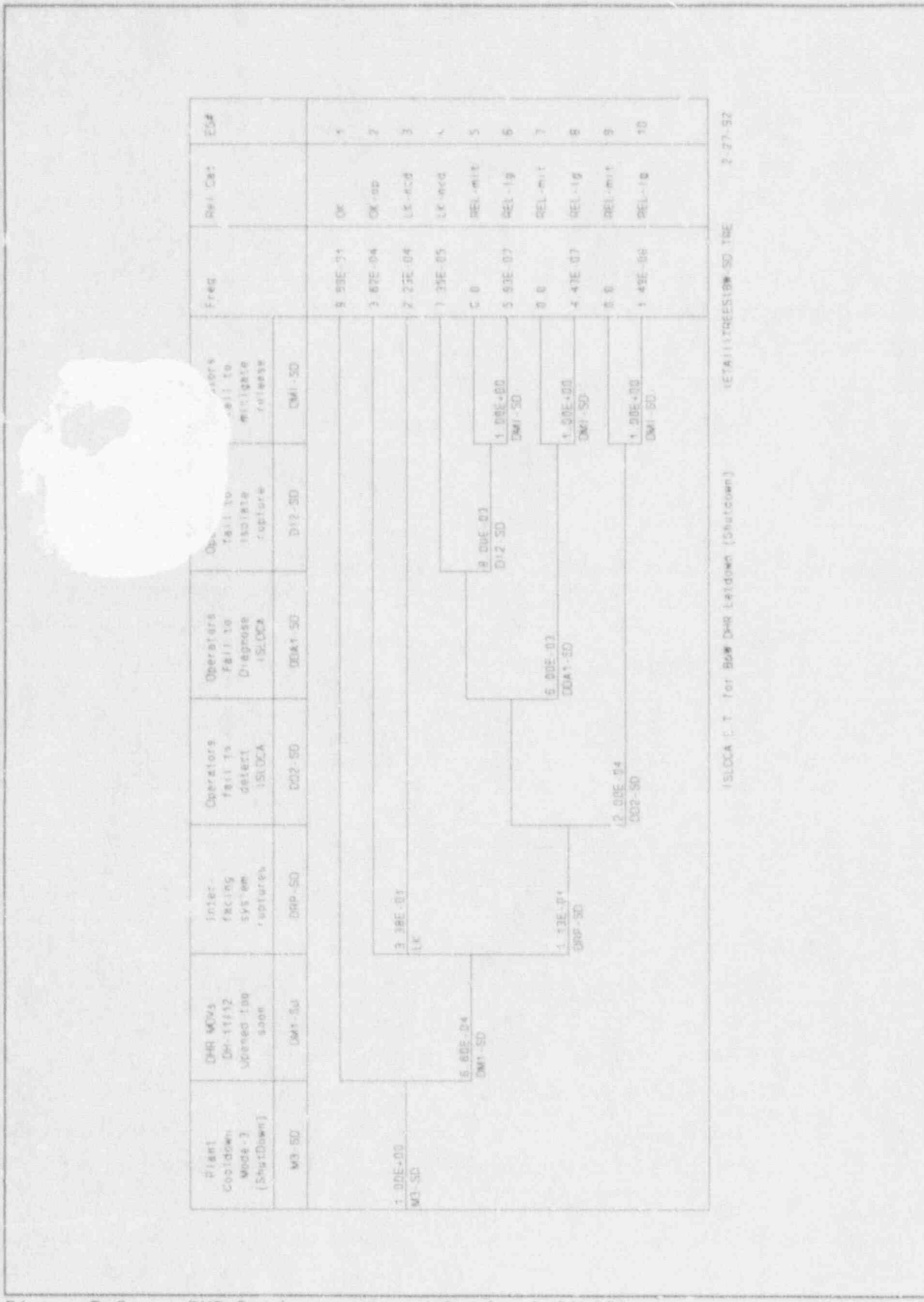


Figure D-6. DHR letdown system (shutdown) ISLOCA sequence event tree.

DRP-SD - Rupture of Low-Pressure System Components. **0.11/0.34**

This eve · represents the probability that, given the DHR letdown valves are opened prematurely, the pressure in the interfacing system exceeds the failure pressure of the system components. This is a function of the RCS pressure at which the premature entry into DHR was made. If the premature entry into DHR is made when RCS conditions have not quite reached the set points required by procedures, there should not be a problem. Table D-1 shows the weighting scheme used for allocating the HEP for prematurely entering DHR cooling as a function of RCS pressure. Table D-2 lists the effect of this weighting on the HEP and the probability of producing a rupture in the interfacing system. The values listed above are the aggregated probabilities that the rupture will be either a large rupture or a small leak, respectively.

Table D-1. Relative weighing of HEP as a function of RCS pressure. (regression used for estimating pressure-dependent HEPs)

RCS Press	Relative HEP	Log(HEP)	Regression Output:	
			Constant	0.666667
			Std Err of Y Est	1.3E-8
2200	0.001	-3	R Squared	1
1600	0.01	-2	No. of Observations	4
1000	0.1	-1	Degrees of Freedom	2
400	1	0	X Coefficient(s)	-0.00167
			Std Err of Coef.	9.6E-12

A RELAP5 model was constructed of the interfacing system in order to estimate the local pressures that would be seen by the various downstream components. The RELAP5 calculations were performed for the range of RCS pressures from 400 to 2100 psig in 100 psig increments (see Appendix F). The local interfacing system pressures were then compared to the estimated failure pressures. The estimated failure pressures were calculated in an independent analysis by IMPELL Corporation (NUREG/CR-5603). A Monte Carlo simulation was used to determine if and where ruptures would occur (described in Appendix H). In each Monte Carlo observation, the RCS pressure was converted to a local system pressure using an empirically derived equation. Both the RCS pressure and the rupture pressures for each system component were randomly sampled from the postulated distributions (i.e., a normal distribution for RCS pressure and lognormal for the failure pressure). The two resulting values were compared and if the system pressure exceeded the failure pressure, the component was assumed to fail. If not, no failure was assumed. The probabilities listed for this event (0.11, 0.34, and 0.55) represent the fraction of the 10,000 Monte Carlo observations that resulted in large, small, and no ruptures, respectively. The system rupture probability at each RCS pressure was weighted by the probability that the valves are opened at that particular pressure by the operators (see Section 4.6.2.1 for further discussion on this).

Table D-2. DHR system rupture probabilities (weighted by the HEP of prematurely opening DH-11/12) as a function of RCS pressure.

RCS Pressure (psig)	HEP	System Rupture Probability			HEP-Weighted System Rupture Probability		
		large	small	no-leak	large	small	no-leak
2200	2.1E-07	1	0	0	2.1E-07	0.0	0
2100	3.1E-07	0.999	0.001	0	3.1E-07	4.3E-10	0
2000	4.5E-07	0.997	0.003	0	4.5E-07	1.1E-09	0
1900	6.7E-07	0.995	0.005	0	6.6E-07	3.1E-09	0
1800	9.8E-07	0.994	0.006	0	9.7E-07	6.3E-09	0
1700	1.4E-06	0.991	0.009	0	1.4E-06	1.3E-08	0
1600	2.1E-06	0.983	0.017	0	2.1E-06	3.7E-08	0
1500	3.1E-06	0.964	0.036	0	3.0E-06	1.1E-07	0
1400	4.5E-06	0.920	0.080	0	4.2E-06	3.6E-07	0
1300	6.7E-06	0.836	0.164	0	5.6E-06	1.1E-06	0
1200	9.8E-06	0.705	0.295	0	6.9E-06	2.9E-06	0
1100	1.4E-05	0.551	0.449	0	7.9E-06	6.4E-06	0
1000	2.1E-05	0.403	0.597	0.0001	8.5E-06	1.3E-05	2.1E-09
900	3.1E-05	0.281	0.718	0.001	8.7E-06	2.2E-05	2.5E-08
800	4.5E-05	0.178	0.810	0.012	8.1E-06	3.7E-05	5.3E-07
700	6.7E-05	0.100	0.809	0.091	6.7E-06	5.4E-05	6.1E-06
600	9.8E-05	0.050	0.580	0.370	4.9E-06	5.7E-05	3.6E-05
500	1.4E-04	0.021	0.193	0.786	3.1E-06	2.8E-05	1.1E-04
400	2.1E-04	0.007	0.012	0.981	1.4E-06	2.6E-06	2.1E-04
	6.6E-04				0.113	0.338	0.548

DD2-SD - Operators fail to detect ISLOCA. 2E-4

This event represents the failure of the operating crew to correctly integrate computer alarms and control room annunciators as indicators of an ISLOCA. This failure occurs after an ISLOCA has been initiated. The HEP for this event includes modeling of key computer alarms (T-362 and sump computer alarm) and control room annunciators (for relief valve open and containment sump alarm). See Appendix E, Figure 19 and Table E16.

DDA1-SD - Operators fail to diagnose ISLOCA (DHR-ShutDown). 0.006

In this scenario, the system failures lie outside the scope of the Loss of DHR procedure. The key events for this scenario involve failure of the crew to determine the fault lies in an area not addressed in the procedure and, troubleshooting to find the fault. The operators would need to determine that an ISLOCA involving the DHR system was in progress, independent of procedural guidance and using control room indications. Although the Loss of DHR procedure will not be of direct utility in isolating critical points in the system to mitigate the ISLOCA, it will help the crew to determine which points in the system are *not* faulted. Troubleshooting outside of the procedure is necessary to identify the faulted points and to determine the flow path through which inventory is being lost. The HEPs for this event were obtained from THERP tables in Chapter 20. See Figure 21 and Tables E17 and E18 in Appendix E.

DI2-SD - Crew Fails to Isolate ISLOCA (DHR-ShutDown). 0.008

This event represents the failure of the crew to isolate the flow path in the system through which RCS leakage is occurring. A prerequisite for this event is the successful identification or determination that the system leak is occurring through motor-operated valves DH-11 and DH-12. Failure to close DH-11 or DH-12 will cause failure to isolate the DHR leak path. These valves can be closed either from the control room or from the panels where they were jumpered (see Appendix E, Figure 22, Table E19).

The HEPs for the control room action correspond to two independent selection errors (i.e., the operator incorrectly presses two switches near the two desired switches) modified for the effect of stress. The HEPs for the control room error in selecting the wrong controls were obtained from THERP Chapter 20 tables for commission errors in selecting a control.

DMI-SD - Operators Fail to Mitigate the Release. 1.0

Many things determine the potential for mitigating a possible radioactive release from an ISLOCA. These include: location of rupture, submergence of the break, presence and operation of fire suppression sprays, design of the auxiliary building (water tight doors, flow paths to the environment, etc), conditions at the time of core degradation (i.e., temperature of the pipes, water and surfaces inside the aux-bldg), and effects of severe accident phenomena such as possible hydrogen generation and burning.

The results of the aux-bldg environmental analysis (see discussion of Break Sequences 1, 2, and 3 in Appendix M) indicate that for this ISLOCA sequence, the break will likely be submerged. However, the calculations performed for this study only consider the time up to failure of all ECCS, and do not examine conditions at the time of core degradation. Therefore, given the uncertainties associated with the parameters mentioned above, the initial assumption used the bounding situation that all releases are unmitigated. Once risk was calculated this assumption was reexamined. Based on the relatively low ISLOCA risk results (i.e. 6 person-rem/Rx-year), it was deemed prudent to not expend additional effort on this issue.

D.5. DHR System Letdown Interface (StartUp) Event Tree - DHR-SU

The DHR system may be overpressurized if the DHR letdown line remains open while the RCS is being heated up and pressurized. A schematic diagram of the DHR interface with the RCS is shown in Figure D-7 and the ISLOCA event tree for this system is shown in Figure D-8. There are two ways in which RCS water can enter the DHR system. One way is via the normal letdown MOVs DH-11 and DH-12. Another way is via the MOV bypass valves DH-21 and DH-23. These are manual locally-operated valves. Although DH-11 and 12 are interlocked to automatically close when the RCS pressure is above 300 psig, the valves always have their control power removed (as required by technical specifications) to prevent inadvertent operation, thus defeating the closure interlock.

M3-SU - Plant Heatup. 1.0

This event represents the occurrence of plant heatup, which takes place with the reactor subcritical. Mode 3 operations cover the range from approximately 280°F and 200 psig, to about 500°F and 2200 psig. Heatup is primarily accomplished using the pressurizer heaters to increase RCS temperature and pressure. (At approximately 500°F and 2150 psig, reactor power is raised to about 5% and the plant goes through startup operations, Mode 2, in anticipation of entry into Mode 1, power operation.) If the plant has just completed an extended outage, the heatup procedure specifies a number of hold points at which periodic surveillances and tests are performed. However, if the outage was brief, most of these items can be omitted and the transition to Mode-2 can be accomplished relatively quickly. Because a plant trip does not necessarily require operation of the DHR cooling system, an estimated average of one startup per year is used for this event.

DM1-SU - DHR Letdown MOVs DH-11 and DH-12 are Left Open. 0.0002

This event models the probability that the DHR system letdown isolation valves, DH-11 and DH-12, are inadvertently left open during plant startup. Normal plant procedure at the Reference B&W Plant is to maintain the valves in a disabled state by removing their control power (thus, defeating the safety feature of the interlock). This is done during power operation to prevent inadvertent opening, and during plant shutdown to prevent inadvertent closure

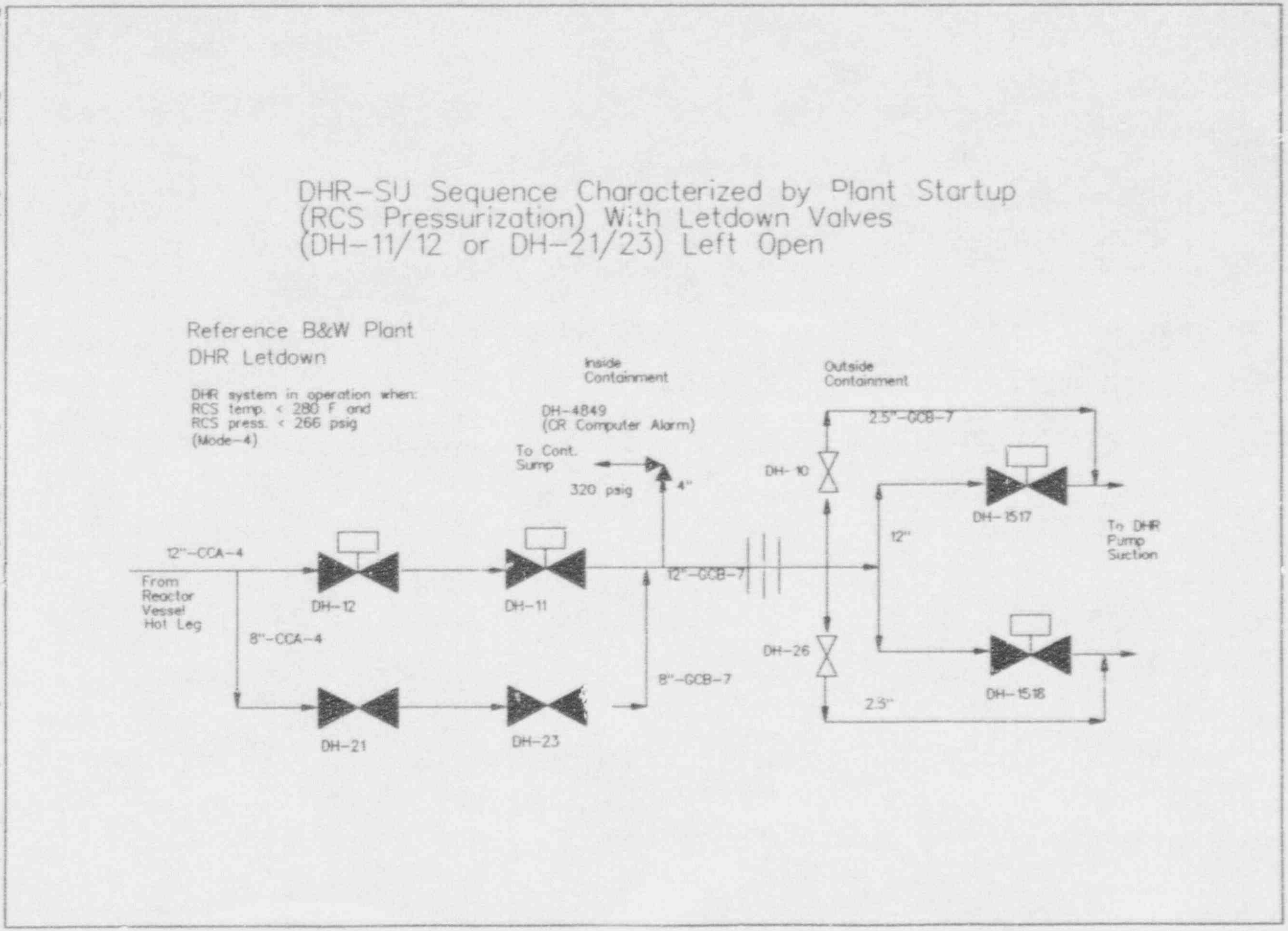


Figure D-7. Schematic diagram of the DHR letdown interface (startup).

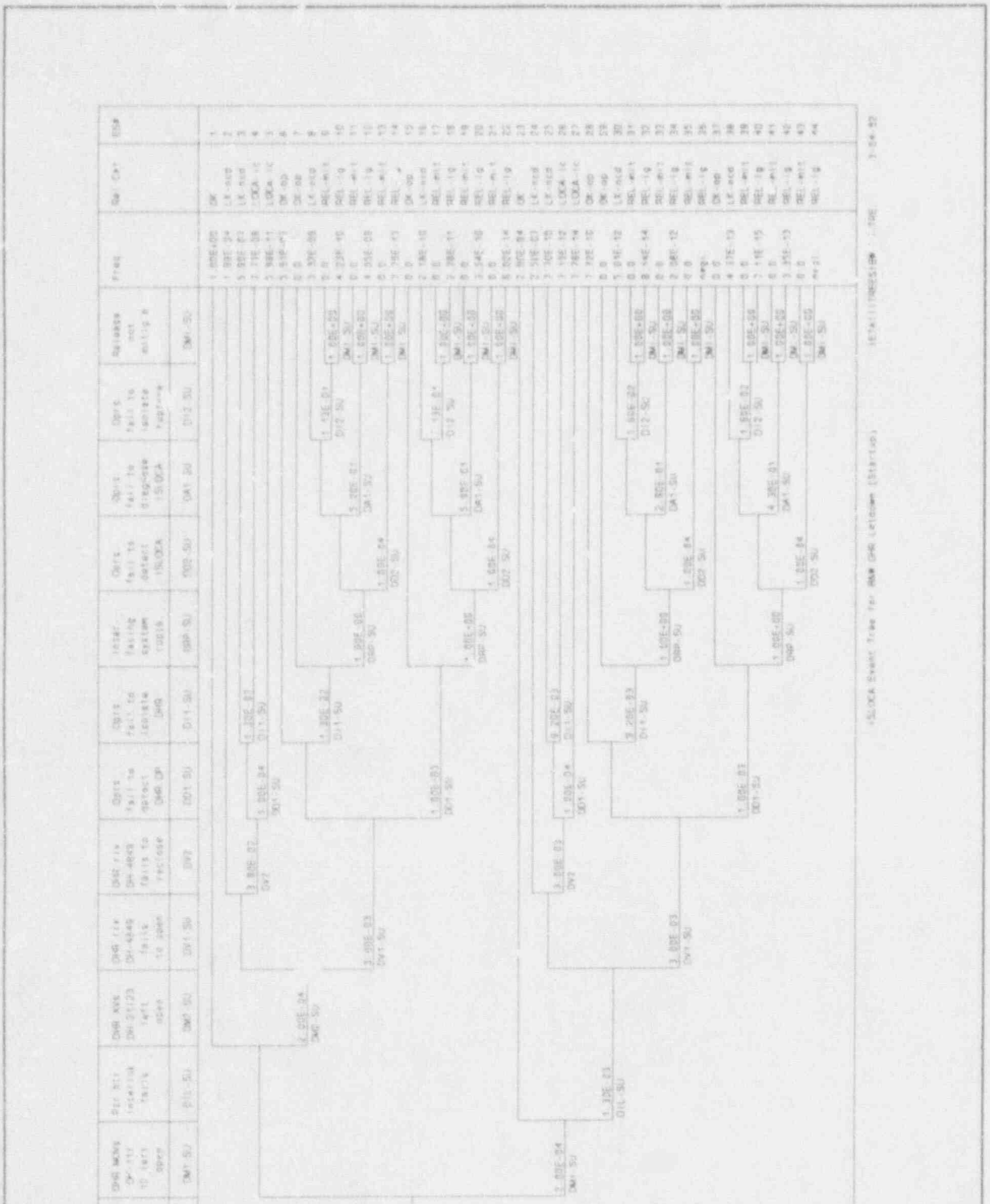


Figure D-8. DHR letdown (startup) ISLOCA sequence event tree.

that would isolate the DHR system. The interlock affording overpressure protection is that for the pressurizer heaters. The failure of this interlock is modeled as part of this event sequence. The control power to the valves is energized only when the valves are to be operated. The startup procedure also requires that the crew verify the position of these two valves and an independent sign off by a separate operator. This startup event is quantified using values from THERP (see Appendix E, Figure 23, 24, and 25, and Table E20).

DIL-SU - Pressurizer Heater Interlock Fails. 1.3E-3

Although DH-11 and 12 are not capable of automatically closing (control power is always removed), the valves are interlocked with the pressurizer heaters such that if the valves are open and the RCS pressure rises above 300 psig, the heaters will not operate. Tripping the pressurizer heaters will prevent pressurization of the RCS above 300 psig. This event models the probability that the interlock fails to trip the pressurizer heaters, and is quantified using a fault tree development that accounts for both hardware and miscalibration faults. The fault tree is shown in Figure D-9 and is quantified using data from Appendix B.

DM2-SU - DHR Bypass Manual Valves DH-21 and DH-23 Left Open. 0.0002

This event models the probability that valves DH-21 and DH-23 are left open following their use during a shutdown. Because these are locally operated valves that are normally locked closed, the likelihood of operators suspecting them to have been left open is assumed to be small. Opening these valves is necessary to stroke test valves DH-11 and DH-12 (which is done while the plant is shutdown). These valves have no remote position indication or hardware control. They are administratively controlled. Communication is a key factor in operators not approaching startup with DH-21 and DH-23 in the open position. The event was modeled to include possibilities for recovery prior to startup. Quantification is bases on THERP, Tables 7, 13, and 22 from Chapter 20 (see Appendix E Figure 28 and Tables E23 and E24).

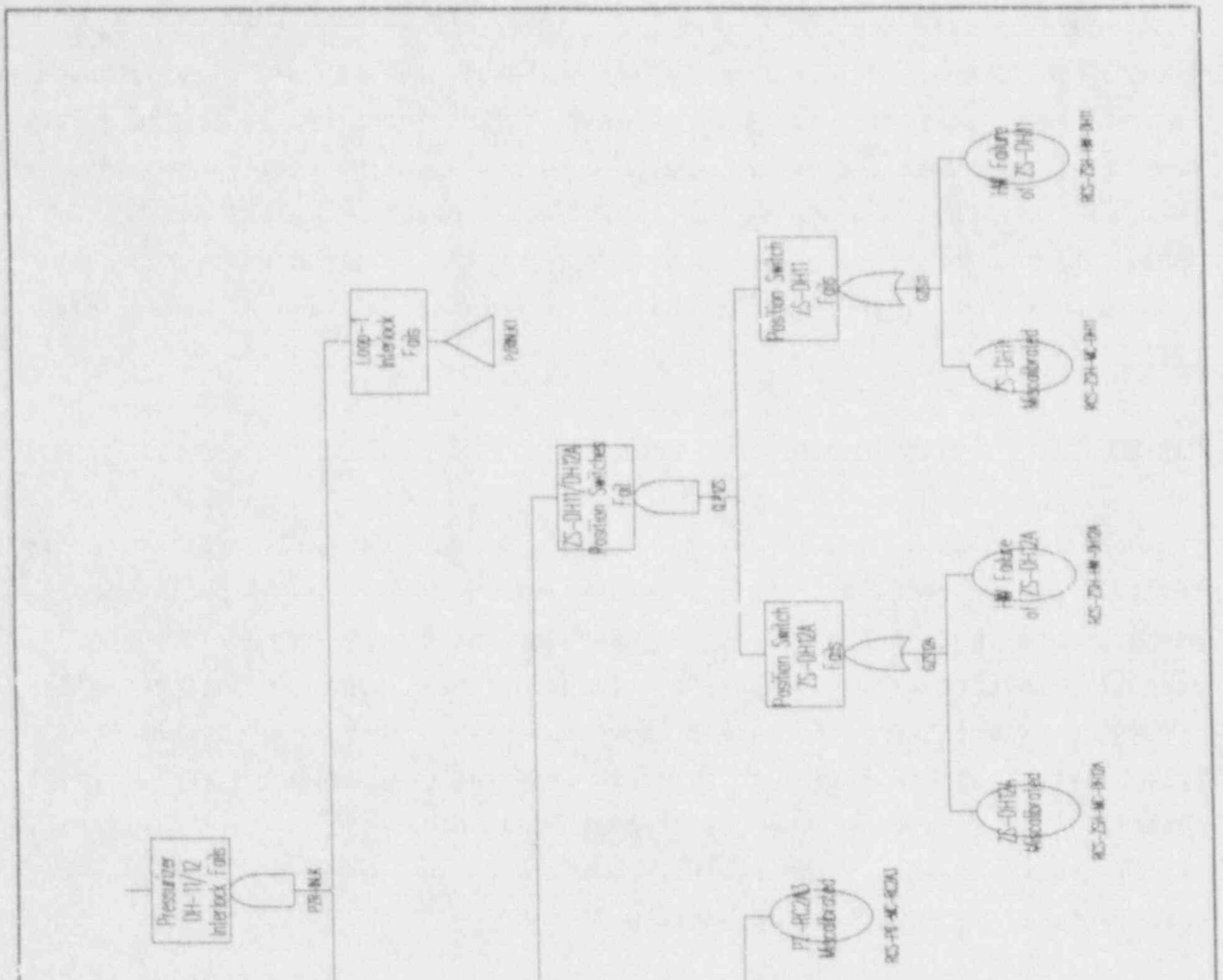


Figure D-9 (1 of 2). Pressurizer heater interlock fault tree.

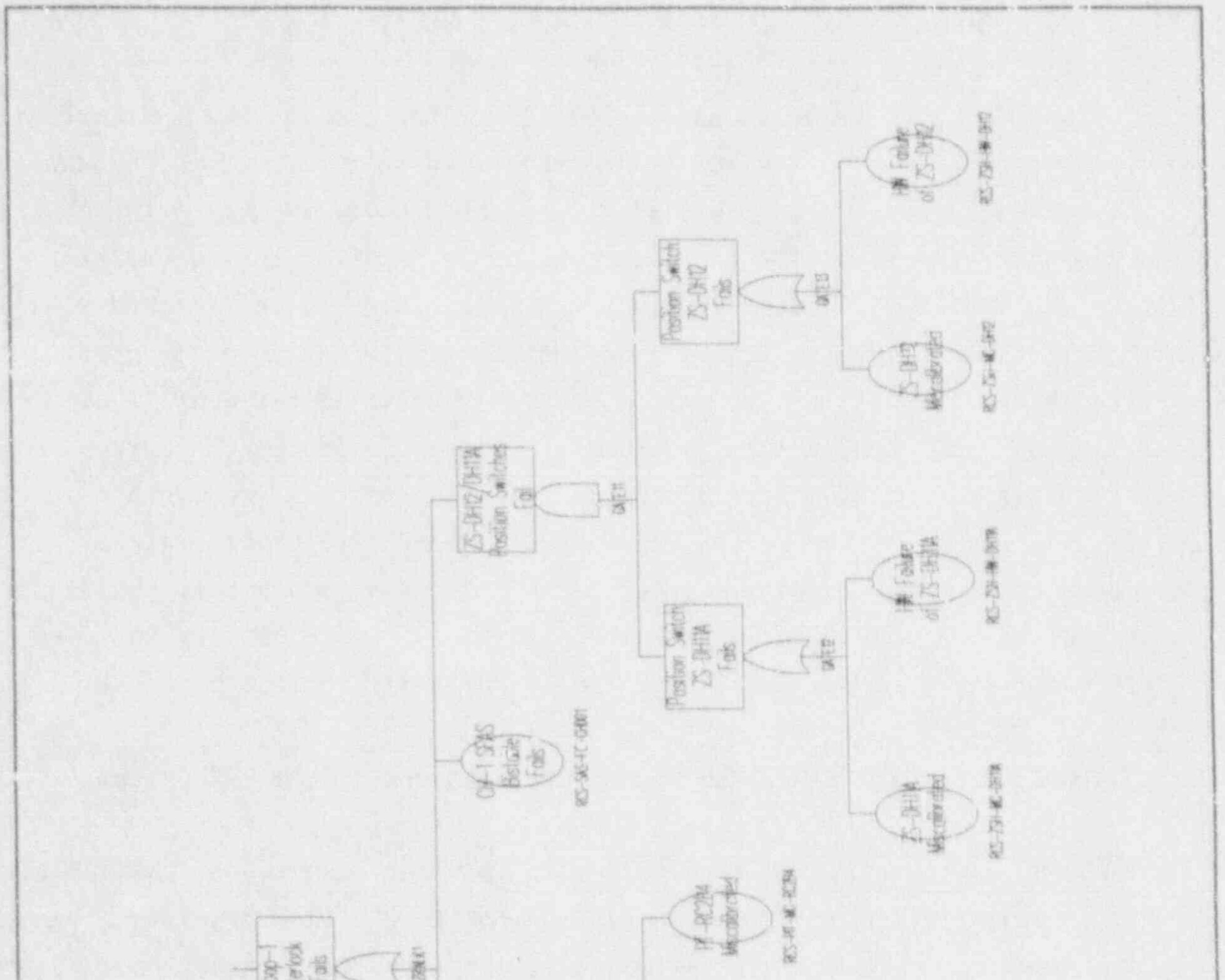


Figure D-9 (2 of 2). Pressurizer heater interlock fault tree.

DV1-SU - DHR Relief Valve, DH-4849, Fails to Open. 3.0E-3

The DHR relief valve is not capable of protecting the DHR system from being overpressurized by the RCS (4-in. relief valve on the 12-in. letdown line), but would provide a highly visible mechanism for informing the control room operators the situation was not normal. In addition to the outlet temperature indicator located in the control room, the relief valve discharges to the containment sump, which is also instrumented. Furthermore, upon opening at its setpoint of 320 psig, the relief valve is designed to pass approximately 1800 gpm, a rate that cannot be replenished by the makeup system. This rate of coolant loss would in turn produce a drop in pressurizer level. Therefore, if the relief valve were to open, the probability of detecting an abnormal condition prior to reaching a pressure that would challenge the DHR system integrity is very high. The probability that the relief valve fails to open is taken from the data listed in Appendix B.

DD1-SU-(A,C) - Operators Fail to Detect Overpressure in the DHR System. 1E-4

If the relief valve DH-4849 to the containment sump opens, the RCS will lose approximately 1,800 gpm to the containment sump. This provides clues to the operators for detection of the overpressure situation. Additionally, the pressurizer level will fall and activate an alarm because of the RCS inventory loss associated with the leak. Prior to rupture, high temperature alarm (T362) indication will be available on the inlet side of the DH pump. This information is presented in both the Reference B&W Plant computer alarm system present in the control room and on annunciator panel 3-4-A1. Event DD1-SU-A,C was modeled using HRA fault trees and quantified with values from THERP (see Appendix E, Figure 26, Table E21).

DD1-SU-(B,D) - Operators Fail to Detect Overpressure in the DHR. 1E-3

The second situation examines the case when DH-4849 fails to open on demand. In these scenarios, the operators must rely on less obvious indications to detect overpressure. The primary indication is the annunciated temperature alarm (DH8B/A) from the inlet side of the DH pumps. The failure rate is higher than that for DD1-SU-A,C because of the relatively short time frame for personnel to detect the overpressure prior to entry into the rupture phase of the sequence. This event was modeled using engineering judgement and quantified with values from THERP (see Appendix E, Figure 30, Table E26).

DI1-SU-(A,B) - Operators Fail to Isolate the RCS from the DHR System. 0.013

This event represents the possibility that operators fail to perform necessary isolation actions. DH-21 and DH-23 are local-manual valves whose positions can be verified only through local inspection of the valves (note that both pairs of valves are located inside containment). The opening of DH-4849 is credited with increasing the probability that the abnormal situation will be correctly diagnosed, but not that the actual isolation actions will themselves be any more or any less difficult. Modeling employed HRA fault trees and values were quantified from THERP. Quantification analysis assumed that communication was required between the RO and EO, that the EO has written down or is handed written instruction, and that personnel will be required to don anti-Cs (see Appendix E, Figures 29 and 32, and Tables E25 and E28).

DI1-SU-(C,D) - Operators Fail to Isolate the RCS from the DHR. 0.0092

Operator actions required to close DH-11 and DH-12 are straightforward. Personnel must energize the control circuits and close either DH-11 or DH-12 in order to isolate the RCS from the DHR. Modeling accounted for the fact that control actions for these valves may be taken from the control room and that instrumentation exists for both valve position indication and control circuit status. THERP values were used to estimate failure probabilities and fault tree logic designed to account for the potential of achieving isolation by closing only one of the two valves [see Appendix E, Figure 27, Table E22 (DI1-SU-C), and Figure 31, Table E27 (DI1-SU-D)].

DRP-SU - Rupture of the Interfacing System. 1.0

The evaluation of previous events on the DHR-SU event tree, included consideration that if an abnormal condition was detected, RCS pressurization would be interrupted while investigations were performed. Therefore, the failure scenarios implicitly include the continued pressurization until a rupture occurs in the DHR/LPI system. Consequently, this event is assigned a probability of 1.0. As a point of reference, the median large-rupture failure probability of the DHR/LPI system occurs at an RCS pressure of about 1100 psig (note that the local pressure in the DHR/LPI system is only 65-95% of the RCS pressure, depending on the exact location within the system).

DD2-SU-(A,B,C,D) - Operators Fail to Detect ISLOCA. **1E-4**

This event is modeled in much the same manner as DD1-SU (A,C) for overpressure detection with the following exceptions. A longer time horizon is available for operators to detect the same indication, the potential for noticing PZR decrease is greater, and an EO may be able to identify water in one of the equipment rooms housing the DH pumps. Failure probabilities were obtained from THERP (see Appendix E, Figure 33 and Table E29).

DA1-SU-(A,B,C,D) - Operators Fail to Diagnose ISLOCA. **(see below)**

This event refers to personnel actions and cognitive activities subsequent to rupture. An ISLOCA has occurred and the degree to which crews will be able to (a) successfully diagnose the event, and (b) determine the appropriate location from which to take isolation actions, rests on two major assumptions. The first is that it will be much more difficult to detect the involvement of locally operated valves DH-21 and DH-23 than it will be to read the indication present in the control room for MOVs DH-11 and DH-12.

Secondly, the persistence of the containment sump level, temperature, and relief valve DH-4849 open indications will provide more clues than will be the case for scenarios wherein the relief valve fails to open. This is because relief valve failure would not involve inventory discharge to the containment sump. Thus, operators would not receive those alarms associated with inventory in the containment sump prior to the occurrence of a rupture.

Engineering judgement was used to quantify these events. HEP calculated values are as follows:

DA1-SU-A 21&23, RVO = 0.52
DA1-SU-B 21&23, RVC = 0.59
DA1-SU-C 11&12, RVO = 0.29
DA1-SU-D 11&12, RVC = 0.43

(See Appendix E, Figures 33 to 36).

DI2-SU-(A,B) - Operators fail to isolate ISLOCA. 0.113

The actions required by control room personnel to isolate the ISLOCA are influenced by working in a moderately high stress environment. The equipment operators will have to wear anti-Cs. Both groups of personnel will be in communication with one another. The probability of this event is estimated from a former HEP value for closing DH-21 and DH-23 to achieve RCS isolation (pre-rupture case) and the availability of access to containment where the valves are located. The failure probabilities were determined by a combination of THERP values and room access probabilities (see Appendix E, Figure 34 and Table E31).

DI2-SU-(C,D) - Operators Fail to Isolate ISLOCA. 0.016

For the DI2-SU-C&D scenarios, operators must energize the valve control circuits, and to close either DH-11 or DH-12 in order to achieve isolation. The modeling and quantification for this series of actions took into account the effects of stress and dependence associated with these actions. The HEP values were determined from THERP (see Appendix E, Figure 35 and Table E32).

DMI-SU - Operators fail to mitigate release. 1.0

Many things determine the potential for mitigating a possible radioactive release from an ISLOCA. These include: location of rupture, submergence of the break, presence and operation of fire suppression sprays, design of the auxiliary building (water tight doors, flow paths to the environment, etc), conditions at the time of core degradation (i.e., emperature of the pipes, water and surfaces inside the aux-bldg), and effects of severe accident phenomena such as possible hydrogen generation and burning.

The results of the aux-bldg environmental analysis (see discussion of Break Sequences 1, 2, and 3 in Appendix M) indicate that for this ISLOCA sequence, the break will likely be submerged. However, the calculations performed for this study only consider the time up to failure of all ECCS, and do not examine conditions at the time of core degradation. Therefore, given the uncertainties associated with the parameters mentioned above, the initial assumption used the bounding situation that all releases are unmitigated. Once risk was calculated this assumption was reexamined. Based on the relatively low ISLOCA risk results (i.e. 6 person-rem/Rx-year), it was deemed prudent to not expend additional effort on this issue.

D.6. Low-Pressure Injection System Interface Event Tree - LPI

A schematic diagram of the low-pressure injection (LPI) interface with the RCS is shown in Figure D-10. The ISLOCA event tree for this system is shown in Figure D-11. This interface represents the classical V-sequence configuration of two check valves in series, forming the pressure isolation boundary between the RCS and LPI system. The system comprises two redundant trains, with each injection line being shared with one core flood tank. Based on work performed on the failure of PIVs, BNL has concluded that PIV check valves on core flood tank discharge lines have experienced a higher failure rate than other check valves (note that this applies to check valves in standby service, see Appendix B).

M1 - Plant Operating at Power (Mode-1). **2.0**

The probability that the plant will be operating at power is conservatively quantified at 1.0. This is multiplied by (2) to account for the presence of the two LPI system injection lines.

LC1 - Backleakage of Pressure Isolation Check Valve CF-30. **7.6E-4**

This event models the random, independent failure of pressure isolation check valve CF-30. The failure mode of interest is the time-dependent (the valve is normally closed with a large differential pressure across it) probability that the valve will allow significant (> 200 gpm) backleakage. The check valve is leak tested whenever the plant has been shutdown and is returning to power. Therefore, failure-to-close events are not considered. A failure probability that applies particularly to core flood tank (CFT) discharge check valves is used to quantify this event. Because of the harsher environment and service the CFT discharge check valves experience, they have a higher failure rate than other check valves (8.7E-8/hr compared to 1.8E-8/hr, see Appendix B). Backleakage events smaller than 200 gpm are not considered, because such leak rates overpressurize the interfacing system slowly. This result is a high likelihood of detection and correction of the ISLOCA precondition before the LPI system integrity is challenged. A fault exposure time of one year (8760 hours) is used in estimating the probability of this event.

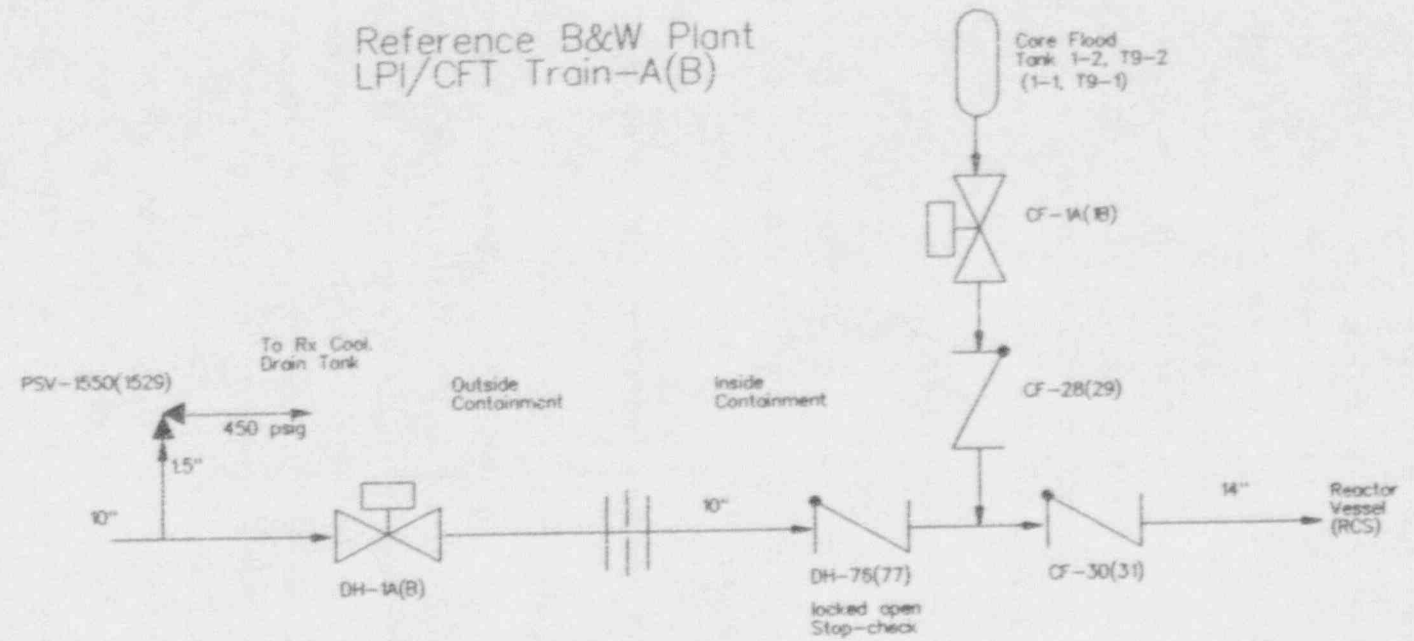


Figure D-10. Schematic diagram of the low pressure injection interface.

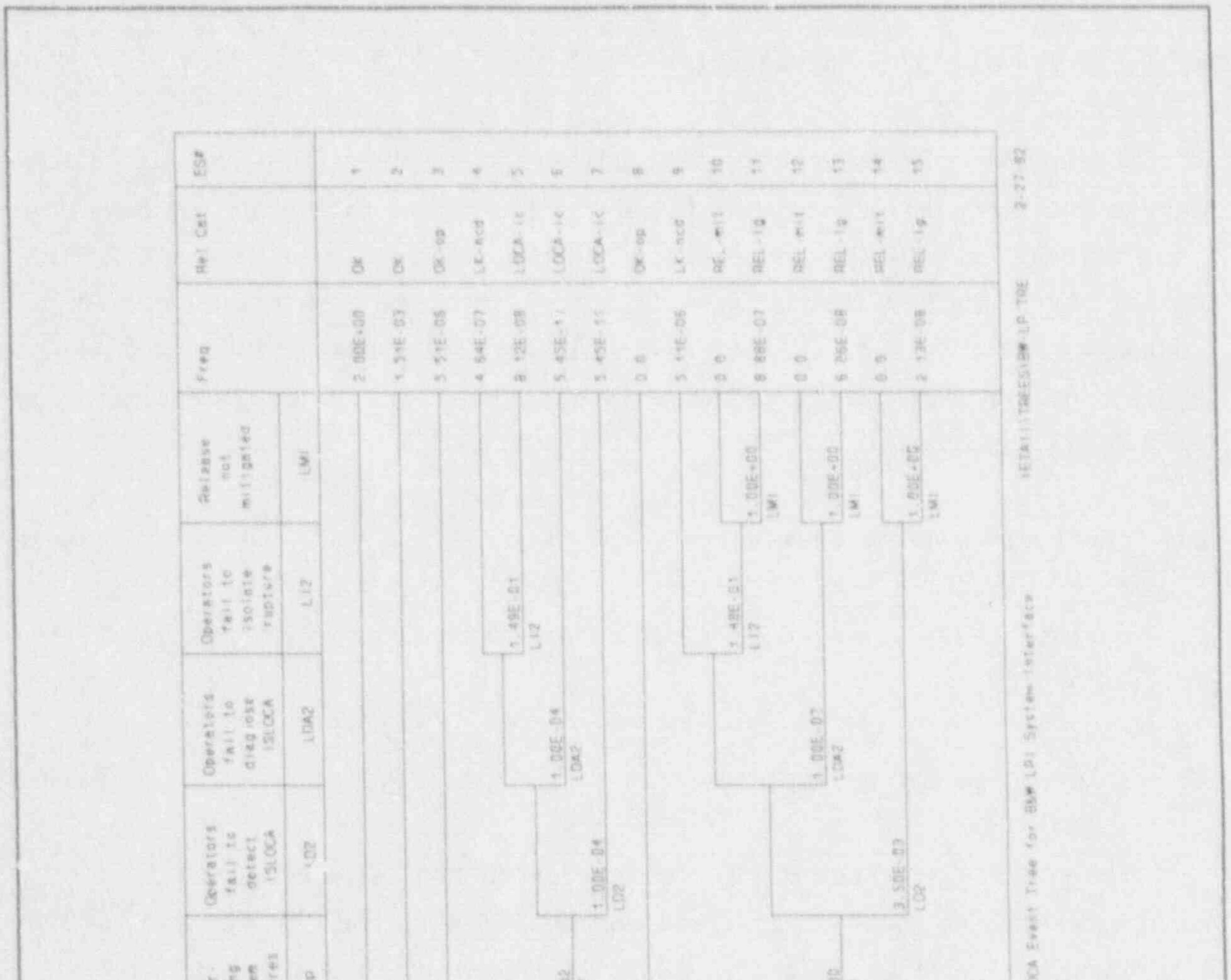


Figure D-11. Low-pressure injection ISLOCA sequence event tree.

LC2 - Check Valve DH-76 Backleaks. 4.0E-3

Check valve DH-76 is also leak tested; therefore, this event is assumed to have the same failure rate as CF-30 (event LC1) This event is modeled as a conditional probability and because the two failures are correlated, the probability of both valves failing is higher than the combination of two independent failure rates. The individual failure probability is 7.6E-4 (EF=10) and the probability of two valves failing is 3.0E-6, as estimated by Monte Carlo sampling.

LC3 - Check Valve CF-28 Backleaks. 4.0E-3

Because check valve CH-28 is also leak tested, event LC3 is quantified the same as LC2.

LRP - Interfacing System Ruptures. 1.0/0.09

The particular check valve combination determines where the overpressurization occurs. If CF-30 and DH-76 fail, the LPI system will be overpressurized. If CF-30 and CF-28 fail, then the RCS water will backleak into the CFTs. LPI overpressurization will result in certain rupture, with the DHR heat exchanger being the most likely failure location (see Appendix H). However, overpressurizing the CFT to 2200 psig results in only about a 9% probability of failure, as described below.

The CFT has two likely failure modes: cylinder rupture and plastic collapse head buckling (see NUREG/CR-5603, Table 2-11), which at 600°F have median failure pressures of 3130 psi and 3330 psi, and uncertainty factors of 0.24 and 0.?7, respectively. Assuming the failure pressure is a lognormally distributed variable, taking the natural logarithm produces a normal distribution. The probability that the failure pressure is below 2200 psi (the RCS system pressure), can then be calculated from tabulated standard normal curve areas.

Cylinder rupture:

P(failure press. < 2200 psi) = phi((ln(2200)-ln(3130))/0.24)
= phi(-1.46)
= 0.0722

Head collapse:

P(failure press. < 2200 psi) = phi((ln(2200)-ln(3330)).0.27)
= phi(-1.52)
= 0.0643

However, given head collapse, there is only a 20% probability that a rupture will occur, therefore:

P(head collapse rupture) = (0.0643) x (0.2) = 0.013

The total failure probability of the CFT is then (assuming the failure modes are independent):

P(total) = P(cyl rupt) + P(head coll) - (P(cyl rupt) x P(head coll))
= 0.0722 + 0.013 - 0.001 = 0.084

LD2-LP - Operators Fail to Detect ISLOCA. 0.0035

The information available to operators regarding overpressurization is high-pressure alarm, and relief valve 1529 opening. The time for overpressurization detection is short (< 2 min). For the operators to fail to detect the ISLOCA, they must fail to detect the alarms associated with the overpressure, and those alarms associated with the rupture such as DH-8B high temperature alarms, and high-temperature computer and annunciator alarms associated with DH pump discharge. A 3-out-of-5 failure gate logic was used to model the operators inability to detect pertinent information. Failure probabilities were determined from THERP (see Appendix E, Figure 40, Table E35).

LDA2-LP - Operators Fail to Diagnose ISLOCA. 0.01

This scenario examines the operators ability to diagnose an ISLOCA after a rupture in the LPI system has occurred. The operator can fail in the correct diagnosis by failing to implement the RCS small leak procedure (BW-OP-2522), thereby not carrying out the appropriate series of actions. Operators can also fail to detect the event signature that would involve the detection of at least four of the following indicators: high-temperature computer alarm T369, high-temperature annunciator alarm DH8B, relief valve

PSV-1529 opens, high-temperature alarm T357 for DH pump discharge, or annunciator alarm for DH pump discharge. After the event signature has been detected, the operators then must reach the knowledge-based conclusion that the event signature is an ISLOCA. This aspect of the diagnosis is knowledge-based, with a failure probability of 0.10. The HEPs were determined using engineering judgement. The task analysis information indicated that the crew could diagnose this event provided that two or more indications were present. The modeling operator failure, therefore, assumed failure if three or more indicators were not properly addressed by the crew (e.g., ignored, misinterpreted), (see Appendix E, Figure 41)

LI2-LP - Operators Fail to Isolate ISLOCA. 0.148

For modeling purposes, isolation is considered to be those actions that the operators take to physically isolate the ISLOCA. The cognitive aspects of the operator determining where to isolate is considered in the modeling of diagnosis (LDA2-LP). Therefore, the operator will fail to isolate the ISLOCA when the operator fails to close either DH-1A or DH-1B The HEP was determined using engineering judgement (see Appendix E, Figure 42).

LMI-LP - Operators Fail to Mitigate ISLOCA. 1.0

Many things determine the potential for mitigating a possible radioactive release from an ISLOCA. These include: location of rupture, submergence of the break, presence and operation of fire suppression sprays, design of the auxiliary building (water tight doors, flow paths to the environment, etc), conditions at the time of core degradation (i.e., temperature of the pipes, water and surfaces inside the aux-bldg), and effects of severe accident phenomena such as possible hydrogen generation and burning.

The results of the aux-bldg environmental analysis (see discussion of Break Sequence 4 in Appendix M) indicate that for this ISLOCA sequence, the break will likely be submerged. However, the calculations performed for this study only consider the time up to failure of all ECCS, and do not examine conditions at the time of core degradation. Therefore, given the uncertainties associated with the parameters mentioned above, the initial assumption used the bounding situation that all releases are unmitigated.

Once risk was calculated this assumption was reexamined. Based on the relatively low ISLOCA risk results (i.e. 6 person-rem/Rx-year), it was deemed prudent to not expend additional effort on this issue.

LD2-CFT - Operators Fail to Detect ISLOCA. 1E-4

For this event, the location of the failure strongly determines the likelihood that it will be detected in a timely manner. The CFT is well instrumented, which is an aid to operators, but the time frame for detection is short (< 2 min). However, monitoring, as required by the plant's tech specs, is routine. Because there are procedures to address abnormal conditions in the CFTs, these operator actions were considered to be rule-based. For the operators to fail to detect the overpressurization, they must either fail to detect the CFT high-pressure alarm or fail to detect that the CF-7A relief valve is open.

Detection of ISLOCA after a rupture has occurred is much easier. In the CFT scenario, the following sources of information are available: high containment sump alarm, containment spray alarm, CFT level drop indication, and radiation alarms. The HEP was determined from THERP and used a three out of six failure logic (see Appendix E, Figure 36, Table E33).

LDA2-CFT - Operators Fail to Diagnose ISLOCA. 1E-4

This scenario examines the operator's ability to diagnose an ISLOCA after a rupture has occurred. The operator can fail in the correct diagnosis by failing to detect at least four of the following indicators: high containment sump level, spray alarms, radiation alarm inside containment, CFT level, SFAS trip, relief valve CF-7A/B open, or CFT high pressure. After the event signature has been detected, the operators then must reach the conclusion that the event is an ISLOCA. Values were taken from THERP and were used in an HRA fault tree with a four out of eight failure logic (see Appendix E, Figure 37, Table E34).

LI2-CFT - Operators Fail to Isolate ISLOCA. **0.149**

Isolation is considered to be those actions that the operators take to physically isolate an ISLOCA. The cognitive aspects of the operator determining where to isolate is considered in the modeling of diagnosis (LDA2-CFT). Therefore, the operator will fail to isolate the ISLOCA only when the operator fails to close CF-1A/B. The value for this HEP was determined from THERP (see Appendix E, Figure 38, no table).

Appendix E

Human Reliability Analysis for the Babcock and Wilcox ISLOCA Probabalistic Risk Assessment

CONTENTS

FIGURES

TABLES

B & W ISLOCA Human Reliability Analysis

This appendix describes in detail the methodology and results of the human reliability analysis (HRA) for the first ISLOCA probabilistic risk assessment (PRA). HRA was used to model the predominant human errors for each significant scenario in the PRA. HRA is a methodological tool for analyzing, predicting, and evaluating work-oriented human performance in quantitative, that is, probabilistic terms. As a diagnostic tool, HRA can be used to identify those factors in the system which lead to less than optimal human performance and can estimate the error rate anticipated for individual tasks. In a given system, or sub-system, HRA can also be utilized to determine where human errors are likely to be most frequent. Traditionally, HRA analysts model human performance through the use of event trees like those found later in this appendix.

The general methodological framework for this ISLOCA HRA was based on guidelines (under development) from the NRC-sponsored Task Analysis-Linked Evaluation Technique (TALENT) Program [E-1] which recommends the use of task analyses, time line analyses, and interface analyses in a detailed HRA. NUREG/CR-1278, the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (THERP) [E-2], recommends similar techniques and, in addition, provides a data base that can be used for estimating human error probabilities (HEPs). Finally, this ISLOCA HRA integrated the steps from the Systematic Human Action Reliability Procedure (SHARP) [E-3], and A Guide for General Principles of Human Action Reliability Analysis for Nuclear Power Generation Stations (draft IEEE standard P1082/D7 [E-4]).

From this combination of approaches, the analysts identified 11 basic steps, summarized below, which were used as guidelines for this HRA. Following this brief summation of the 11 steps is a detailed explanation of how each step was applied to the HRA process. The 11 basic steps are as follows:

1. Select the team and train them on relevant plant functions and systems. (IEEE P1082)

2. Familiarize the team with the plant through the use of system walkdowns, simulator observations, etc. (IEEE P1082)
3. Ensure that the full range of potential human actions and interactions is considered in the analysis. (SHARP) (IEEE P1082)
4. Construct the initial model of the relevant systems and interactions. (IEEE P1082)
5. Identify and screen specific human actions that are significant contributors to the safe operation of the plant. This was accomplished through detailed task analyses, time line analyses, observations of operator performance in the plant and in the simulator, and evaluations of the human-machine interface. (SHARP and IEEE P1082)
6. Develop a *detailed* description of the important human interactions and associated key factors necessary to complete the plant model. This description should include the key failure modes, an identification of errors of omission/commission, and a review of relevant performance shaping factors. (SHARP) (IEEE P1082)
7. Select and apply appropriate HRA techniques for modeling the important human actions. (SHARP)
8. Evaluate the impact on ISLOCA of significant human actions identified in Step 6. (SHARP)
9. Estimate error probabilities for the various human actions and interactions, determine sensitivities, and establish uncertainty ranges. (SHARP) (IEEE P1082)
10. Review results (for completeness and relevance). (IEEE P1082)
11. Document all information necessary to provide an audit trail and to make information understandable. (SHARP)

The following paragraphs explain in detail how each of the preceding steps was completed. Since the PRA/HRA process is iterative in nature, the reader should note that several sections of this 11 step method were repeated to refine the analysis.

The first two steps in this process required the selection of a PRA/HRA team and their subsequent training on the plant and its relevant systems. The PRA/HRA team from the INEL was composed of three members: a nuclear engineer (for the PRA), a human factors engineer (for the HRA), and an electrical engineer (with extensive experience in both the PRA and HRA approaches). To familiarize, or train themselves, the team members reviewed the following:

- mechanical and electrical system descriptions (e.g., the reactor coolant, residual heat removal, safety injection, and chemical and volume control systems),
- a sourcebook of plant systems and schematic drawings,
- the plant's Final Safety Analysis Report (FSAR),
- the plant's Technical Specifications [E-5],
- plant procedures (operating, abnormal, emergency, maintenance, administrative, etc.), station directives, and operational practices,
- piping and instrumentation diagrams (P&IDs),
- the types, capacities, and locations of check valves/motor-operated valves identified as being pressure isolation valves,
- training materials such as flow charts, lesson plans, etc.,
- crew composition (for control room and auxiliary building operators) and level of training/experience,
- significant precursor information from general ISLOCA-related LERs,

This training/familiarization process for the plant's systems was enhanced by a two-week visit to the plant and by a second one week data gathering trip.

Step #3 required that significant human actions and interactions be incorporated into the ISLOCA PRA analysis. This was accomplished through an extensive data collection process during the plant visit. As part of the data collection, the utility provided written procedures, training materials, and P&ID drawings. This data was supplemented by interviews and detailed task analyses with both licensed and non-licensed nuclear operators in the plant. Observations of control room personnel, the use of the utility's simulator, and system walkdowns with licensed and non-licensed operators supplied additional information.

The initial plant models were constructed in the fourth step. Using the plant-specific data gathered in Step #3, the HRA analysts worked with the PRA analyst and systems engineering personnel to specify human actions related to the postulated ISLOCA scenarios. Significant attention was given to latent, or precursor, human errors during normal operations which could lead to inoperable equipment or misaligned valves. Examples of these precursor actions included: jumpering of valves to defeat protective interlocks, maintenance procedures, in-service testing practices, and administrative procedures governing the generation and completion of work packages.

The HRA analysts also examined active, or initiator, failures which could lead to an ISLOCA, and post-initiating human errors during responses to abnormal situations. Examples of initiator failures included violations of Technical Specifications, procedural violations (such as early entry into decay heat removal), selection of the incorrect vent path, and reconfiguring plant equipment. For post-initiating errors, the HRA team examined operator responses following a significant break outside containment. Specifically, the HRA analysts looked at operator actions entailing detection, diagnosis, recovery, and isolation.

The fifth step required the HRA analysts to identify those human actions which are significant contributors to the effective operation and safety of the plant. Using the data collected in Step #3, in conjunction with a review of operational procedures and training materials, the HRA team screened the various human actions, identifying those which had a significant impact on plant operations and/or safety with respect to ISLOCA. These significant human actions were included in the PRA event trees, and they helped guide the activities in the next step.

The output from the preceding step (i.e., Step #5) was a group of important human actions, for specific ISLOCA scenarios, which were described in generic, functional terms (e.g., operators recover system). In the sixth step, the analysts expanded the description of each of these key human actions from a functional description into specific operator tasks and subtasks (e.g., operator opens valve DH-23, or operator closes valve DH-11). By breaking down the human actions into specific tasks and subtasks associated with individual equipment and procedures, the analysts began to identify specific failure modes, root causes, and failure effects. The description of each task/subtask was enhanced by referencing significant performance shaping factors (PSFs) which affected a given task. These PSFs were derived from the task analyses, time line analyses, evaluation of the human-machine interface, and direct observations of operator performance. Examples of PSFs included:

1 - the quality of the human-machine interface,
2 - written procedures (emergency, abnormal, maintenance, etc.),
3 - P&IDs,
4 - response times for systems and personnel,
5 - communication requirements,
6 - whether the operator actions were skill, rule, or knowledge-based,
7 - crew experience,
8 - levels of operator stress in different scenarios,
9 - feedback from the systems in the plant,
10 - task dependence and operator dependence,
11 - location of the task (e.g., control room, auxiliary building, etc.),
12 - training for individual operator actions, including ISLOCA situations.

Each PSF was seen as casting either a positive or negative influence on the basic HEP, that is, as either decreasing or increasing the probability of failure for a given human action. For example, some of the positive PSFs found at the plant included the following:

* Workload alone was insufficient to introduce either initiating events or precursors for ISLOCA

* Newly introduced operating schematics could prove to be viable operator aids

* Operators' practice of repeating verbal instruction increases the probability for effective oral communication, and

* The presence of consistent labeling in the control room contributes to positive operator performance.

Negative PSF findings include the following:

* The lack of operator awareness regarding ISLOCA;

* Lack of spacific training on ISLOCA;

* Lack of proper notes, cautions, and warnings in procedures related to ISLOCA;

* A lack of awareness that the computer high-pressure alarm on the HPI line could be caused by either leaky check valves or by the makeup and purification system operation;

* Lack of a valve status board in the control room, and absence of procedures for acknowledging computerized alarms.

* No main control board alarm or pressure indication was observed for the DHR system.

* Tagging was mixed, it seemed quite good in some areas and not as consistent in others.

For this HRA analysis, the majority of influences from specific PSFs were implicitly modeled as each HEP was identified and quantified using various THERP tables and engineering judgement. A careful examination of these tables will show how individual basic HEPs can only be identified after associated PSFs are specified. Stress and dependence were explicitly modeled (using THERP) as two of the more significant PSFs. From a human performance perspective, high levels of stress lead to higher probabilities of human error. Generally, a person's short-term memory (STM) can retain from five to nine items of information for brief periods. However, as stress increases, this capacity shrinks to levels where STM can only hold three to five items. This well documented finding interacts with a phenomenon called cognitive tunnel vision where high levels of stress cause an operator's visual and perceptual abilities to begin shrinking into a limited focus so that only one or two salient aspects of his environment are featured. Also, as stress continues to increase, the operator begins to retreat from current conditions, relying on previously learned (perhaps incorrect) patterns of behavior. For purposes of this HRA analysis, stress level was considered optimal with three exceptions: (a) When personnel were sent into containment, (b) when personnel were attempting to isolate the ISLOCA, or (c) when site evacuation was said to occur. THERP procedures allow for modifying HEP values as a function of stress level and where such modifications are made they are noted.

In several of the ISLOCA scenarios, low (LD), moderate (MD), and high (HD) levels of dependence were assigned between the control room supervisor (CRS) or shift supervisor (SS) and the licensed reactor operator (RO). As used in THERP, dependence refers to the level of interaction between two or more workers. Dependence is usually modeled on a scale which ranges from complete dependence (where a second worker fails on a given task because of the failure of a primary worker on the same task) to complete independence (zero dependence or ZD).

A detailed data collection form (see Figures #1 and #2 in this appendix) was developed as an aid in the HRA data collection, task analyses, and the decomposition and description activity just mentioned. This data form served as a template which guided the collection of the requisite information, in sufficient detail, for each task or subtask in the dominant ISLOCA sequences. Additional items of information, for each human action, were added to these forms as new details surfaced (i.e., details from follow-up telephone conversations with plant personnel, the ISLOCA inspection report for this plant, and a comparison of procedural steps to P&IDs).

The output from the preceding step (#6) is an extensive list of operator tasks and subtasks (with their associated PSFs) for each human action in the dominant PRA sequences. These detailed tasks are the required input for the seventh step, where appropriate HRA techniques for modeling the significant human actions were selected and applied. For each human action, the analysts selected an appropriate technique for task modeling and quantification. Because most of the human actions in this HRA involved the use of various p 1

Sequence ID ______________ Task ID ____________ Subtask ID ________

Crew size & composition ________________________________

Who does task/subtask? ________________________________

Crew experience: Low_____ Optimal_____ Moderate_____ High_____

Is time limit important for this task/subtask? Yes or No

Time to perform task/subtask (after diagnosis/decision) ____________

Median response time for whole task_______ Std Dev _______

Plant/system time available ________________________

If task not successfully completed, what is next action?__________

____

____

____

and type of alarms competing for attention ________________

____

Quality of plant interface: Excellent___ Good___ Fair___ Poor___ Very Poor

Operators' Stress: Low___ Optimal___ Moderate___ High____

Type of instrument/control ________________________

HF notes on controls ________________________

____

Consequence of improper performance High___ Medium___ Low___

Explain: ________________________________

Feedback/system response to operator action ________________

____

Operation routine: Yes or No Operation/transient understood: Yes or No

Proc Reqd: Yes or No Proc covers case: Yes or No

Proc well written: Yes or No Proc understood: Yes or No

Proc practiced: Yes or No How much practice/training on task? ________

Cognitive Behavior: Skill_____ Rule_____ Knowledge_____

Tagging: Yes or No Describe: ________________________

Recovery Actions: Checklists_____ Inspections_____ 2nd Person_____

Feedback from Annunciators_____ Alarms_____ Displays_____

____

Figure 1: ISLOCA Data Collection Form, page 1

p. 2

Local or Remote operation? Explain: ______________________

Type of clothing during action: ______________________

____

Tasks or subtasks done step-by-step_____ or Dynamic_____

Dependence: Is the order of the tasks critical Yes or No

Does the success/failure of one action affect the success/failure of the next

Yes or No Explain ______________________

If 2 men do the job, does the action of either one affect the success/failure of the next? Yes or No Explain: ______________________

Is the job done with rest stops_____ or continuous performance_____?

Is there any radiation safety or caution for this job? Yes or No

If yes, what dosage?__________ mrem

HF comments of plant-specific PSF's: ______________________

____

____

____

Additional Comments/observations: ______________________

____

Figure 2: ISLOCA Data Form, page 2

written procedures, THERP-type HRA event trees were used in modeling a majority of the human actions in the detailed analysis. However, not all ISLOCA scenarios were best represented by THERP event trees alone. In those cases, HRA fault trees were used in conjunction with the typical THERP event trees. The fault trees and THERP event trees were used in a detailed analysis to estimate the probability of human error for each of the dominant human actions. Quantification techniques included THERP, NUCLARR [E-8], and engineering judgement. For each human failure, basic HEPs were calculated using one of these techniques and were then modified using performance shaping factors (PSFs) to realistically describe the work processes at the utility.

Prior to the quantification, or estimation of human error probabilities, the PRA and HRA specialists reviewed and evaluated the significant human actions, and their associated PSFs, for each of the dominant ISLOCA sequences (Step #8). After this evaluation, the HRA analysts developed the HRA event trees and fault trees used to model the significant human actions, and their associated PSFs, for each of the dominant ISLOCA sequences (Step #8). According to the SHARP method, the development and use of these HRA fault and event trees "provides a disciplined approach for explicitly evaluating alternative actions and, if properly interpreted, may provide the rationale for including some human errors known as acts of commission in the event trees." This HRA modeled errors of commission and omission, which are identified on specific branches of the event trees seen later in this appendix.

Assigning HEP estimates to each of the subtasks was the major activity in Step #9 - Quantification. Traditionally, HRA analysts model human performance through the use of an event tree like Figure 3, which represents "HDA2MU", operators fail to diagnose ISLOCA. For example, on the top left, Event "a" - operators select RCS small leak procedure BW-OP-2522 - is the success path. Failure to accomplish this task is modeled as Event "A" - Operators fails to select small leak procedure. When a second operator is involved, such as in Event "B" - second operator fails to select small leak procedure, the action of this second operator may be modeled in a recovery branch, as shown in Figure 3. Since the second operator is in the control room in this scenario, he/she also has an opportunity to select BW-OP-2522, the small leak procedure. If successful, this becomes a recovery action because it would bring the model back to the success path (via the dotted lines in Figure 3).

Individual error branches on each of the HRA event trees were quantified using techniques from THERP, NUCLARR, see [ ]) and engineering judgement. Specific human actions were assigned an estimate of a basic, or unmodified, HEP. These basic HEP estimates were then revised using performance shaping factors (PSFs) to realistically describe the work process at the plant. Each PSF was either positive or negative and, accordingly, either decreased or increased the likelihood of a given human error. For example, an analog meter, like a pressure gauge, which does not have easily seen limit marks, may be judged to have a negative PSF and there would be a higher probability for human error in reading the gauge. Individual PSFs were derived from the task analyses, time line analyses, evaluation of the human-machine interface, and direct observations of operator performance. They are presented as part of the ISLOCA Inspection Report [12].

Finally, all possible failure paths (i.e., sequences that included either single or multiple human errors leading to a failure of the action modeled by the HRA tree) were identified and used to estimate the total failure probability for the action modeled in the HRA tree, in accordance with the THERP guidelines. As depicted by Figure 3, each human error event tree may have several unique error paths. For example, event "A" and event "B" constitute an error path in which the first RO (reactor operator) fails to select BW-OP-2522, the small leak procedure (event "A"). This error action is followed by the failure of a second RO to select the same procedure (event "B"). In a similar manner, failure path "A-b-C-D" models a sequence where the RO fails to select the small leak procedure, the second RO recovers from this error by correctly selecting BW-OP-2522 (event "b"), only to have both RO's fail at actions "C" and "D", the steps which would determine if there was a leak by comparing the rate of makeup to the rate of letdown. Probabilities

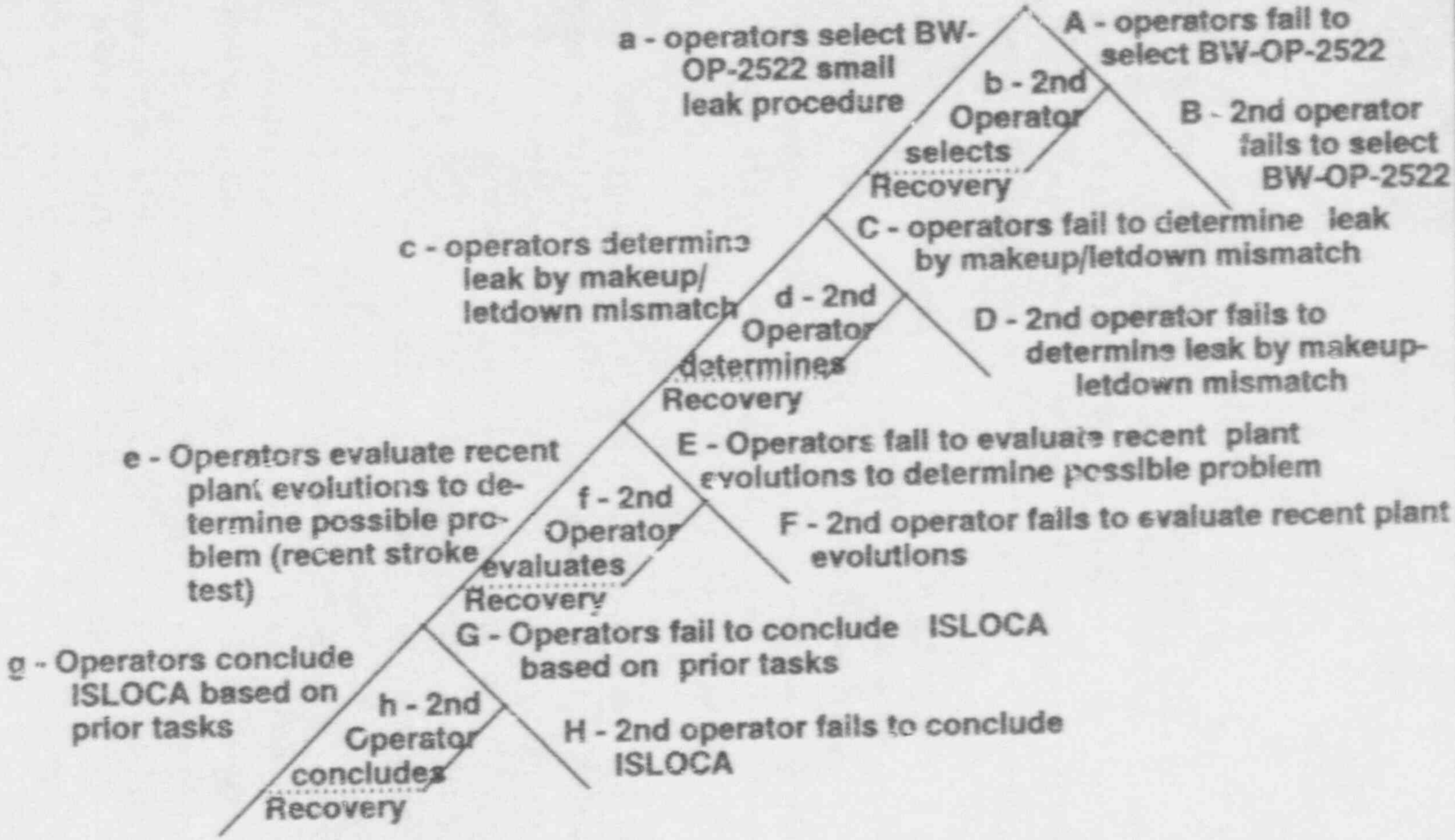


Figure 3: HRA Event Tree for HDA2MU, Operators Fail to Diagnose ISLOCA....

for each unique error path were calculated by multiplying each HEP on a given error path by other HEPs on the same path. For example, the error rate for path "A-B" would be calculated by multiplying the HEP of failure "A" (0.013) by that for failure "B" (0.161), resulting in a nominal HEP (0.002) for that specific path. Other error paths for this event tree include: "A-b-c-E-F", "a-c-E-F", and "a-C-D", etc. The individual error path failure probabilities were then summed to give the total event tree failure probability.

Individual error paths were identified and failure probabilities were estimated using the HEPs and tables from THERP. (The probabilistic values in the THERP tables are to be considered as median values from a lognormal distribution). In those non-procedural tasks where THERP was unable to generate a realistic model, two other techniques were used to generate HEP's. The first method used NUCLARR[7] which is an automated data base management system used to process, store, and retrieve human and equipment reliability data. NUCLARR was developed by the United States Nuclear Regulatory Commission to provide the risk analysis community with a repository of human error and hardware failure rate data that can be used to support a variety of analytical techniques for assessing risk. The human error component of NUCLARR complies with the specifications and procedures as described in NUREG/CR-4010, Specifications of a Human Reliability Data Bank for Conducting HRA Segments of PRA's for Nuclear Power Plants. The second technique relied upon engineering judgement to generate estimates of HEPs.

Basic median HEPs were converted to basic mean HEPs which have the same influence from relevant PSFs. Table E1 lists the basic median HEPs and nominal mean HEPs for the event tree depicted in Figure 3 (HDA2-MU). This table enumerates the basic human actions/errors, the basic or unmodified HEPs (median and mean), their sources from the table and item number in THERP, whether the action was modeled as being performed in a step-by-step mode or dynamically, PSF modifier values and the related THERP source, level of dependency, and finally, the nominal, or modified, mean HEP with its error factor (derived from THERP HEPs or THERP Table 20-20). NOTE: the 6-digit accuracy for numerical values in the following tables is an artifact of the software used for quantification and does not imply 6-digit precision.

Table E1: HEPS for HDA2-MU ROs Fail to Diagnose ISLOCA

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	ROs fail to select BW-OP-2522; small leak procedure	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
B	2nd RO fails to select small leak procedure	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
C	ROs fail to determine leak by makeup/letdown mismatch	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
D	2nd RO fails to determine leak from makeup/letdown mismatch	0.1	5.0	T20-22 3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
E	ROs fail to evaluate recent plant evolutions to determine problem	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	[illegible]317	10.0
F	2nd RO fails to evaluate recent plant evolutions to determine problem	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
G	ROs fail to conclude ISLOCA based on prior tasks	0.0001	30.0	T20-3 #5	SBS	1	T20-16 [illegible]	ZD	0.000847	0.000847	30.0
H	2nd RO fails to conclude ISLOCA based on prior tasks	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0

Table E2: Failure Paths and Total Failure Probabilities

for HDA2-MU

	Failure Path	Calculations	Results
1	AB	0.013317 x 0.161383	0.002149
2	AbCD	0.013317 x 0.013317 x 0.161383	0.000028
3	AbCdEF	0.013317 x 0.013317 x 0.013317 x 0.161383	*
4	AbCdEfGH	0.013317 x 0.013317 x 0.013317 x 0.000847 x 0.161383	*
5	AbCdeGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
6	AbcEF	0.013317 x 0.013317 x 0.161383	0.000028
7	AbcEfGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
8	AbceGH	0.013317 x 0.000847 x 0.161383	0.000001
9	aCD	0.013317 x 0.161383	0.002149
10	aCdEF	0.013317 x 0.013317 x 0.161383	0.000028
11	aCdEfGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
12	aCdeGH	0.013317 x 0.000847 x 0.161383	0.000001
13	acEF	0.013317 x 0.161383	0.002149
14	acEfGH	0.013317 x 0.000847 x 0.161383	0.000001
15	aceGH	0.000847 x 0.161383	0.000136
		Total Failure Probability	**0.006**
		Error Factor (IRRAS 4.0)	**14.93**

Table E2 lists the individual failure paths for Figure 3, HDA2-MU, and the resulting failure probabilities for each path, including how the failure probabilities were calculated (again 6-digit numbers do not imply 6-digit precision for HEP estimates). (As a note for subsequent tables, failure probabilities of "*" on the tables signify negligible error rates which were less than 10^{-6}.) Table E2 also lists a total failure probability for each event tree, which is simply the sum of the failure probabilities from the individual failure paths. As indicated in Table E2, the total failure probability for the HDA2-MU event tree in Figure 3 is estimated to be about 0.006. As a point estimate, given the PSFs discussed earlier, an RO, or group in the CR, can be expected not to enter the correct procedure after detecting a loss of coolant, about six out of a thousand.

As discussed in Section 4.2 of the main report, the estimates of human error probabilities obtained from THERP are generally treated as point estimates with a given error factor. The authors of THERP indicate that there is insufficient data, at this time, to accurately determine the true shape of the underlying probability distribution associated with these point estimates and that these distributions are unimportant. Quoting from THERP (pages 7-6 through 7-8):

> "Although we would like to have data clearly showing the distributions of human performance for various NPP (nuclear power plant) tasks, there is ample evidence that the outcomes of HRAs are relatively insensitive to assumptions about s h distributions...."

The authors then provide several examples to support a general conclusion:

> "the assumption of normal, lognormal, or other similar distributions will make no material difference in the results of HRA analyses for NPP operations. In some cases, this insensitivity may result from a well designed system that has so many recovery factors that the effect of any one human error on the system is not substantial.... For computational convenience, one might wish to assume the same distribution for

probabilities of human failure as the one used for probabilities of equipment failure, as was used in WASH-1400."

To summarize, the authors of THERP "suggest" that HRA analysts "assume" the point estimates from THERP are medians from a lognormal distribution, even though such an assumption is "speculative" at best.

While the THERP approach (treating the HEPs as median values from a lognormal distribution) has certain computational and interpretational advantages, it has one distinct drawback, with respect to PRAs. In most PRAs, hardware failure probabilities are assumed to be lognormally distributed. The HEPs are multiplied by hardware failure probabilities when calculating core damage frequencies. This requires a median to be multiplied by a mean, a procedure which does not result in a mean value of the core damage frequency. A mean core damage frequency can be obtained by converting the median HEP values (from an assumed lognormal distribution) to mean HEP values, thereby allowing the necessary multiplications.

This HRA adopted THERP's recommendation to treat each HEP as a median value from a lognormal distribution. Detailed HRA analyses were conducted for each of the significant scenarios identified in this ISLOCA PRA. Tables E1 and E2 summarize the results of these analyses, i.e., by converting the median HEPs to mean HEPs using the following formulas:

$$\textit{Mean HEP} = \exp\left(\mu + \frac{\sigma^2}{2}\right);$$

$$\textit{where } \tilde{x} = \textit{the Median HEP};$$

$$\mu = \ln\tilde{x};\ \textit{and},$$

$$\sigma = \frac{\ln(\textit{ErrorFactor})}{1.645}$$

Converting median HEPs (from an assumed lognormal distribution) to mean HEPs allowed uncertainties in human error to be included in calculations of the uncertainty in core damage frequency. The actual conversions to mean HEPs were accomplished by inserting the basic, median HEPs in each event tree into the equations above. The resulting mean HEPs were then modified by appropriate PSFs and used in the appropriate error branch on specific event trees to calculate error path and total failure probabilities for each event tree .

A careful review of Table E1 will show that the conversion from median to mean HEPs can cause problems with the resulting confidence interval. The reader may recall that individual HEPs are considered a point estimate with some uncertainty, e.g., a confidence interval, surrounding it. Generally, this confidence interval is defined by calculating the upper bound (95th percentile) and lower bound (5th percentile) for each HEP. The upper bound is found by multiplying the nominal (modified-median) HEP by its associated error factor (EF) and the lower bound results by dividing the nominal (modified-median) HEP by the same EF. For example, if the basic median HEP for event "A" (Table C1) were modified for higher stress (multiplied be a factor of 2), it would become a value of 0.01 (the nominal mean HEP equals 0.03), the resulting upper bound is 0.3 (0.03 x an EF of 10). Likewise, the lower bound is 0.003 (0.03 divided by the EF of 10).

However, when a basic HEP is modified by several PSFs, including dependency, problems with the confidence interval begin to arise. For example, imagine an event with a basic median HEP of 0.0001 and an EF of 10. When this HEP is converted to a mean value and modified for stress and high dependence, the resulting nominal mean HEP is 0.5 with an EF of 5 (from THERP Table 20-20, #5). If one calculates the upper bound for this HEP by multiplying this value by the EF (or more correctly by multiplying the modified median value, 0.5, by the EF), the result is a value of 2.5; this value is an anomaly, because the maximum value for a probability is constrained to be less than or equal to one (i.e., unity). To correct this difficulty, the nominal mean HEP and EF were adjusted using a constrained lognormal distribution (see Kelly, Auflick, and Haney, 1992 for a detailed discussion). The revised mean HEP would be 0.279 with an EF of 2.5. (NOTE: when this situation occurs in the following tables, the resulting revised nominal mean HEP and EF are shown in the table as the values with a "#", just below the old values for the related event.

Tables E3 through E7 summarize the HEP revisions for each sequence and individual actions in this HRA. These tables list the identifier for each human action, a brief description of the human action, the mean HEP, and error factor-EF, calculated from an uncertainty analysis using IRRAS 4.0, or engineering judgement.

Table E3: HPI Scenario Involving Quarterly Stroke Test for 2A, MU&P Flow

Identifier	Human Action	Mean Hep (EF)
HV1-MU	HP vent line open	0.0013 (2.94)
HM1-MU	HP MOV2A opened for test	1.0
HM2-MU	Operators fail to close HP MOV2A	0.008 (2.27)
HV2-MU	HP vent line open(per procedure)	1.0
HD2-MU	Operators fail to detect ISLOCA	0.0028 (7.40)
HDA2-MU	Operators fail to diagnose ISLOCA	0.006 (14.93)
HI2-MU	Operators fail to isolate ISLOCA	0.002 (3)

Table E4: HPI Scenario Involving Quarterly Stroke Test, No MU&P Flow

Identifier	Human Action	Mean Hep (EF)
HM1-HP	HP MOV2B opened for test	1.0
HV1-HP	HP vent line open	0.0013 (2.94)
HD2-HP	Operators fail to detect ISLOCA	0.0014 (9.50)
HDA2-HP	Operators fail to diagnose ISLOCA	0.006 (14.93)
HI2-HP	Operators fail to isolate ISLOCA	0.002 (3)

Table E5: Shut-down Scenario Involving Promature Opening of DH11 & DH12

Identifier	Human Action	Mean Hep (EF)
DM1-SD	Operators open DH11 & 12 too soon	0.00066 (10.01)
DD2-SD	Operators fail to detect ISLOCA	0.0002 (10.79)
DDA1-SD	Operators fails to diagnose ISLOCA	0.006 (14.93)
DI2-SD	Operators fail to isolate ISLOCA	0.008 (5)

Table E6: Start-up Scenario Involving DHR System

Identifier	Human Action	Mean Hep (EF)
DM1-SU	DH11 & 12 left open	0.0002 (3.53)
DD1-SU-A,C	Operator fails to detect overpressure given that relief valve opens	0.0001 (16.40)
DI1-SU-C	Operators fails to isolate RCS	0.0092 (3.0)
DM2-SU	DH 21 & 23 left open	0.0002 (4.85)
DI1-SU-A	Operators fail to isolate RCS	0.013 (2.37)
DD1-SU-B,D	Operator fails to detect overpressure, given relief valve closed	0.001 (3.0)
DI1-SU-D	Operators fail to isolate RCS	0.0092 (3.0)
DI1-SU-B	Operator fails to isolate RCS from DHR	0.013 (2.37)
DD2-SU,A-D	Operator fails to detect abnormality(rupture)	0.0001 (22.99)
DA1-SU-A	Operator fails to diagnose ISLOCA	0.52 (1.6)
DA1-SU-B	Operator fails to diagnose ISLOCA	0.59 (1.5)
DA1-SU-C	Operator fails to diagnose ISLOCA	0.29 (2.5)
DA1-SU-D	Operator fails to diagnose ISLOCA	0.43 (1.9)
DI2-SU-A	Operator fails to isolate ISLOCA	0.113 (4.26)
DI2-SU-B	Operator fails to isolate ISLOCA	0.113 (4.26)
DI2-SU-C	Operator fails to isolate ISLOCA	0.016 (2.99)
DI2-SU-D	Operator fails to isolate ISLOCA	0.016 (2.99)

Table E7: Low Pressure Injection System ISLOCA Scenario

Identifier	Human Action	Mean Hep (EF)
LD2-CFT	Operators fail to detect ISLOCA	0.0001 (2.05)
LDA2-CFT	Operators fail to diagnose ISLOCA	0.0001 (43.37)
LI2-CFT	Operators fail to isolate ISLOCA	0.149 (5)
LD2-LP	Operators fail to detect ISLOCA	0.0035 (11.15)
LDA2-LP	Operators fail to diagnose ISLOCA	0.01 (10)
LI2-LP	Operators fail to isolate ISLOCA	0.148 (5)

In the final two steps (#9 and #10) of the HRA process, the analysts reviewed the results of the HRA and documented all of the information needed to provide an audit trail. As final HRA failure probabilities were generated for each ISLOCA sequence, the HRA analysts consulted with the PRA analyst and a systems engineer regarding the validity, completeness, and relevance of the results. During these reviews, several questions arose which required more information. Several telephone calls were placed to operations personnel at the plant and detailed interviews or walkthroughs were conducted with a past shift supervisor from the plant.

The last step necessitated the documentation of the data, methodology, and results from this HRA to provide an audit trail. This was accomplished by creating a data notebook containing the completed data forms, pertinent procedures, working notes from the ISLOCA inspection, and the NRC ISLOCA inspection report.

Modeling Of Human Actions And Estimated Human Error Probabilities

This section describes t ᵒA event trees and the HEP estimates for the human actions identified as sign. ant for the B&W ISLOCA HRA. The following tables present HRA event trees or HRA fault trees, subtask HEP tables documenting HEP estimation for each subtask branch on the trees. For those instances where a THERP type (HRA) event tree was used for modeling, additional tables are provided which show failure path calculations and total failure probability estimates for each human action. (These calculations are already an integral part of the HRA fault trees).

For purposes of the HRA analysis, stress level was considered optimal with three exceptions: (a) When personnel were sent into containment, (b) when personnel were attempting to isolate the ISLOCA, or (c) when site evacuation was said to occur. THERP procedures allow for modifying HEP values as a function of stress level and where such modifications are made they are noted. For purposes of this study, a lower bound of 1.0E-5 for human actions was assumed. This lower bound on the failure rate estimate includes the possibility of recovery actions by other members of the crew. In addition, most instances for which there is such a low number include situations where the time frame to respond and to recover from the abnormal event is relatively long, that is, it is easily measured in hours as opposed to minutes.

A description of important human actions modeled for the HRA analysis along with their corresponding failure rates is contained in Appendix D of this report. These actions include pre-initiating events, event initiation or detection, diagnosis, isolation, and mitigation for the HPI, DHR, and LPI sequences. It is important to review the event sequence descriptions, in order to understand aspects of the work environment and task demands as they influence safety and performance. The strengths and weaknesses of existing procedures, training, and instrumentation help to determine future strategy for ensuring an adequate response to the threat of ISLOCA. It should be noted that some events are illustrated only by fault trees and for some events, the event tree and associated HEP data table and failure path, represent only one box of the fault tree. Please consult the Table of Contents in front of this report for page numbers of specific figures and tables.

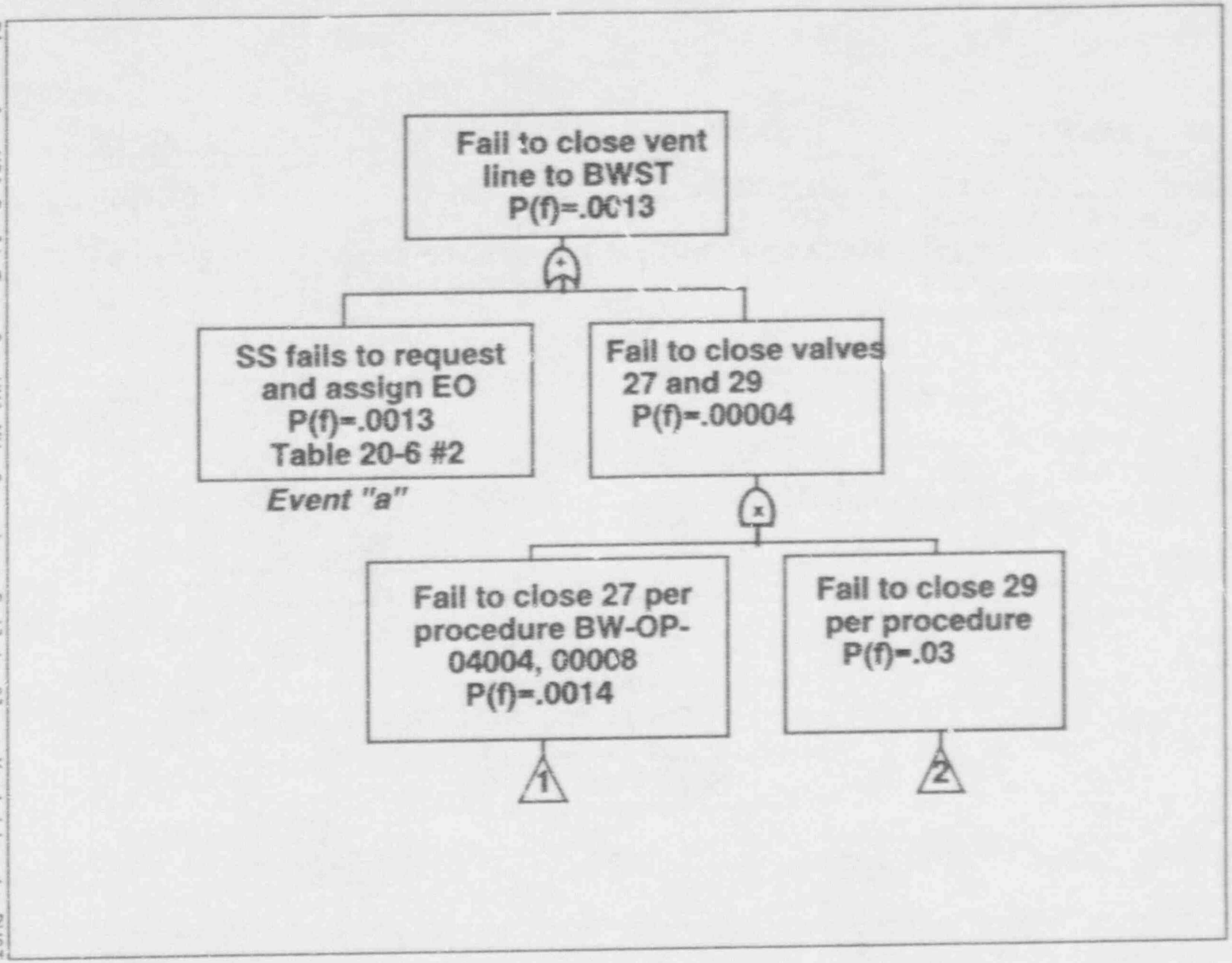


Figure 4: HRA Fault Tree for HV1-MU, Operators Fail to Close Vent Line to BWST

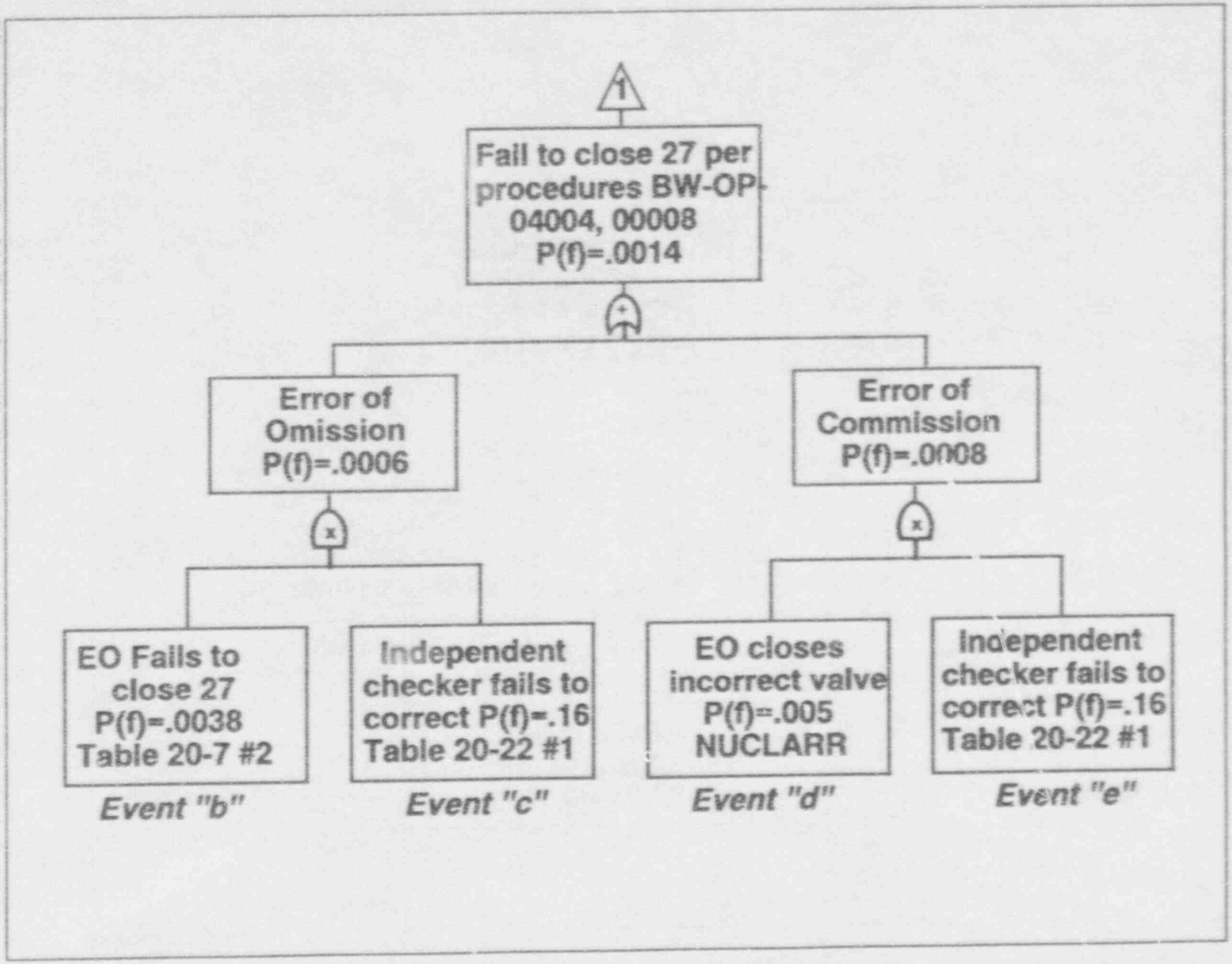


Figure 5: HRA Fault Tree for HVI-MU, cont.

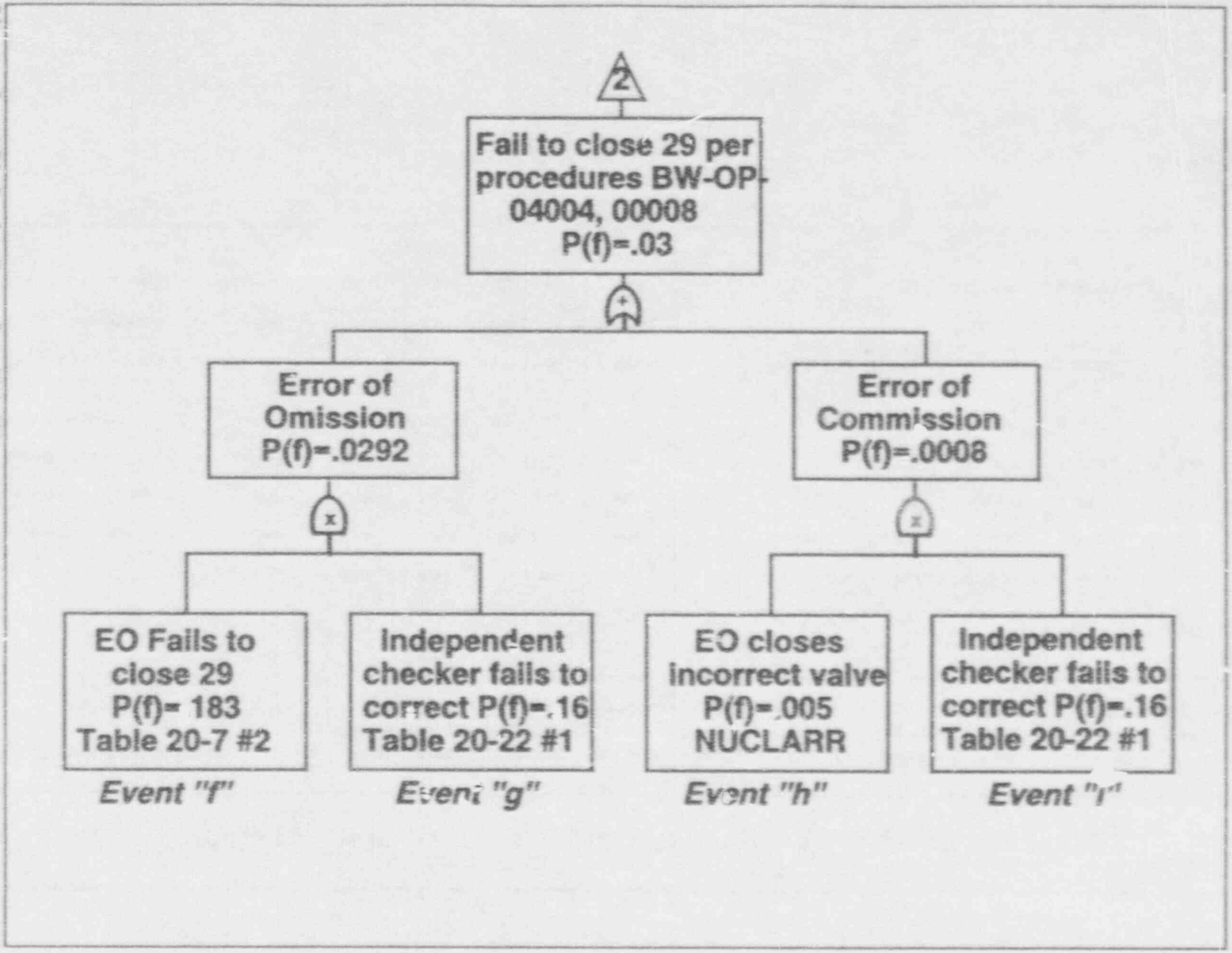


Figure 6: HRA Fault Tree for HV1-MU, cont.

Table E8: HEPS for HV1-MU Fail to Close Vent Line to BWST

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	SS fails to request and assign EO	0.001	3.0	T20-6 #2	SBS	1	T20-16 #2a	ZD	0.001249	0.001249	3.0
B	EO fails to close 27	0.003	3.0	T20-7 #6	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
C	Independent checker fails to correct error	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
D	EO closes incorrect valve	0.005	1.0	NUCLARR	SBS	1	———	ZD	0.005	0.005	1.0
E	Independent checker fails to correct error	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
F	EO fails to close 29	0.003	3.0	T20-7 #2	SBS	1	T20-17 (10-11)	MD	0.003749	0.146071	3.6
G	Independent checker fails to correct error	0.1	5.0	T20-22 #5	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
H	EO closes incorrect valve	0.005	1.0	NUCLARR	SBS	1	———	ZD	0.005	0.005	1.0
I	Independent checker fails to correct error	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0

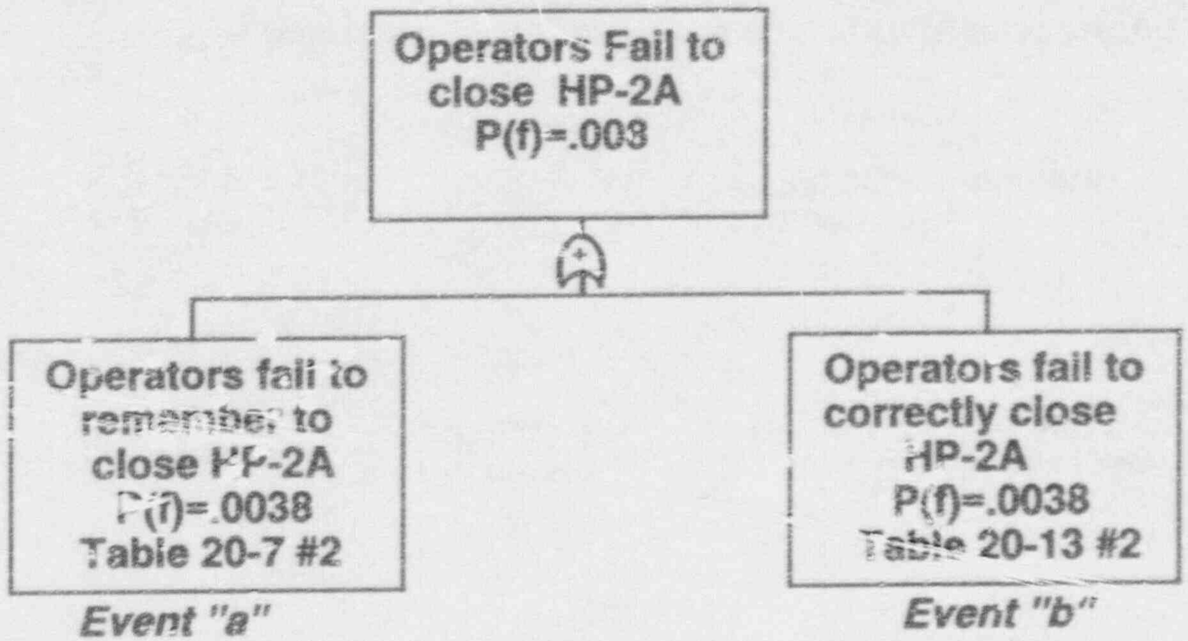
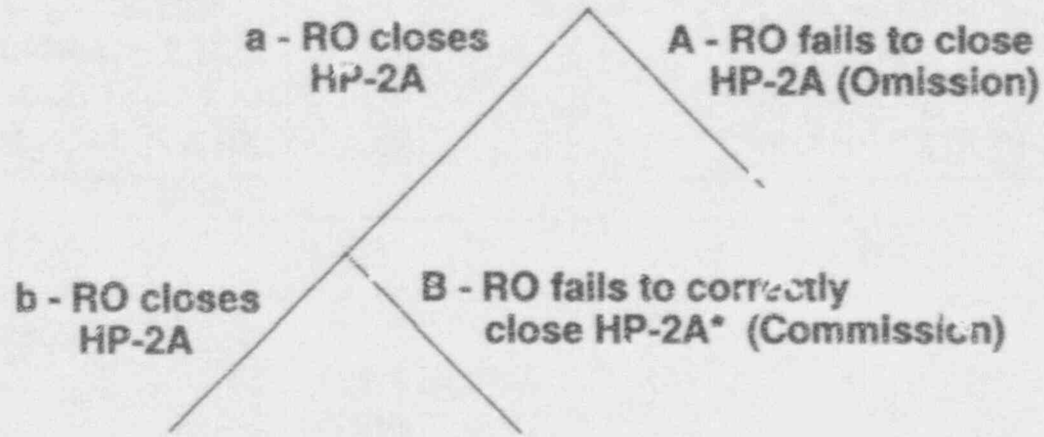


Figure 7: HRA Fault Tree for HM2-MU, ROs Fail to Close HP-2A MOV



* Includes failure on selection of the appropriate pushbutton

Figure 8: HRA Event Tree for HM2-MU, ROs Fail to Close HP-2A MOV

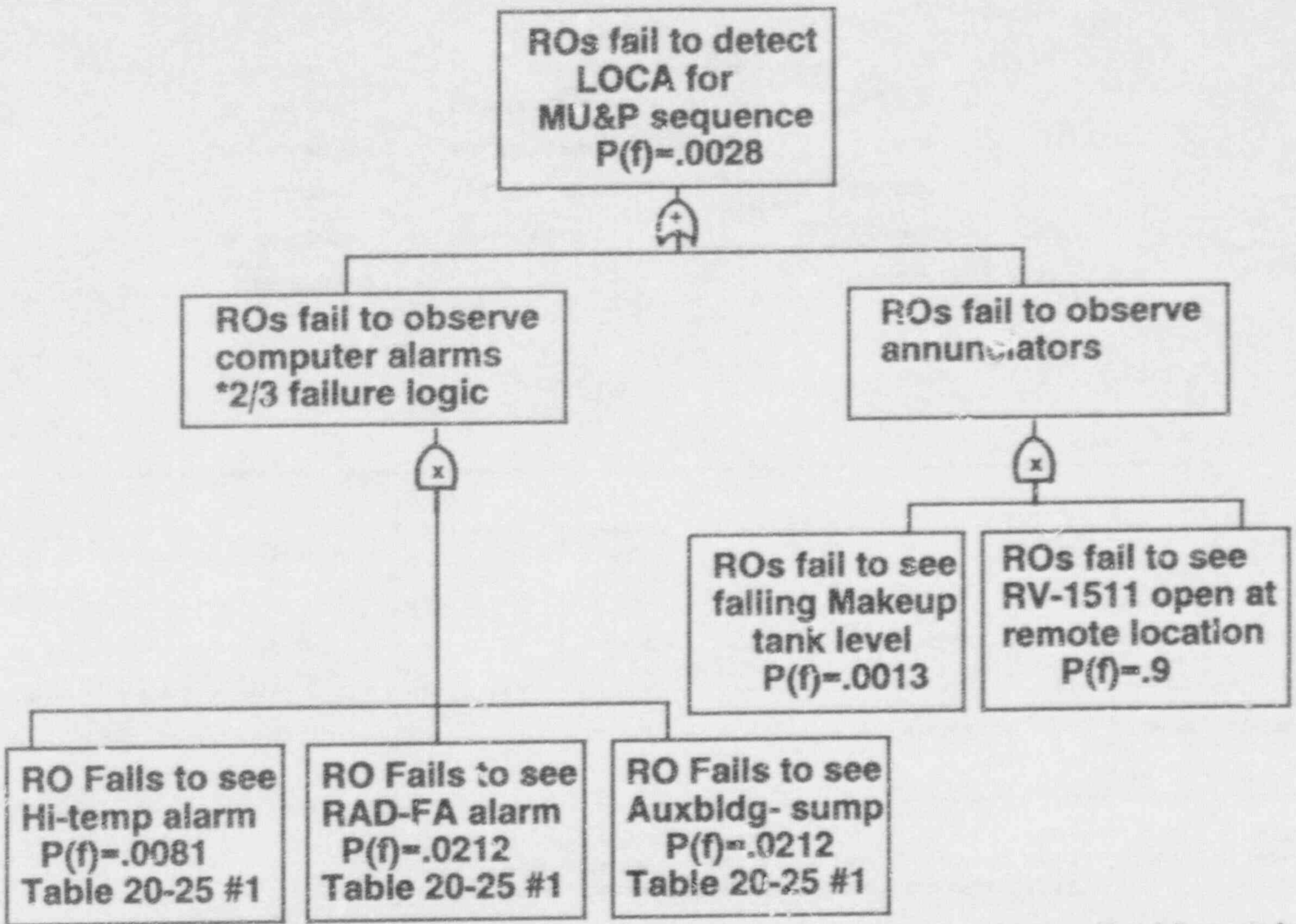
Table E9: HEPS for HM2 MU Operators Fail to Close HP-2A

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	RO fails to close HP-2A (Omission)	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.0038	0.0038	3.0
B	RC fails to correctly close HP-2A (Commission)	0.003	3.0	T20-13 #6	SBS	1	T20-16 #2a	ZD	0.0038	0.0038	3.0

Table E10: Failure Paths and Total Failure Probabilities

HM2-MU Operators Fail to Close HP-2A

	Failure Path	Calculations	Results
1	A	0.0037'9	0.0038
2	aB	0.003749	0.0038
		Total Failure Probability	0.008
		Error Factor (IRRAS 4.0)	2.27



* Recognition requires observation of 2 computer-based alarms and 1 of 2 available annunciators. High failure rate for observation of 1511 opening is based on the need for personnel to happen to be in the aux building concurrent with the incident, since only local indication is available; Nominal mean HEPs from Table E11 modified for recovery actions.

Figure 9: HRA Fault Tree for HD2-MU, Operator Fails to Detect ISLOCA

Table E11: HEPS for HD2-MU Operator Fails to Detect LOCA

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	ROs fail to observe HI-temp alarm	0.05	5.0	T20-25 #1	DYN	1	T20-16 #3	ZD	0.080692	0.080692	5.0
B	ROs fail to observe RAD-FA-alarm	0.05	5.0	T20-25 #1	DYN	1	T20-16 #3	MD	0.080692	0.212021	5.6
C	ROs fail to observe Aux bldg sump	0.05	5.0	T20-25 #1	DYN	1	T20-16 #3	MD	0.080692	0.212021	5.0
D	ROs fail to observe MU tank level decrease	0.005	10.0	T20-23 #6k	DYN	1	———	ZD	0.013317	0.013317	10.0
E	ROs fail to observe RV1511 open at remote location	0.9	1.0	Eng. Judgement	DYN	1	———	ZD	0.9	0.9	1.0

Cutset Analysis		Frequency	Total Mean
MU-tank level	RV1511 open	1.2E-003	
RAD-FA alarm	Aux. Bldg. Sump	4.5E-004	
Hi-temp alarm	Aux. Bldg. Sump	1.7E-004	
Hi-temp alarm	RAD-FA alarm	1.7E-004	
			2.75E-003

Operators fail to diagnose
ISLOCA by not implementing
procedures and selecting
effective course of action
(see Figure E11)

P(f) = .006

Figure 10: HRA Fault Tree for HDA2-MU, Operators Fail to Diagnose ISLOCA

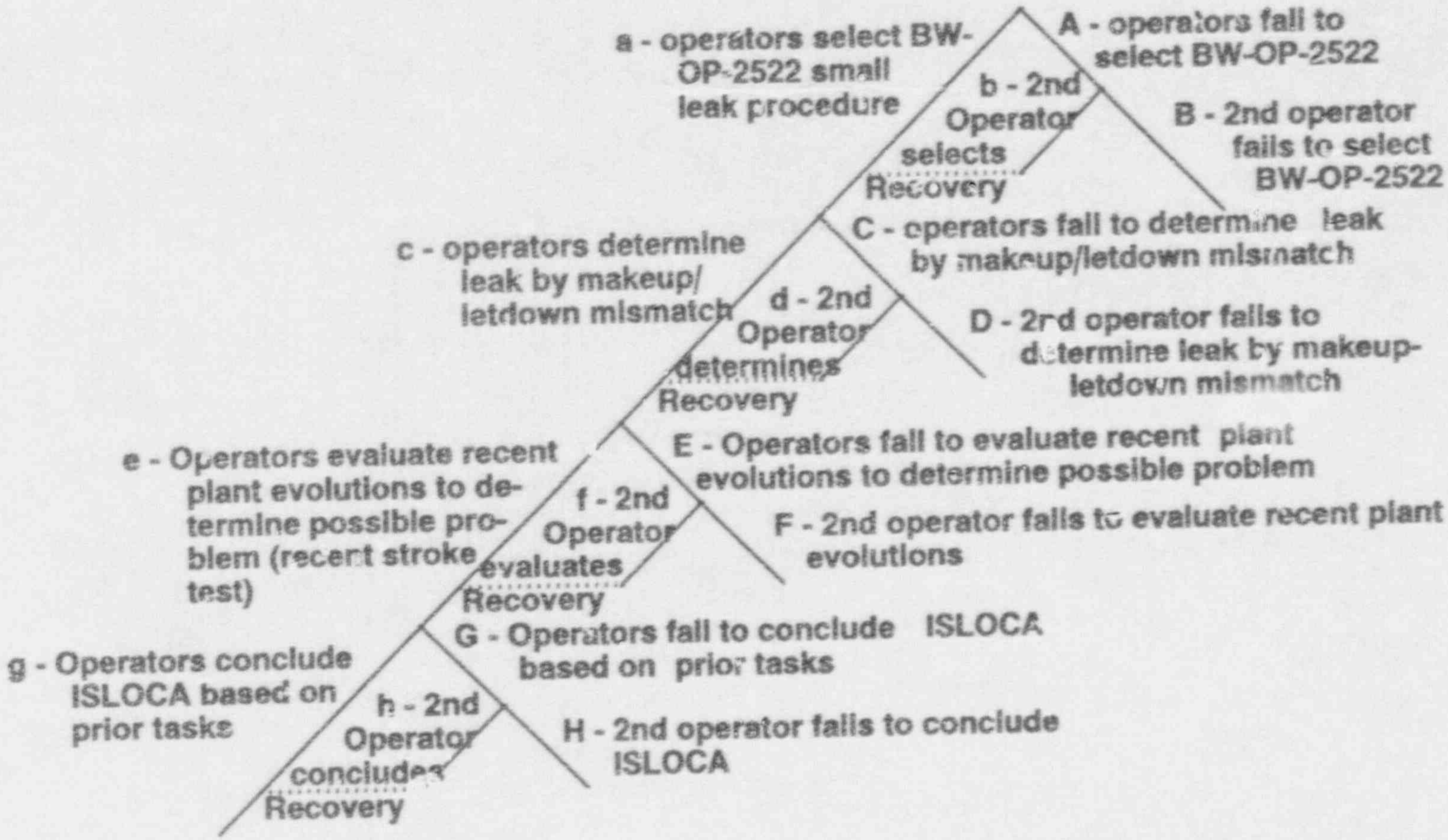


Figure 11: HRA Event Tree for HDA2-MU, Operators Fail to Daignose ISLOCA

Table E12: HEPS for HDA2-MU ROs Fail to Diagnose ISLOCA

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	ROs fail to select BW-OP-2522; small leak procedure	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
B	2nd RO fails to select small leak procedure	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
C	ROs fail to determine leak by makeup\letdown mismatch	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
D	2nd RO fails to determine leak from makeup\letdown mismatch	0.1	5.0	T20-22 3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
E	ROs fail to evaluate recent plant evolutions to determine problem	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
F	2nd RO fails to evaluate recent plant evolutions to determine problem	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0
G	ROs fail to conclude ISLOCA based on prior tasks	0.0001	30.0	T20-3 #5	SBS	1	T20-16 #2a	ZD	0.000847	0.000847	30.0
H	2nd RO fails to conclude ISLOCA based on prior tasks	0.1	5.0	T20-22 #3	SBS	1	T20-16 #2a	ZD	0.161383	0.161383	5.0

Table E13: Failure Paths and Total Failure Probabilities

for HDA2-MU

	Failure Path	Calculations	Results
1	AB	0.013317 x 0.161383	0.002149
2	AbCD	0.013317 x 0.013317 x 0.161383	0.000028
3	AbCdEF	0.013317 x 0.013317 x 0.013317 x 0.161383	*
4	AbCdEfGH	0.013317 x 0.013317 x 0.013317 x 0.000847 x 0.161383	*
5	AbCdeGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
6	AbcEF	0.013317 x 0.013317 x 0.161383	0.000028
7	AbcEfGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
8	AbceGH	0.013317 x 0.000847 x 0.161383	0.000001
9	aCD	0.013317 x 0.161383	0.002149
10	aCdEF	0.013317 x 0.013317 x 0.161383	0.000028
11	aCdEfGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
12	aCdeGH	0.013317 x 0.000847 x 0.161383	0.000001
13	acEF	0.013317 x 0.161383	0.002149
14	acEfGH	0.013317 x 0.000847 x 0.161383	0.000001
15	aceGH	0.000847 x 0.161383	0.000136
		Total Failure Probability	0.006
		Error Factor (IRRAS 4.0)	14.93

Operators fail to isolate,
crew fails to close HP-2A

P(f) = .002 (EF=3)*
Tables 20-12(#3) & 20-16(#4)

* Value allows for assumption that part of the time the crew will continue the HP2A stroke test, use the heuristic of undoing what they have just done, or isolating based on cues in the procedure. Stress was assumed to be at moderate levels during the transient.

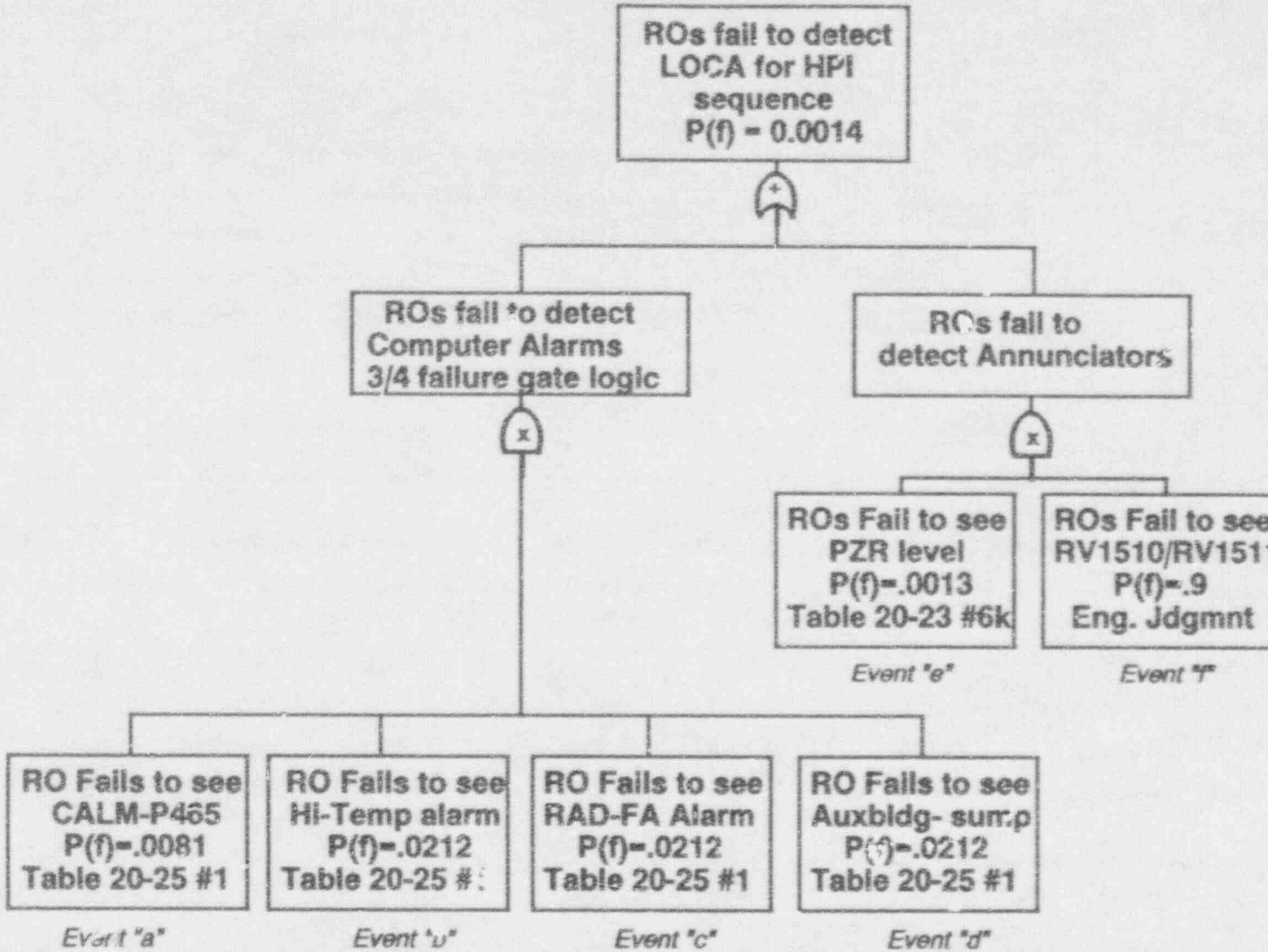
Figure 12: HRA Fault Tree for HI2-MU, Operators Fail to Isolate MU&P

Fail to close vent line to BWST

P(f) = .0013*

* The suggested rate factors in use of BW-OP-04004 & BW-OP-00008 procedures for locked valve verification which include independent verification & sign-off. The logic tree & calculations are similiar to that presented in Figures 4, 5, &6.

Figure 13: HRA Fault Tree for HV1-HP, Fail to Close Vent Line to BWST



NOTE: Nominal mean HEPs from Table E14 have been modified to account for recovery actions by other operators.

Figure 14: HRA Fault Tree for HD2-HP, Operators Fail to Detect ISLUCA

Table E14: HEPS for HD2-HP Operators Fail to Detect LOCA

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
a	Operators fail to observe CALM-P465	0.05	5.0	T20-25 #1	DYN	1	T20-16 #2	ZD	0.080692	0.080692	5.0
b	Operators fail to observe HI-Temp alarm	0.05	5.0	T20-25 #1	DYN	1	T20-16 #3	MD	0.080692	0.212021	5.0
c	Operators fail to observe RAD-FA alarm	0.05	5.0	T20-25 #1	DYN	1	T20-16 #3	MD	0.080692	0.212021	5.0
d	Operators fail to observe Aux. building sump alarm	0.05	5.0	T20-25 #1	DYN	1	T20-16 #3	MD	0.080692	0.212021	5.0
e	Operators fail to observe pressurizer level	0.005	10.0	T20-23 #6k	DYN	1	T20-16 #3	ZD	0.013317	0.013317	10.0
f	Operators fail to observe RV1510 or RV1511	0.9	1.0	Eng. Jdgmnt.	DYN	1	---------	ZD	0.9	0.9	1.0

Cutset Analysis			Frequency	Total Mean
PZR level	RV1510/1511		1.2E-003	
Hi-temp alarm	RAD-FA alarm	Aux bldg. sump	9.5e-006	
CALM P465	Hi-temp alarm	RAD-FA Alarm	3.6E-006	
CALM P465	RAD-FA alarm	Aux. bldg. sump	3.6E-006	
CALM P465	Hi-temp alarm	Aux. bldg. sump	3.6E-006	
				1.4E-003

Operators fail to diagnose ISLOCA by not implementing procedures and selecting effective course of action (see Figure E11)

P(f) = .006

* Logic structure is as presented in Figure E11 and calculated as presented in Tables E12 and E13.

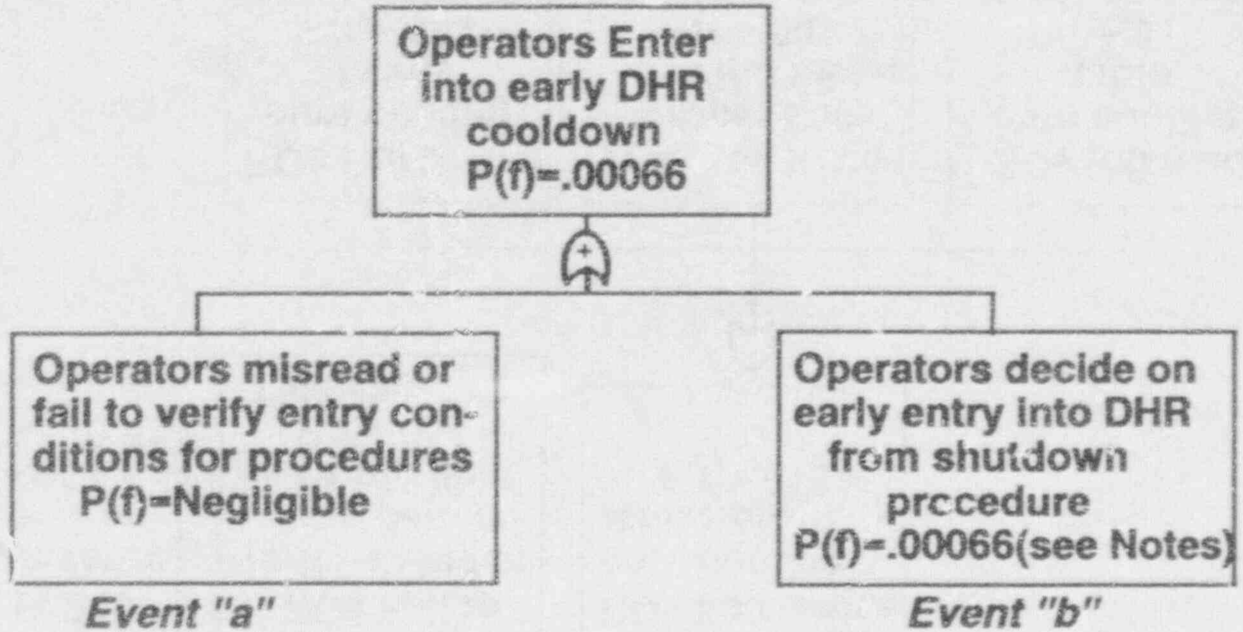
Figure 15: HRA Fault Tree for HDA2-HP, Fail to Diagnose ISLOCA

Operators fail to isolate, crew fails to close HP-2B

P(f) = .002 (EF=3)*
Tables 20-12(#3) & 20-16(#4)

*** Value includes estimates of "if there were means and methods available" what is the likelihood they could be successfully employed. This value has been modified for moderate stress due to a possible site evacuation being underway per procedure BWNPS-Eplan Rev. 13 (requirements to declare a site area emergency).**

Figure 16: HRA Fault Tree for HI2-HP, Operators Fail to Isolate, No MU&P



Notes: This cognitive action HEP was determined by engineering judgement and reflects the possibility of a joint decision by the SS and RO. The basis for the HEP estimate includes sanctioned jumpering of interlocks which exist in current SD procedures. Allowance has been made for a refusal by the I&C during the execution of this procedure.

Figure 17: HRA Fault Tree for DM1-SD, Operators Prematurely Open DH11 & DH12

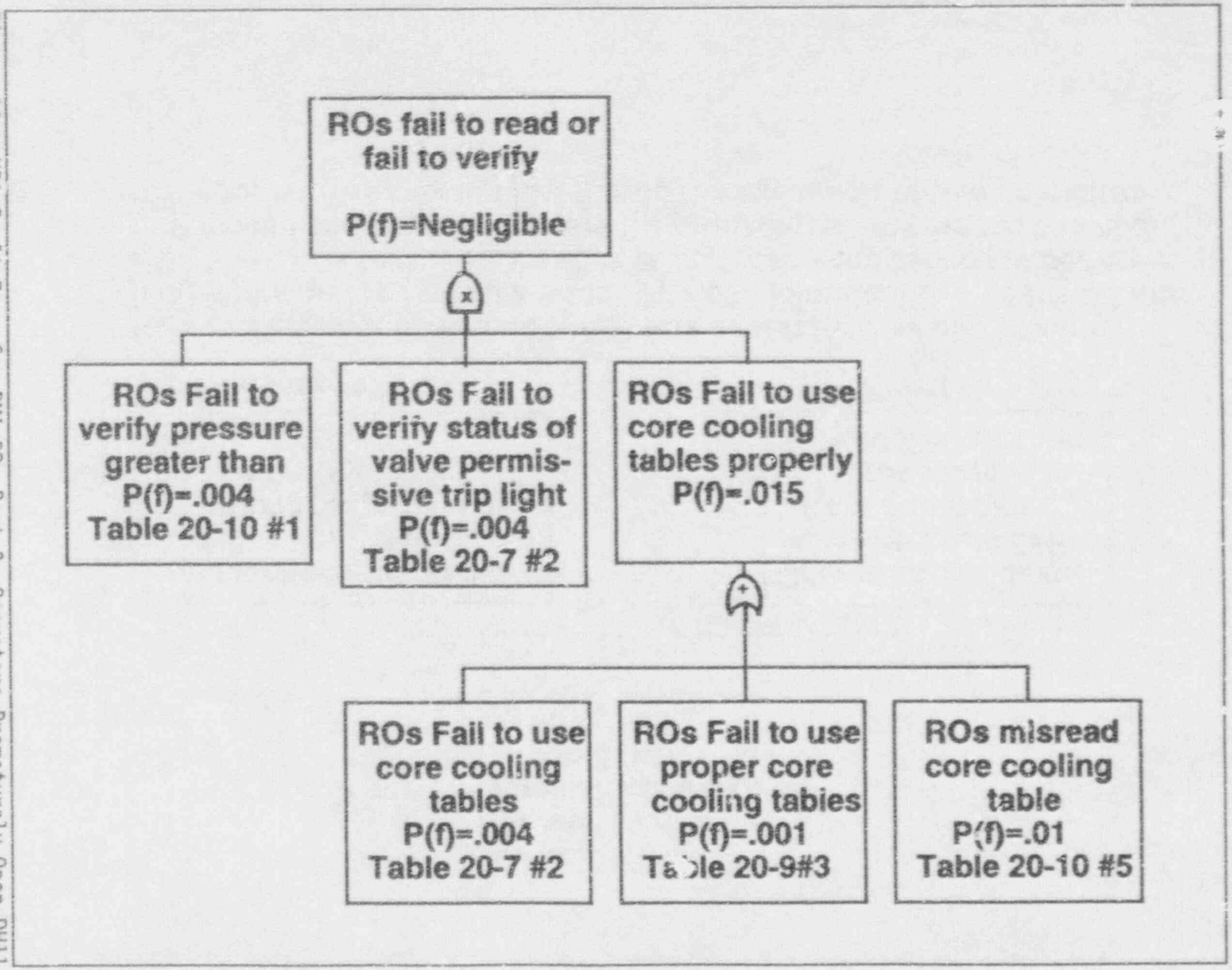
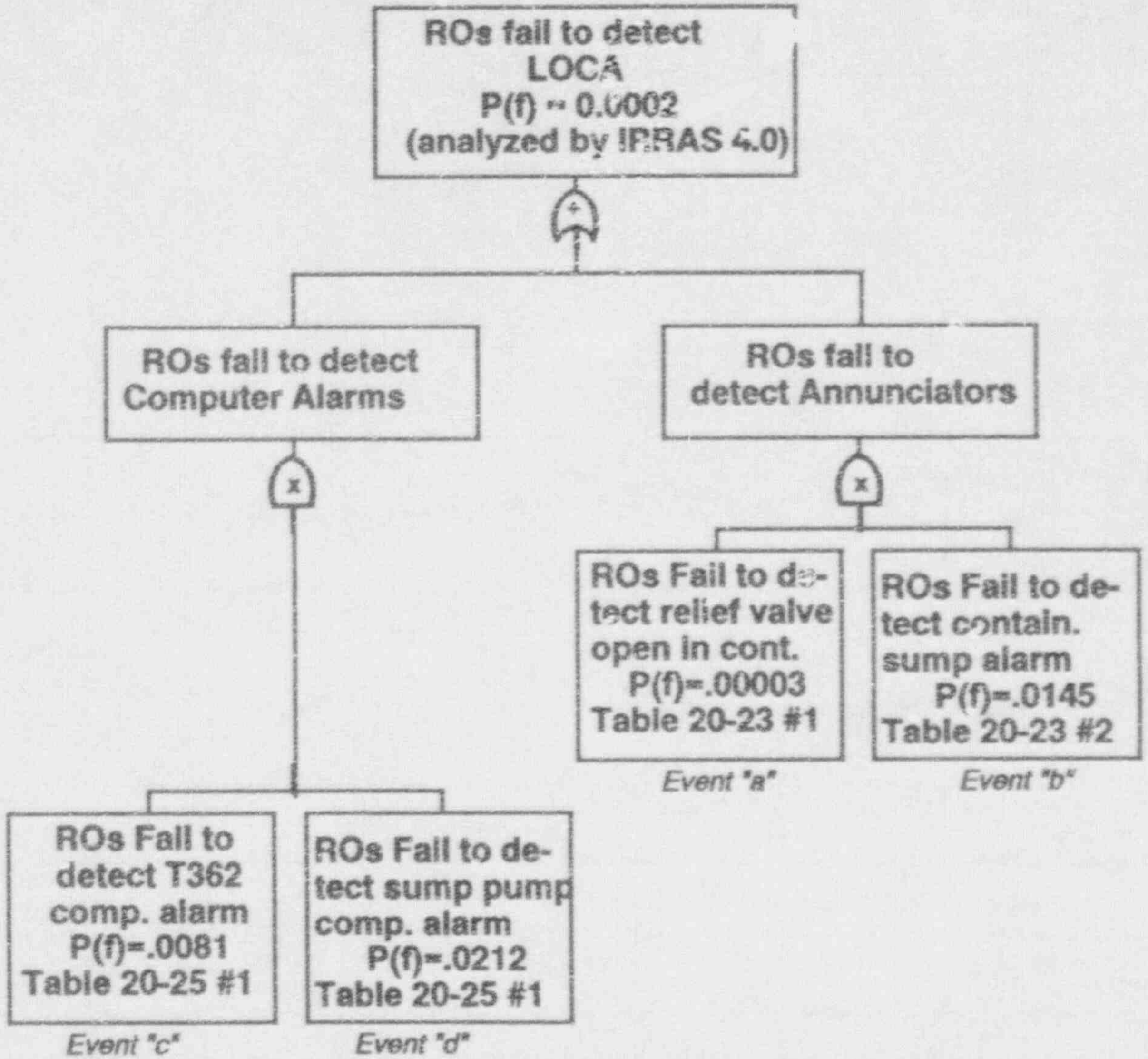


Figure 18: HRA Fault Tree for DMI-SD, Part 2, Operators Prematurely Open DH11 & DH12

Table E15: HEPS for DM1-SD Part 2, Operators Prematurely Open DH11 & DH12

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
a	Operators fail to verify pressure greater than allowed	0.003	3.0	T20-10 #1	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
b	Operators fail to verify status of valve permissive trip light	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
c	Operators fail to use core cooling tables	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
d	Operators fail to reference proper core cooling tables	0.001	3.0	T20-9 #3	SBS	1	T20-16 #2a	ZD	0.001249	0.001249	3.0
e	Operators misread core cooling table	0.01	3.0	T20-10 #5	SBS	1	T20-16 #2a	ZD	0.012498	0.012498	3.0



NOTE: Nominal mean HEPs from Table E16 have been modified to account for recovery actions by other operators.

Figure 19: HRA Fault Tree for DD2-SD, Operators Fail to Detect ISLOCA

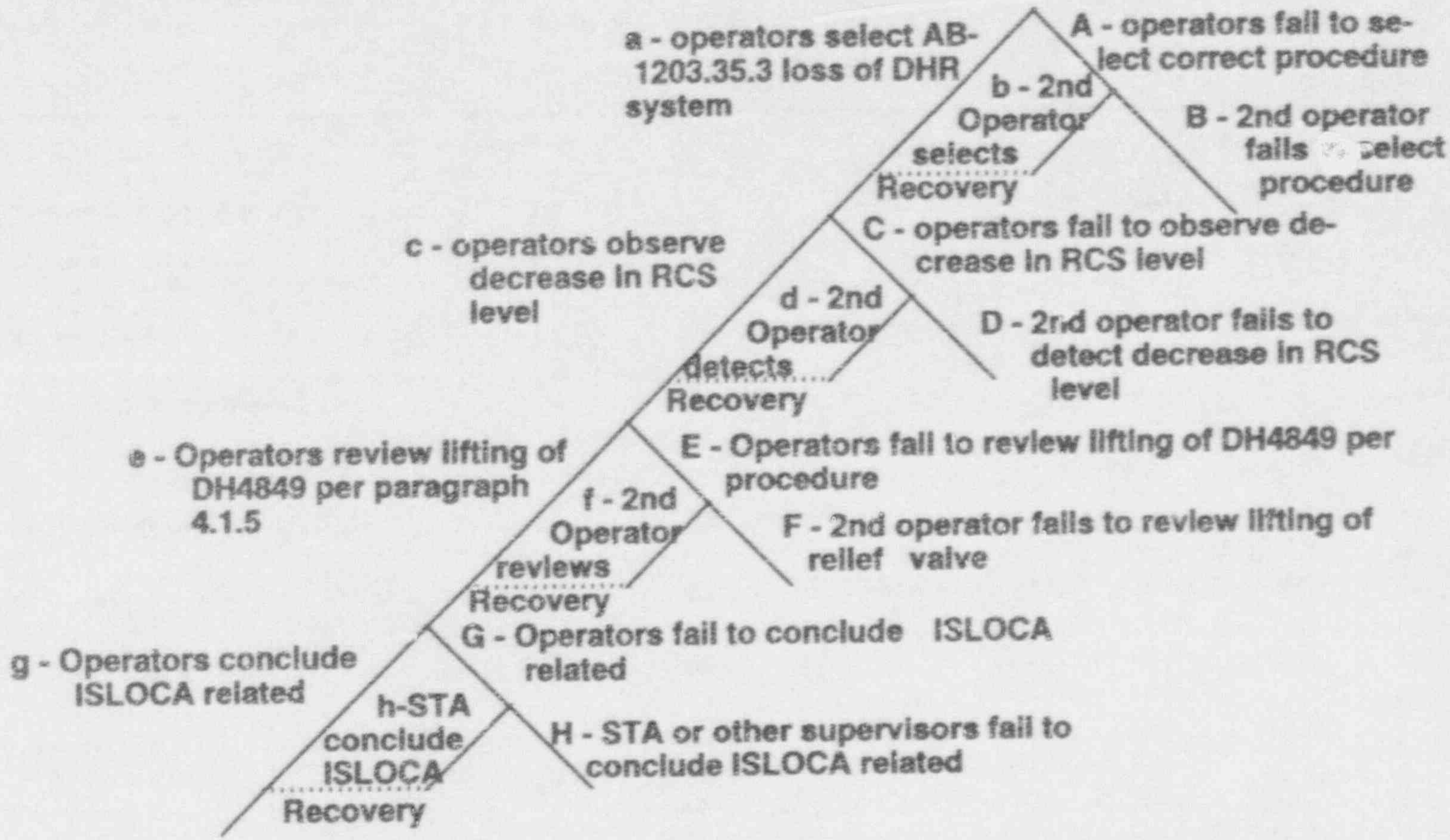
Table E16: HEPS for DD2-SD Operators Fail to Detect LOCA												
	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
a	Operators fail to detect relief valve #4849 opens in containment	0.0001	10.0	T20-[illegible] #1	SBS	1	T20-16 #2a	ZD	0.000266	0.000266	10.0
	Operators fail to detect containment sump level alarm	0.001	10.0	T20-23 #2	SBS	1	T20-16 #2a	MD	0.002663	0.14514	5.0
c	Operators fail to detect T362 computer alarm	0.05	5.0	T20-25 #1	SBS	1	T20-16 #3	ZD	0.080697	0.080692	5.0
d	Operators fail to detect sump pump computer alarm	0.05	5.0	T20-25 #1	SBS	1	T20-16 #3	MD	0.080692	0.212021	5.0

Operators fail to diagnose ISLOCA by not implementing procedures and selecting effective course of action (see Figure E21)

P(f) = .006

* Logic structure is as presented in Figure E21 and calculated as presented in Tables E17 and E18.

Figure 20: HRA Fault Tree for DDA1-SD, Operators Fail to Diagnose ISLOCA



Includes the heuristic of "undoing what was just done", as well as working through the appropriate procedure steps

Figure 21: HRA Event Tree for DDA1-SD, Operators Fail to Diagnose ISLOCA

Table E17: HEPS for DDA1-SD; Operators Fail to Diagnose ISLOCA

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to select AB1203.35.6; loss of DHR system	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
B	Second operator fails to select correct procedure	0.1	5.0	T20-22 #1	SBS	1	T20-16 #1	ZD	0.161383	0.161383	5.0
C	Operators fail to observe decrease in RCS level	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
D	Second operator fails to observe decrease in RCS level	0.1	5.0	T20-22 #1	SBS	1	T20-16 #1	ZD	0.161383	0.161383	5.0
E	Operators fail to review lifting of relief valve D44879 per paragraph 4.1.5	0.005	10.0	T20-6 #4	SBS	1	T20-16 #2a	ZD	0.013317	0.013317	10.0
F	Second operator fails to review lifting of relief valve D44879	0.1	5.0	T20-22 #1	SBS	1	T20-16 #1	ZD	0.161383	0.161383	5.0
G	Operators fail to conclude ISLOCA	0.0001	30.0	T20-3 #5	SBS	1	T20-16 #2a	ZD	0.000847	0.000847	30.0
H	STA or other supervisor fails to conclude event is ISLOCA related	0.1	5.0	T20-22 #1	SBS	1	T20-16 #1	ZD	0.161383	0.161383	5.0

Table E18: Failure Paths and Total Failure Probabilities

DDA1-SD; Operators Fail to Diagnose ISLOCA

	Failure Path	Calculations	Results
1	AB	0.013317 x 0.161383	0.002149
2	AbCD	0.013317 x 0.013317 x 0.161383	0.000028
3	AbCdEF	0.013317 x 0.013317 x 0.013317 x 0.161383	*
4	AbCdEfGH	0.013317 x 0.013317 x 0.013317 x 0.000847 x 0.161383	*
5	AbCdeGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
6	AbcEF	0.013317 x 0.013317 x 0.161383	0.000028
7	AbcEfGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
8	AbceGH	0.013317 x 0.000847 x 0.161383	0.000001
9	aCD	0.013317 x 0.161383	0.002149
10	aCdEF	0.013317 x 0.013317 x 0.161383	0.000028
11	aCdEfGH	0.013317 x 0.013317 x 0.000847 x 0.161383	*
12	aCdeGH	0.013317 x 0.000847 x 0.161383	0.000001
13	acEF	0.013317 x 0.161383	0.002149
14	acEfGH	0.013317 x 0.000847 x 0.161383	0.000001
15	aceGH	0.000847 x 0.161383	0.000136
		Total Failure Probability	**0.006**
		Error Factor(IRRAS 4.0)	**14.93**

Crew fails to isolate
for DHR-SD ISLOCA
by closing MOV
P(f)=.008#

Notes: * HEP from Table 20-7 #2; Modified for high stress, Table 20-16 #4

Figure 22: HRA Fault Tree for DI2-SD Crew Fails to Isolate ISLOCA

Table E19: HEPS for DI2-SD Crew Fails to Isolate DHR Cooldown (SD)

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Crew fails to isolate for DHR-SD ISLOCA by closing MOV	0.003	3.0	T20-7 #2	DYN	2	T20-16 #4	ZD	0.004	0.008	3.0

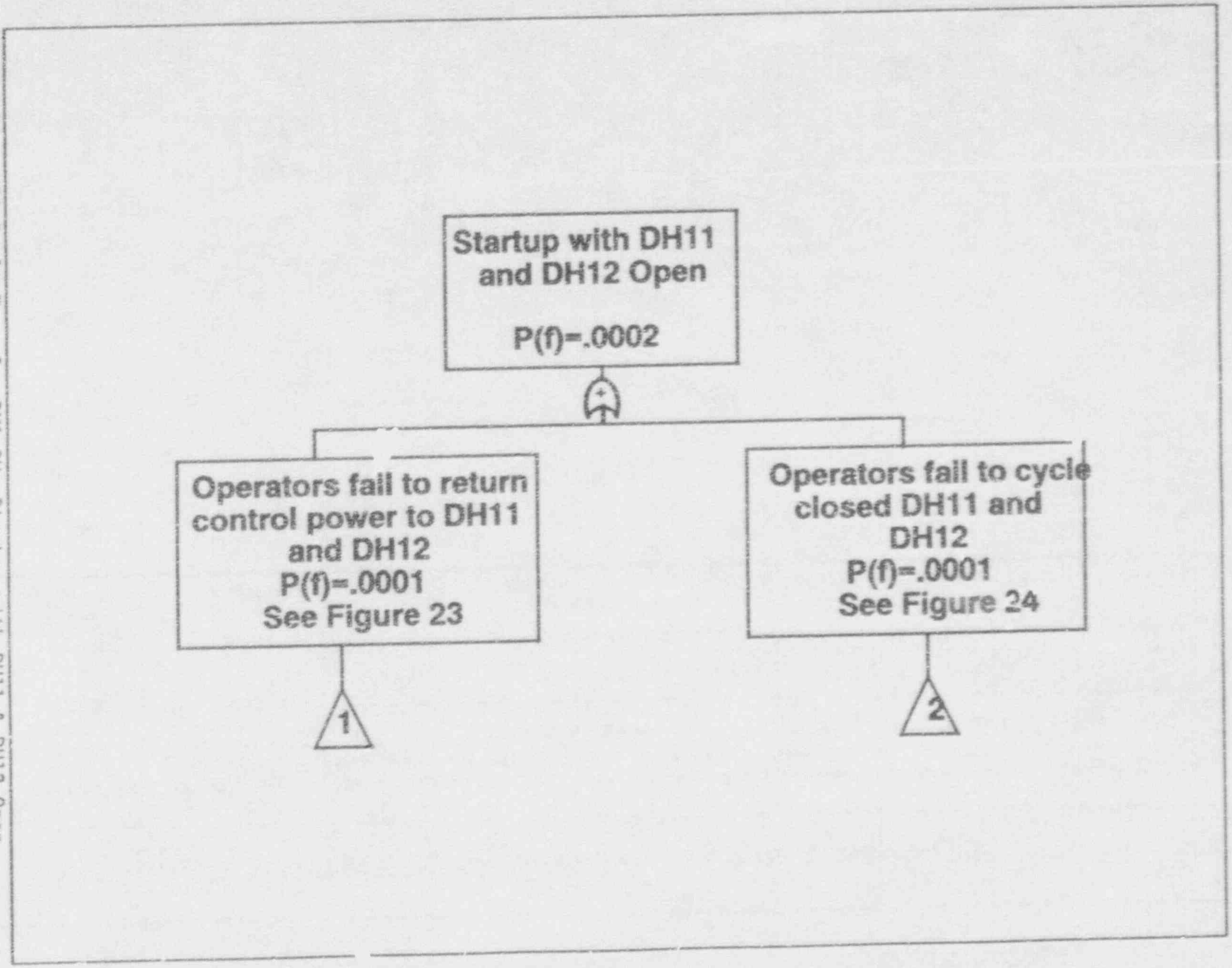


Figure 23: HRA Fault Tree for DM1-SU, Startup with DH11 & DH12 Open

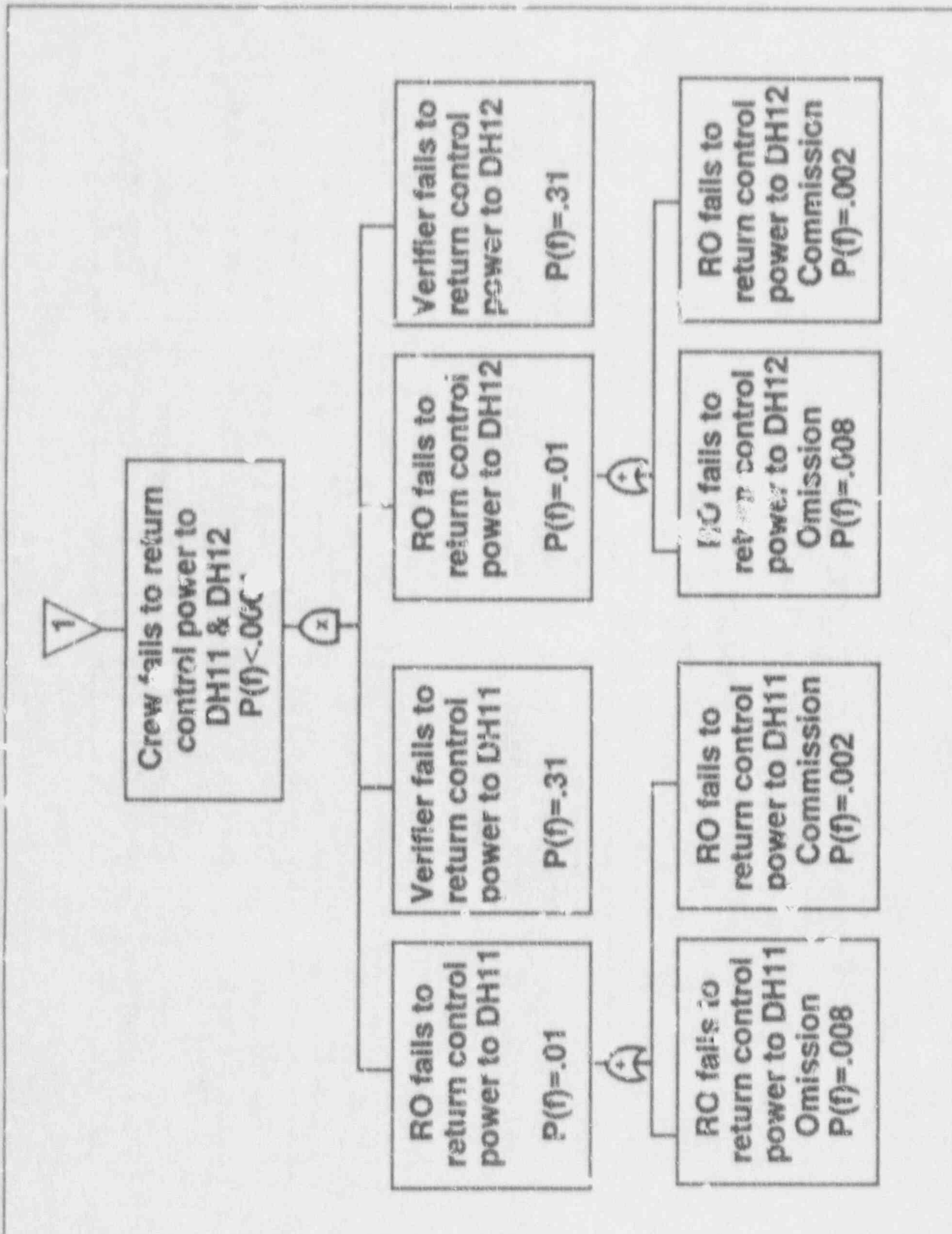


Figure 24: HRA Fault Tree for DM1-SU (cont.)

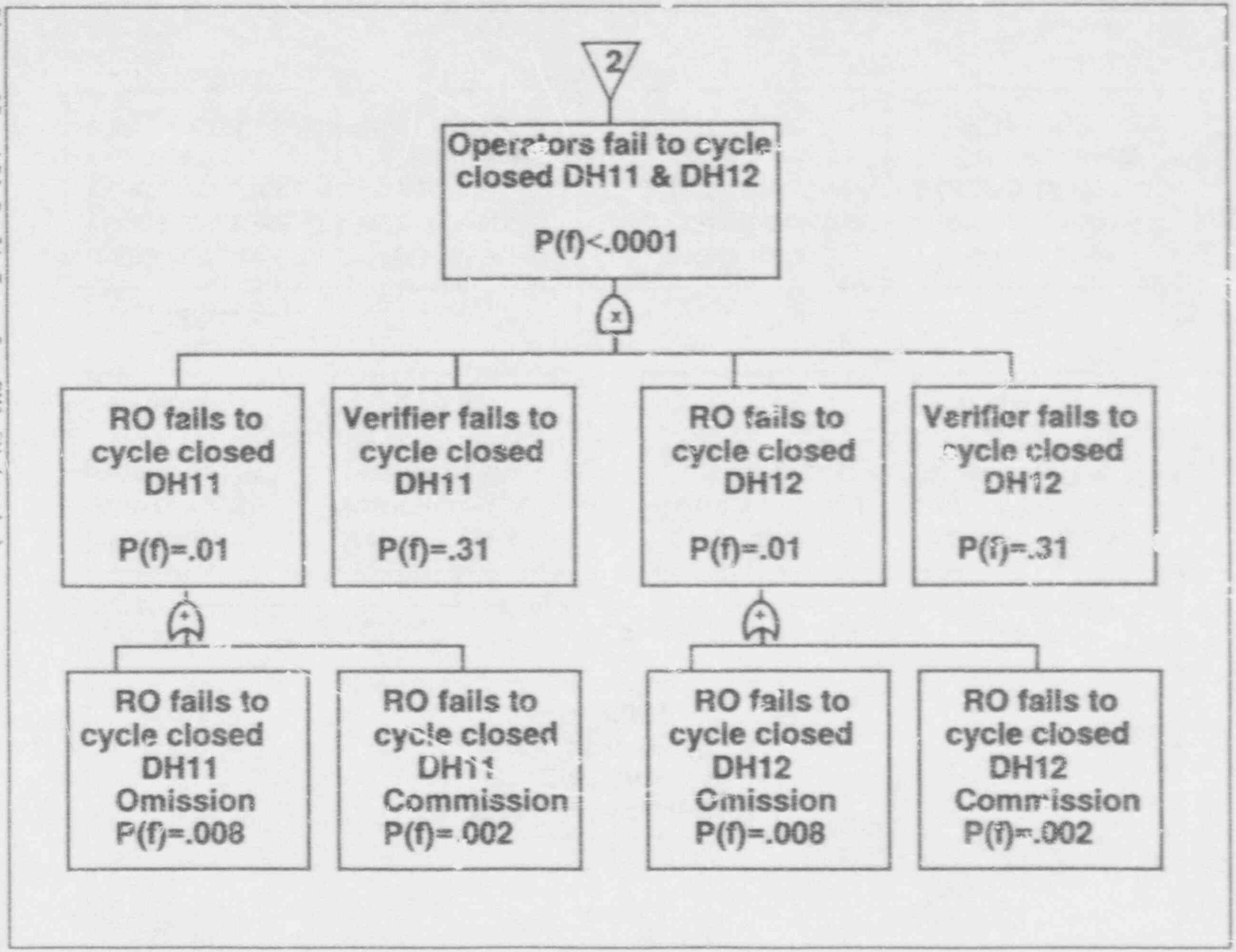


Figure 25: HRA Fault Tree for DMI-SU (cont.)

Table E20: HEPS for DM1-SU, Start-up with DH11 & DH12 Open

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	RO fails to return control power to DH11 (omission)	0.003	3.0	T20-6 #2	SBS	2	T20-16 #4	ZD	0.003749	0.008	3.0
B	RO fails to return control power to DH11 (commission)	0.001	3.0	T20-12 #3	SBS	2	T20-16 #4	ZD	0.001249	0.002	3.0
C	Verifier fails to return control power to DH11	0.1	5.0	T20-22 #1	SBS	2	T20-16 #4	HD	0.161383	0.661383	5.0
										* 0.31	2.3
D	RO fails to return control power to DH12 (omission)	0.003	3.0	T20-6 #2	SBS	2	T20-16 #4	ZD	0.003749	0.008	3.0
E	RO fails to return control power to DH12 (commission)	0.001	3.0	T20-12 #3	SBS	2	T20-16 #4	ZD	0.001249	0.002	3.0
F	Verifier fails to return control power to DH12	0.1	5.0	T20-22 #1	SBS	2	T20-16 #4	HD	0.161383	0.662383	5.0
										* 0.31097	2.3
G	RO fails to cycle closed DH11 (omission)	0.003	3.0	T20-6 #2	SBS	2	T20-16 #4	ZD	0.003749	0.008	3.0
H	RO fails to cycle closed DH11 (commission)	0.001	3.0	T20-12 #3	SBS	2	T20-16 #4	ZD	0.001249	0.002	3.0
I	Verifier fails to cycle closed DH11	0.1	5.0	T20-22 #1	SBS	2	T20-16 #4	HD	0.161383	0.661383	5.0
										* 0.31	2.3
J	RO fails to cycle closed DH12 (omission)	0.003	3.0	T20-6 #2	SBS	2	T20-16 #4	ZD	0.003749	0.008	3.0
K	RO fails to cycle closed DH12 (commission)	0.001	3.0	T20-12 #3	SBS	2	T20-16 #4	ZD	0.001249	0.002	3.0

Table E20: HEPS for DM1-SU , Start-up with DH11 & DH12 Open (cont.)

Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
L. Verifier fails to cycle closed DH12	0.1	5.0	T20-22 #1	SBS	2	T20-16 #4	HD	0.161343	0.661	5.0
									* 0.31	2.3

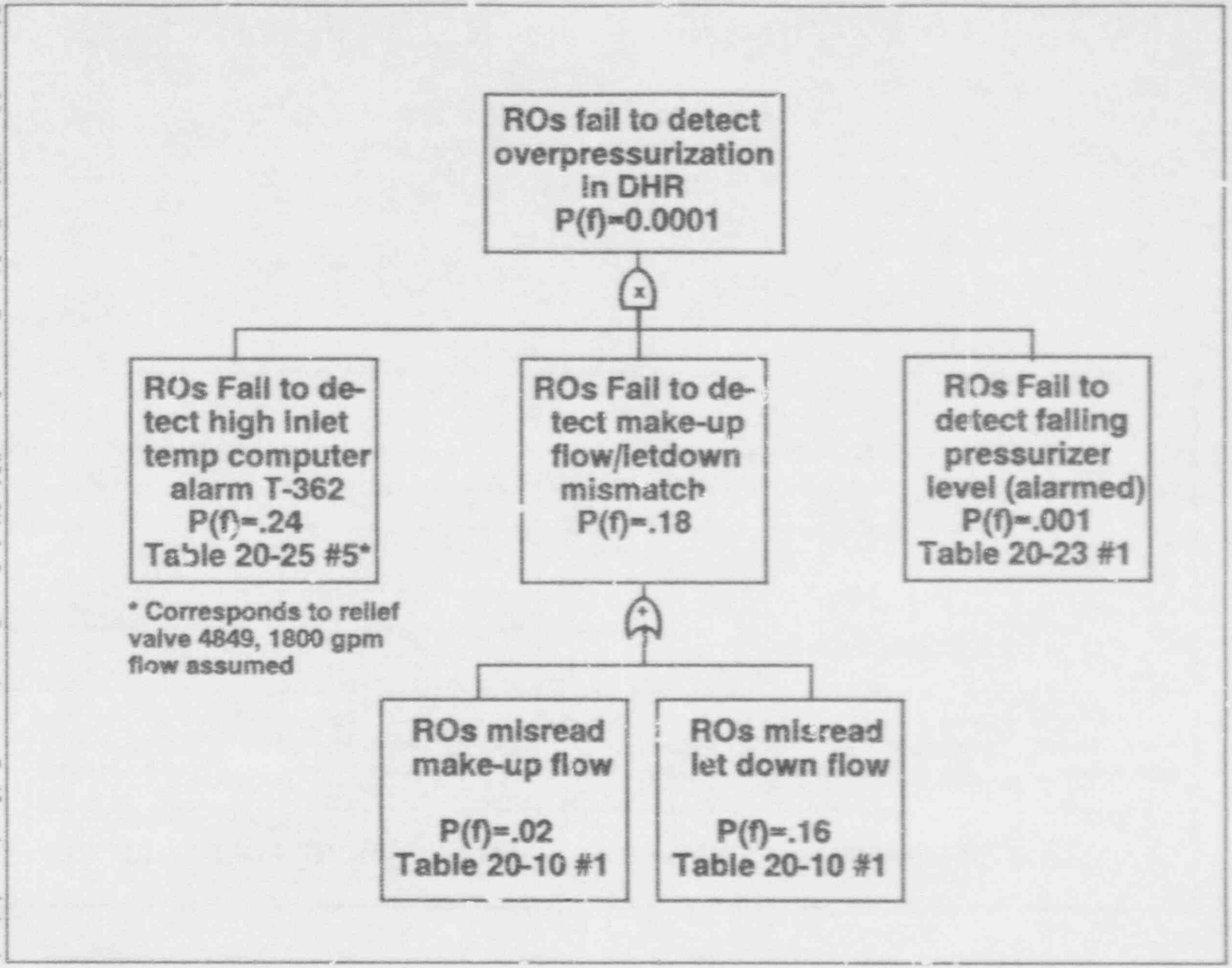


Figure 26: HRA Fault Tree for CDI-SU-A,C, Operators Fail to detect Overpressurization in DHR System, Relief Valve Opens

Table E21: HEPS for DD1-SU-A,C Operators Fail to Detect Overpressurization

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to detect high inlet temp computer alarm T362	0.05	5.0	T20-25 #1	DYN	5	T20-16 #5	ZD	0.080692	0.403459	5.0
										* 0.242542	2.7
B	Operators misread make-up flow	0.003	3.0	T20-10 #1	DYN	5	T20-16 #5	ZD	0.003749	0.018747	3.0
C	Operators misread let down flow	0.003	3.0	T20-10 #1	DYN	5	T20-16 #5	MD	0.003749	0.158926	3.0
D	Operators fail to detect decreasing pressurizer level	0.0001	10.0	T20-23 #1	DYN	5	T20-16 #5	ZD	0.000266	0.001331	10.0

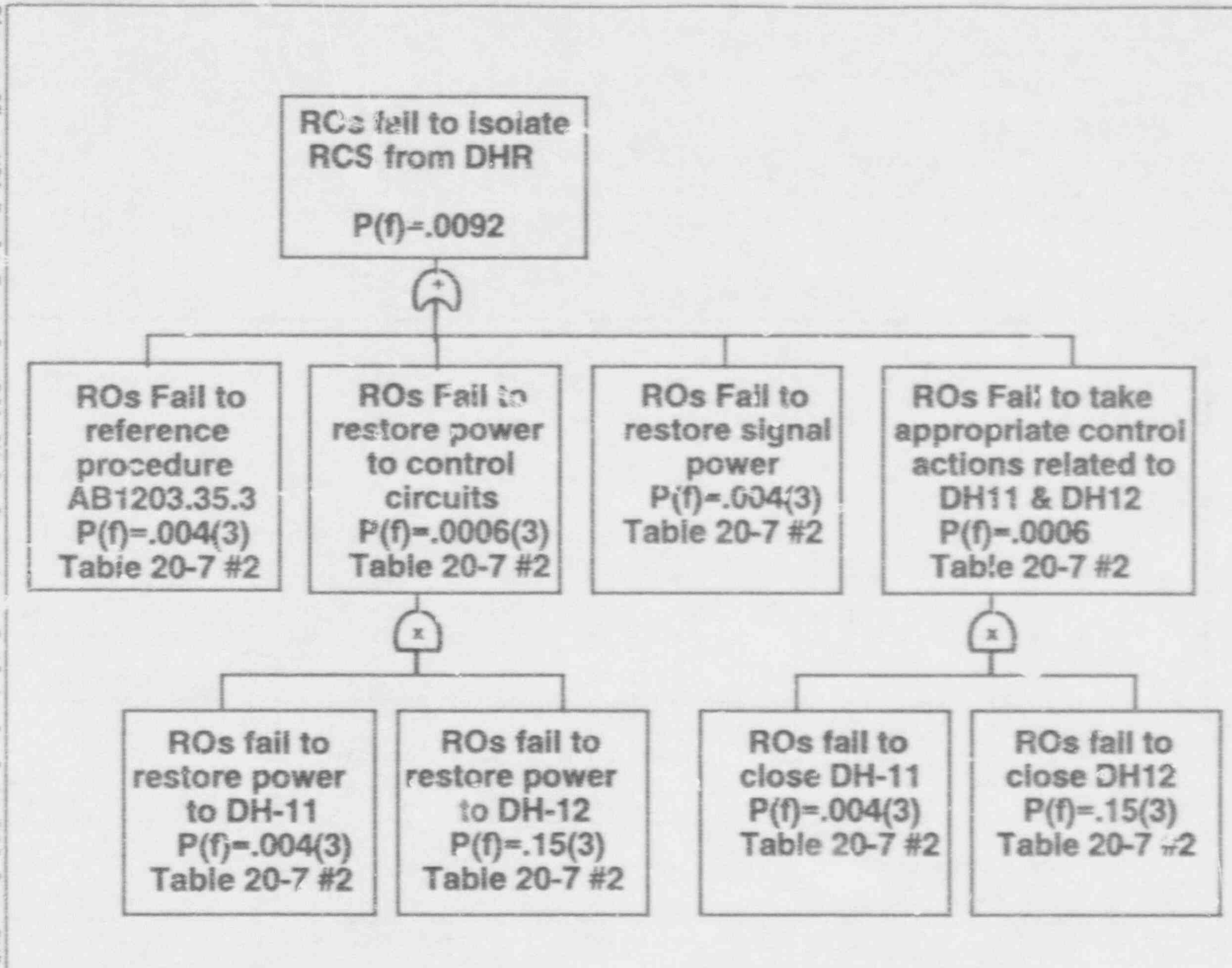


Figure 27: HRA Fault Tree for DI1-SU-C, Operators Fail to Isolate RCS from DHR, Relief Valve Opens

Table E22: HEPS for DH-SU-C Operators Fail to Isolate DH11&12, Relief Valve Open

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to reference procedure AB1293.35.3 p(f)	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
B	Operators fail to restore signal power	0.063	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
C	Operators fail to restore power to control circuits for DH-11	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
D	Operators fail to restore control power to control circuits for DH-12	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	MD	0.003749	0.146071	3.0
E	Operators fail to close DH-11	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
F	Operators fail to close DH-12	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	MD	0.003749	0.146071	3.0

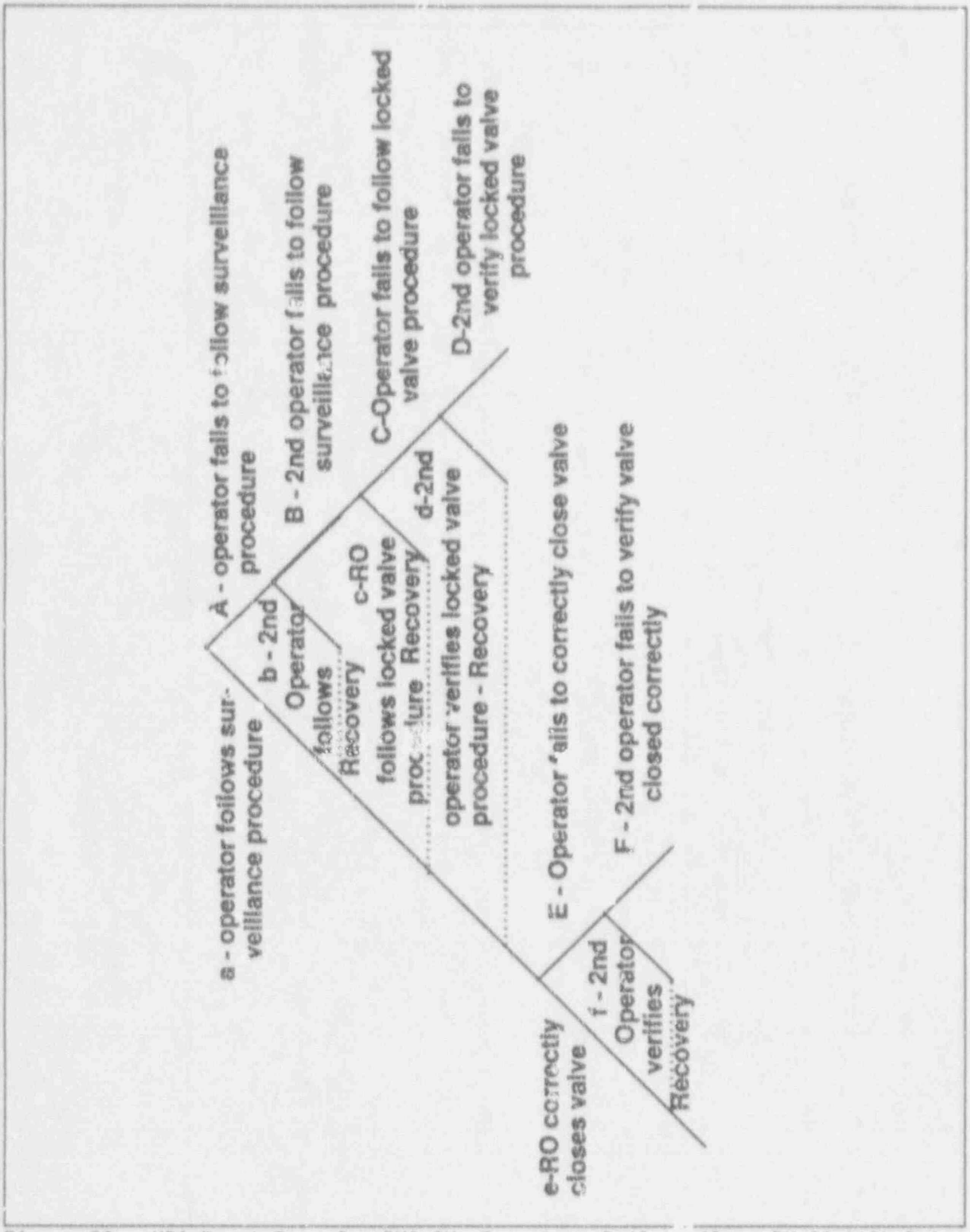


Figure 28: HRA Event Tree for DM2-SU, Operators Fail to Close DH 21 & 23 Prior to Start-up

Table E23: HEPS for DM2-SU Operators Fail to Close DH21 & 23 Prior to Start-up

Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A Operator fails to follow surveillance procedure	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
B Second operator fails to follow surveillance procedure	0.1	5.0	T20-22 #1	SBS	1	T20-16 #2a	MD	0.161383	0.281186	5.0
									* 0.188932	3.1
C Operator fails to follow locked valve procedure	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
D Second operator fails to verify locked valve procedure	0.1	5.0	T20-22 #1	SBS	1	T20-16 #2a	MD	0.161383	0.281186	5.0
									* 0.188932	3.1
E Operator fails to correctly close local valve	0.001	3.0	T20-13 #1	SBS	1	T20-16 #2a	ZD	0.001249	0.001249	3.0
F Second operator fails to verify valve closed correctly	0.1	5.0	T20-22 #1	SBS	1	T20-16 #2a	MD	0.161383	0.281186	5.0
									* 0.188932	3.1

Table E24: Failure Paths and Total Failure Probabilities

DM2-SU Operators Fail to Close DH21 & 23 Prior to Start-up

	Failure Path	Calculations	Results
1	ABCD	0.003749 x 0.188932 x 0.003749 x 0.188932	*
2	ABCdEF	0.003749 x 0.188932 x 0.003749 x 0.001249 x 0.188932	*
3	ABcEF	0.003749 x 0.188932 x 0.001249 x 0.188932	*
4	AbEF	0.003749 x 0.001249 x 0.188932	*
5	aEF	0.001249 x 0.188932	0.000236
		Total Failure Probability	0.000237
		Error Factor (IRRAS 4.0)	4.85

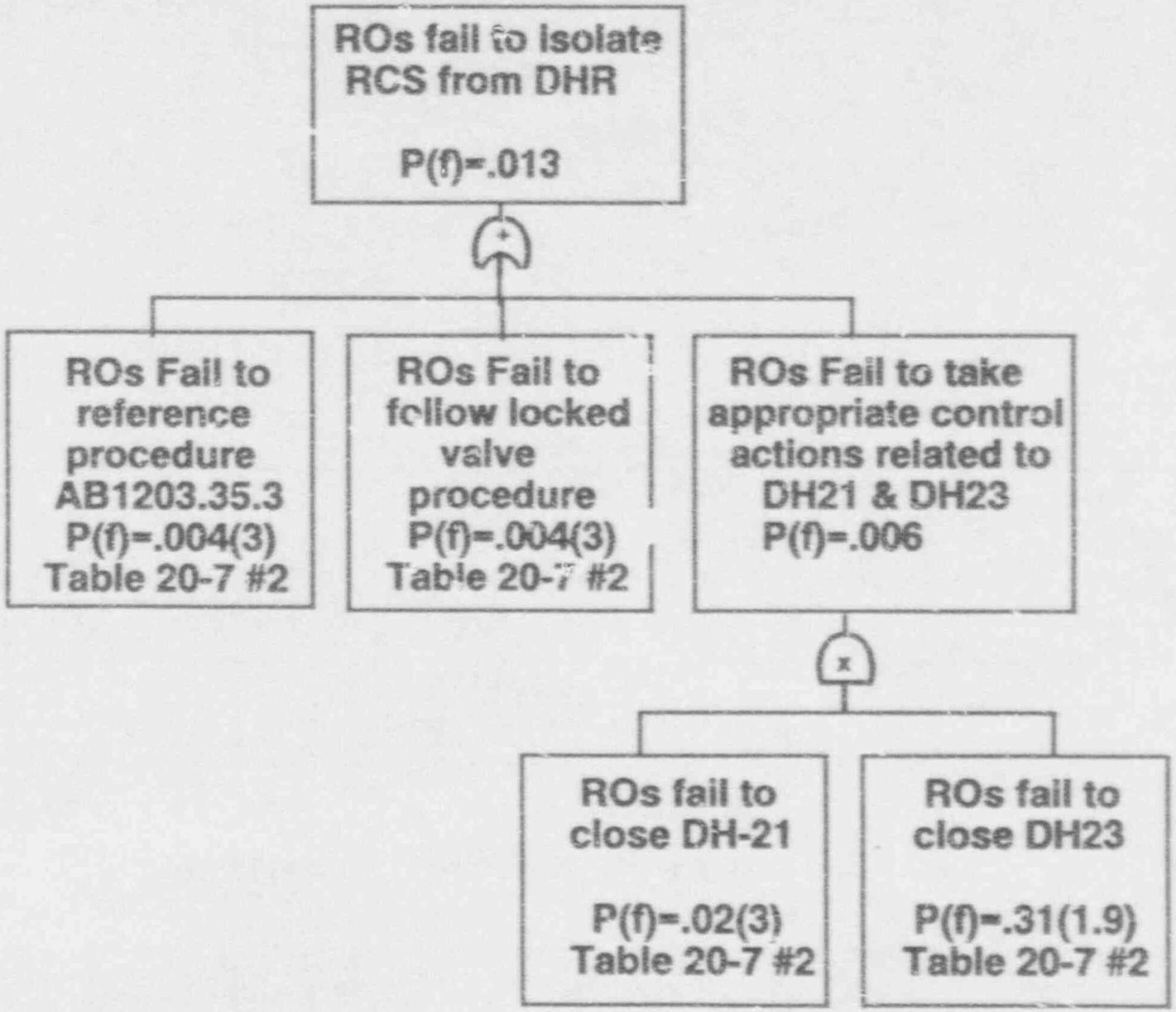


Figure 29: HRA Fault Tree for DII-SU-A, Operators Fail to Isolate RCS from DHR

Table E25: HEPS for DH-SU-A Operators Fail to Isolate RCS from DHR

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to reference procedure AB1203.35.3	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
B	Operators fail to follow locked valve procedures OP-04004 & OP-00008	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
C	Operators fail to correctly close DH-21	0.003	3.0	T20-7 #2	SBS	5	T20-16 #5	ZD	0.003749	0.018747	3.0
D	Operators fail to correctly close DH-23	0.003	3.0	T20-7 #2	SBS	5	T20-16 #5	HD	0.003749	0.509373	3.0
										* 0.312441	1.9

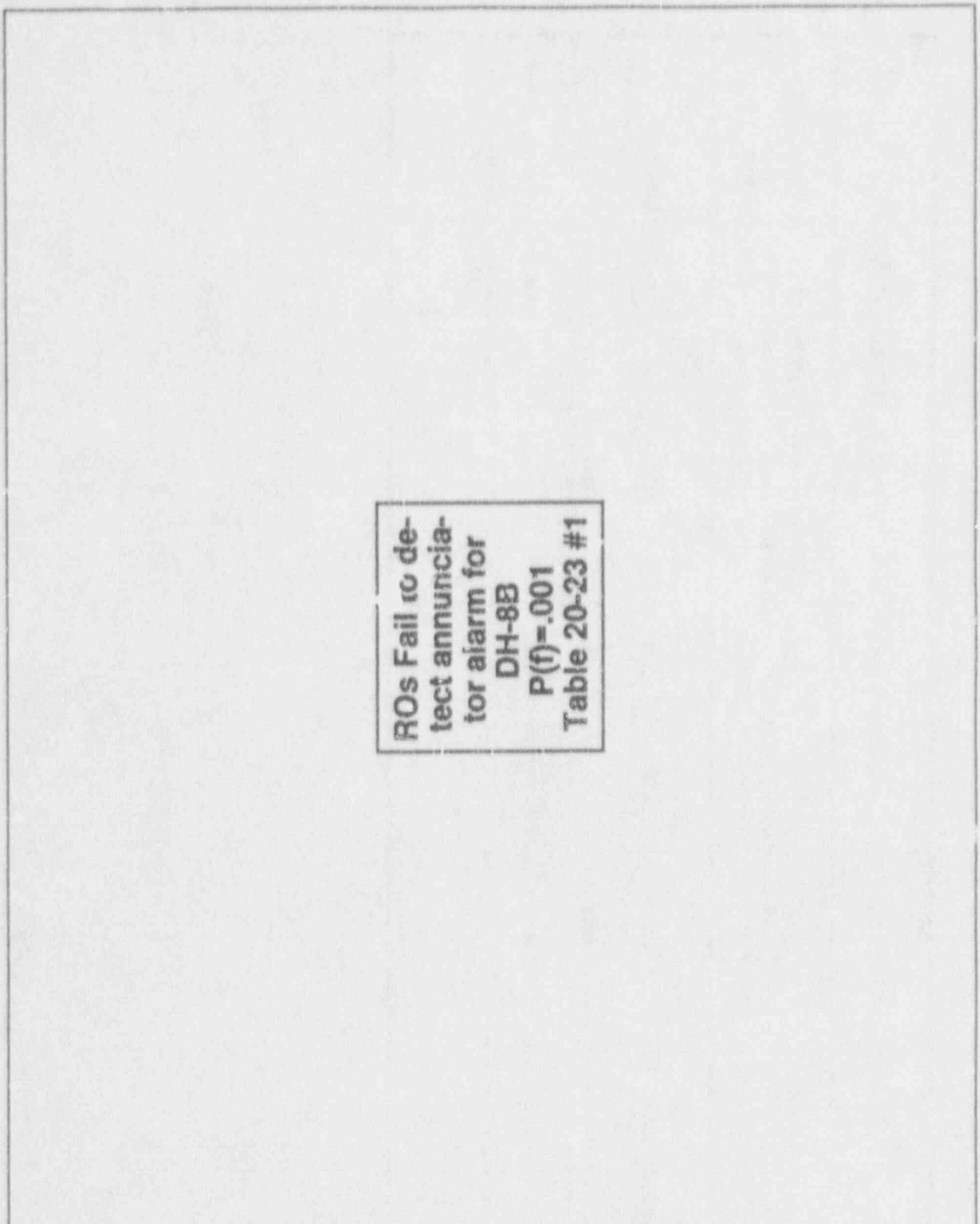


Figure 30: HRA Fault Tree for DDI-SU-B,D Operator Fail to Detect Overpressurization in DHR System, Relief Valves Closed

Table E26: HEPS for DD1-SU-B,D Operators Fail to Detect Overpressurization											
	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to detect annunciator alarm for DH-8B	0.0001	10.0	T20-23 #1	DYN	5	T20-16 #5	ZD	0.000266	0.001331	10.0

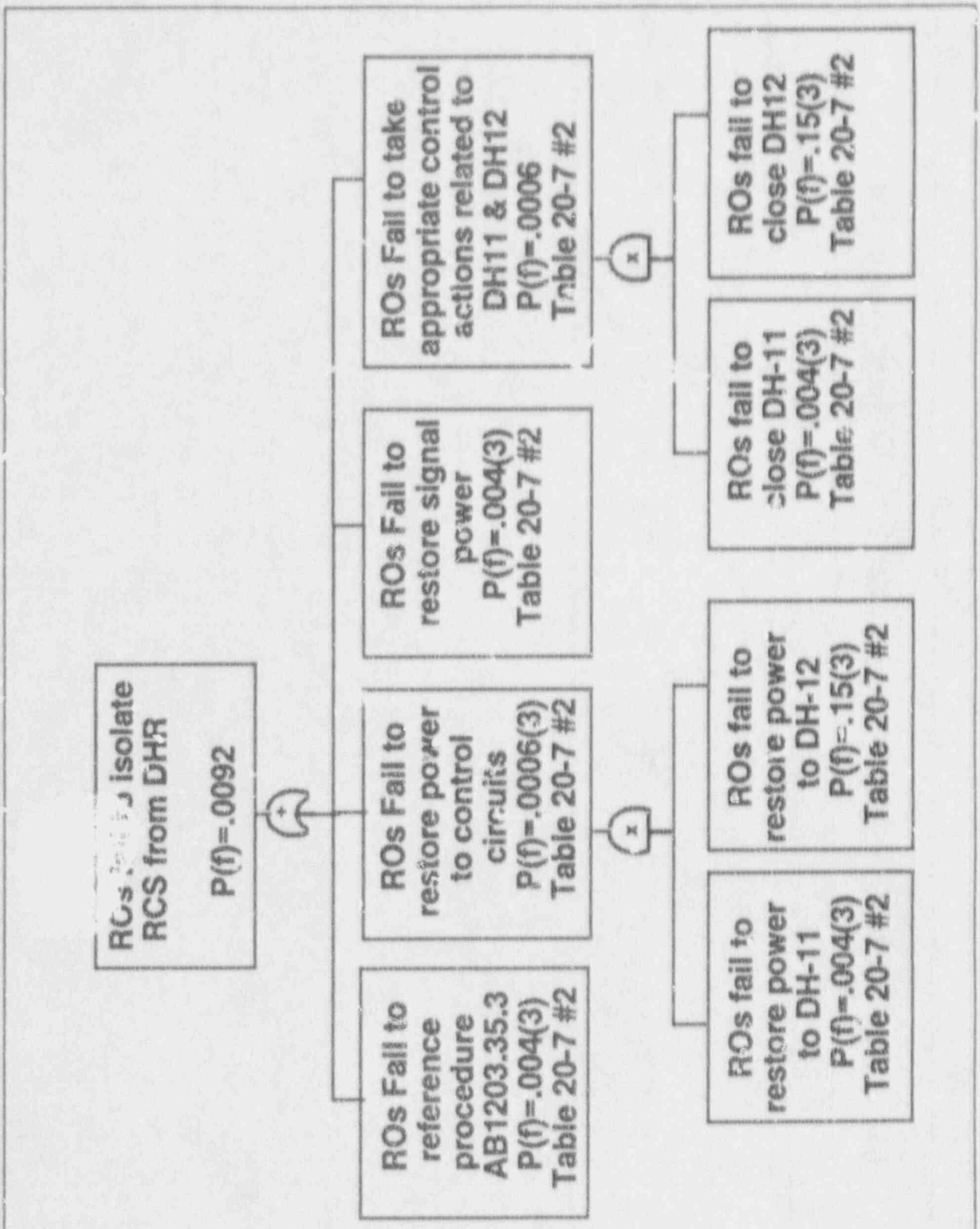


Figure 31: HRA Fault Tree for DH-SU-D Operators Fail to Isolate DH-11 & DH-12, Relief Valve Closed

Table E27: HEPS for DI1-SU-D Operators Fail to Isolate DH11&12, Relief Valve Closed

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to reference procedure AB1203.35.3 ,.(f)	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
B	Operators fail to restore signal power	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
C	Operators fail to restore power to control circuits for DH-11	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
D	Operators fail to restore control power to control circuits for DH-12	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	MD	0.003749	0.146071	3.0
E	Operators fail to close DH-11	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
F	Operators fail to close DH-12	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	MD	0.003749	0.146071	3.0

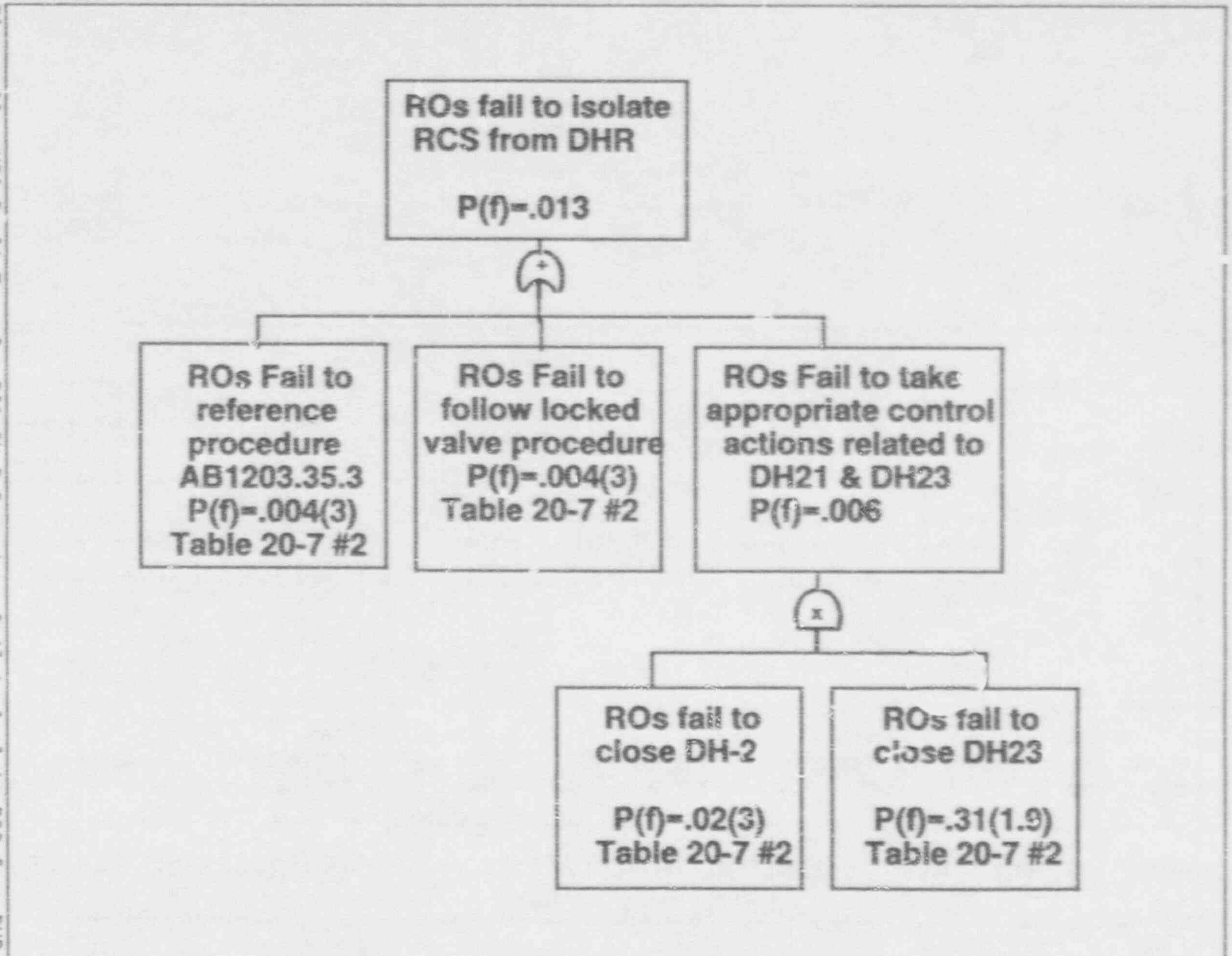
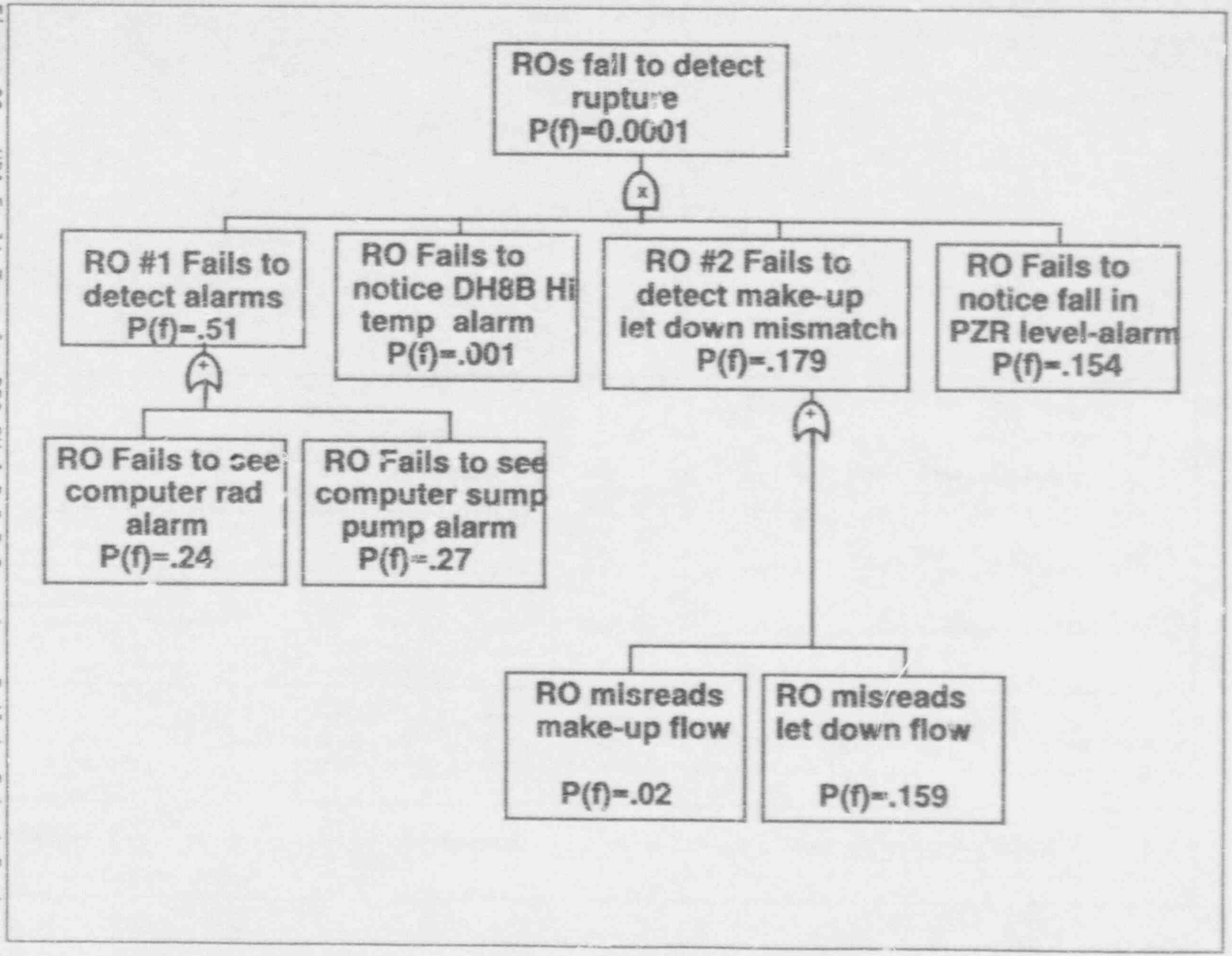


Figure 32: HRA Fault Tree for DI1-SU-B Operators Fail to Isolate RCS from DHR, DH21 & 23, Relief Valve Opens

Table E28: HEPS for DH-SU-B Operators Fail to Isolate RCS from DHR-Relief Open

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to reference procedure AB1203.35.3	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
B	Operators fail to follow locked valve procedures OP-04004 & OP-00008	0.003	3.0	T20-7 #2	SBS	1	T20-16 #2a	ZD	0.003749	0.003749	3.0
C	Operators fail to correctly close DH-21	0.003	3.0	T20-7 #2	SBS	5	T20-16 #5	ZD	0.003749	0.018747	3.0
D	Operators fail to correctly close DH-23	0.003	3.0	T20-7 #2	SBS	5	T20-16 #5	HD	0.003749	0.509373	3.0
										* 0.312441	1.9

Figure 33: HRA Fault Tree for DD2-SU-A,B,C,D Operators Fail to Detect Rupture



	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	RO fails to notice computer rad alarm	0.05	5.0	T20-25 #1a	DYN	5	T20-16 #5	ZD	0.080692	0.403459	5.0
										* 0.242542	2.7
B	RO fails to notice computer sump alarm	0.05	5.0	T20-25 #1a	DYN	5	T20-16 #5	MD	0.080692	0.488679	5.0
										* 0.274803	2.5
C	RO fails to notice DH-8B high temp alarm (annunciated)	0.0001	10.0	T20-23 #2	DYN	5	T20-16 #5	ZD	0.000266	0.001331	10.0
D	RO misreads make-up flow	0.003	3.0	T20-10 #1	DYN	5	T20-16 #5	ZD	0.003749	0.018747	3.0
E	RO misreads let-down flow	0.003	3.0	T20-10 #1	DYN	5	T20-16 #5	MD	0.003749	0.158926	3.0
F	RO fails to notice fall in pressurizer level (annunciated)	0.001	10.0	T20-10 #2	DYN	5	T20-16 #5	MD	0.002663	0.154272	5.0

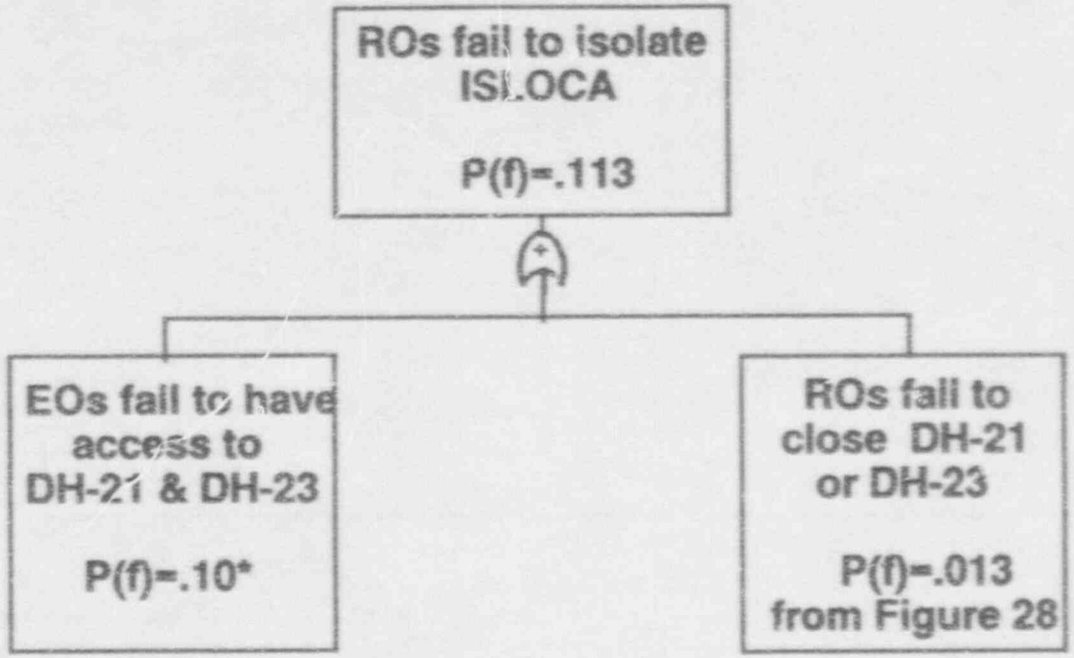
Table E29: HEPS for DD2-SU(A-D) ROs Fail to Detect Rupture;Relief Fails Open

Table E30: HRA Engineering Judgement for DAI-SU-A,B,C,D, Operators Fail to Diagnose ISLCOA, DHR Start-up Sequence

*Median HEP (EF)		*Mean HEP (EF)
DAI-SU-A	0.6 (10)	0.52 (1.6)
DAI-SU-B	0.8 (10)	0.59 (1.5)
DAI-SU-C	0.2 (10)	0.29 (2.5)
DAI-SU-D	0.4 (10)	0.43 (1.9)

***ASSUMPTIONS:** 1-Failure to implement procedure AB 1203.35.3; 2-Failure to interpret overpressurization signature; 3-Failure to recognize event signature as being ISLOCA; 4-Rates are estimates based on engineering judgement.

Figure 34: HRA Fault Tree for DI2-SU-A,B, Operators Fail to Isolate DH-21 & DH-23



* Assumes that, on average, 10% of the time the break will occur in the same room as DH-21 & DH-23; HEPs calculated from engineering judgement

Table E31: HEPS for DI2-SU-A,B Operators Fail to Isolate ISLOCA											
	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	EOs fail to gain access to DH-21 and DH-23	0.1	1.0	Eng. Jdgmnt.	SBS	1	———	ZD	0.1	0.1	1.0
B	ROs fail to close DH-21 or DH-23 (from Figure 28 & Table E25)	0.013	1.0	Figure 27	SBS	1	T20-16 #2a	ZD	0.013	0.013	1.0

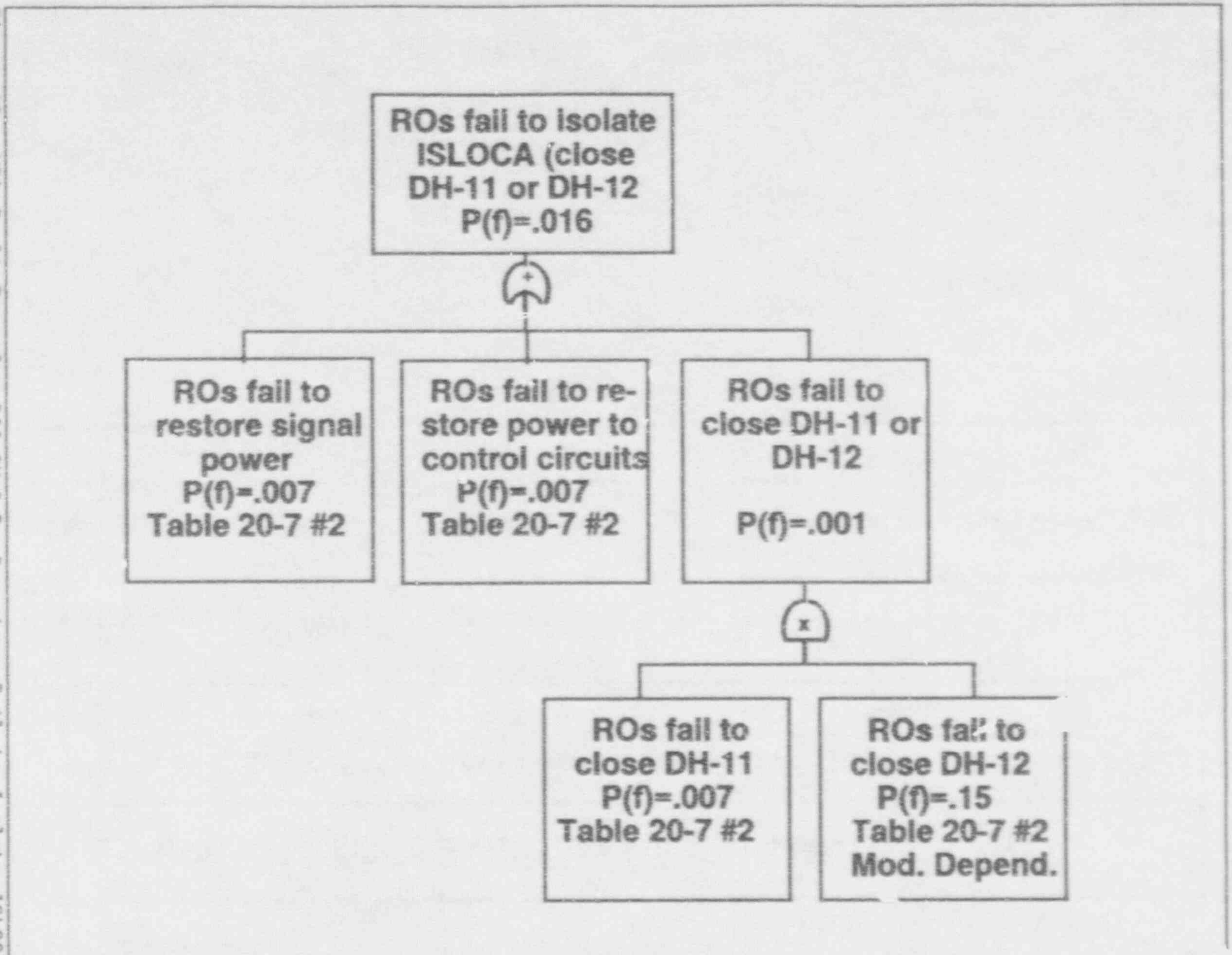


Figure 35: HRA Fault Tree for DI2-SU-C,D, Operators Fail to Isolate ISLOCA (Close DH-11 or DH-12)

Table E32: HEPS for DI2-SU-C,D Operators Fail to Isolate ISLOCA (Close DH 11& 12

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to restore signal power	0.003	3.0	T20-7 #2	SBS	2	T20-16 #4	ZD	0.003749	0.007499	3.0
B	Operators fail to restore power to control circuits	0.003	3.0	T20-7 #2	SBS	2	T20-16 #4	ZD	0.003749	0.007499	3.0
C	Operators fail to close DH-11	0.003	3.0	T20-7 #2	SBS	2	T20-16 #4	ZD	0.003749	0.007499	3.0
D	Operators fail to close DH-12	0.003	3.0	T20-7 #2	SBS	2	T20-16 #4	MD	0.003749	0.149284	3.0

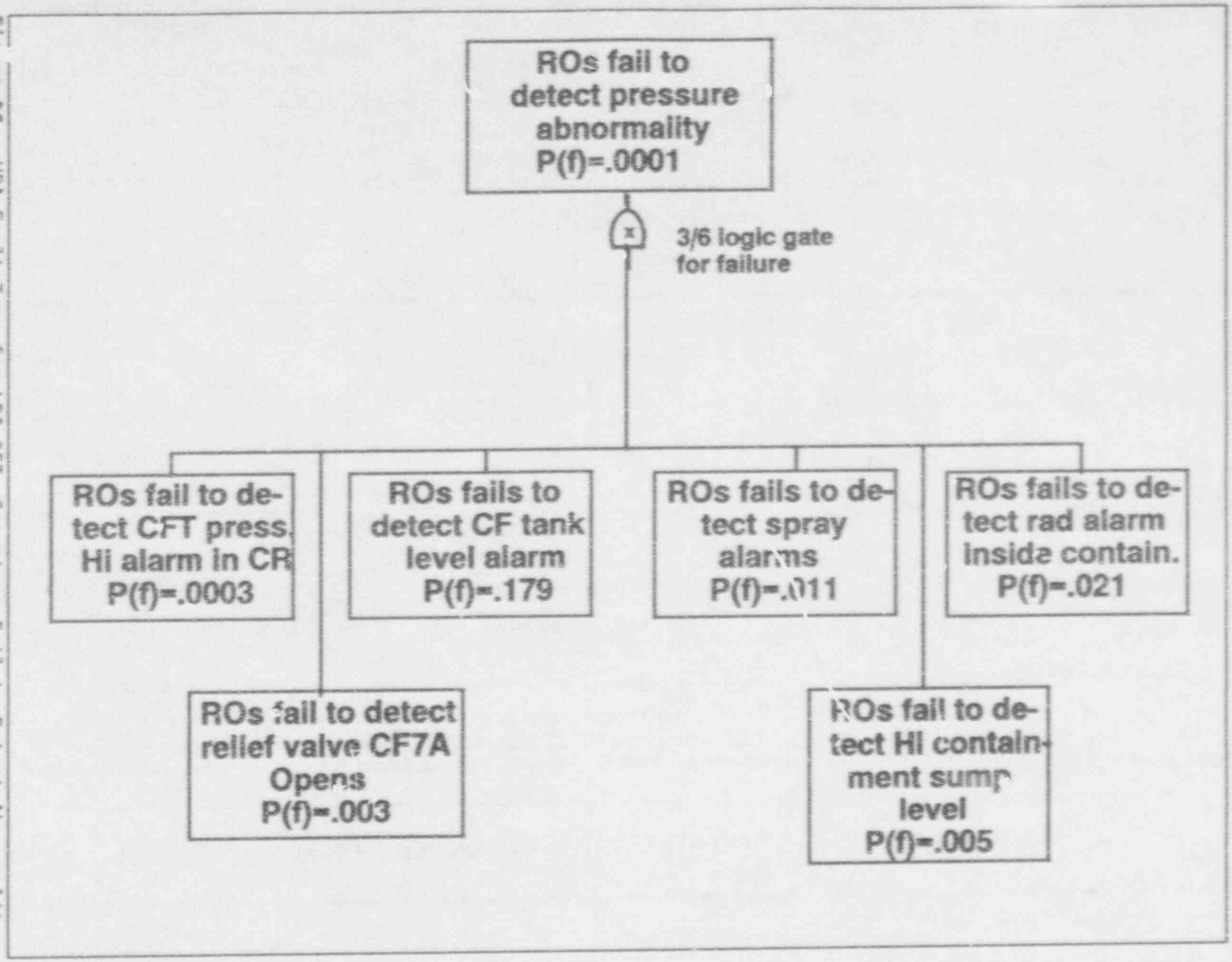


Figure 36: HRA Fault Tree for LD2-CFT, Operators Fail to Detect Abnormality

Table E33: HEPS for LD2-CFT Operators Fail to Detect Pressure Abnormality

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modif for PS	Modifier Source	THERP Depend-ency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to detect CFT pressure high alarm in control room	0.0001	10.0	T20-23 #6	DYN	1	T20-16 #2a	ZD	0.000266	0.000266	10.0
B	Operators fail to detect alarm for relief valve CF7A opens	0.001	10.0	T20-23 #6	DYN	1	T20-16 #2a	ZD	0.002663	0.002663	10.0
C	Operators fail to detect high containment sump level alarm	0.002	10.0	T20-23 #6	DYN	1	T20-16 #2a	ZD	0.005327	0.005327	10.0
D	Operators fail to detect spray alarms	0.004	10.0	T20-23 #6	DYN	1	T20-16 #2a	ZD	0.010654	0.010654	10.0
E	Operators fail to detect rad alarm inside containment	0.008	10.0	T20-23 #6	DYN	1	T20-16 #2a	ZD	0.021308	0.021308	10.0
F	Operators fail to detect CF tank level	0.016	10.0	T20-23 #6	DYN	1	T20-16 #2a	MD	0.042616	0.179385	5.0

Cutset Analysis LD2-CFT				
Cutset			Frequency	Total Mean
Spray alarms	Rad alarm	CF tank alarm	4.1E-005	
Cont. sump	Rad alarm	CF tank alarm	1.9E-005	
CF7A alarm	Rad alarm	CF tank alarm	1.1E-005	
Cont. sump	Spray alarms	CF tank alarm	9.8E-006	
CF7A alarm	Spray alarms	CF tank alarm	5.9E-006	
CF7A alarm	Cont. sump	CF tank alarm	2.7E-006	
Cont. sump	Spray alarms	Rad alarm	1.2E-006	
CFT hi-press	Rad alarm	CF tank alarm	1.1E-006	
CF7A alarm	Spray alarms	Rad alarm	6.9E-007	
CFT hi-press	Spray alarms	CF tank alarm	5.9E-007	
CF7A alarm	Cont. sump	Rad alarm	3.1E-007	
CFT hi-press	Cont. sump	Rad alarm	2.7E-007	
CF7A alarm	Cont. sump	Spray alarms	1.7E-007	
CFT hi-press	CF7A alarm	CF tank level	1.6E-007	
CFT hi-press	Spray alarms	Rad alarm	6.9E-008	
CFT hi-press	Cont. sump	Rad alarm	3.2E-008	
CFT hi-press	CF7A alarm	Rad alarm	1.9E-008	
CFT hi-press	Cont. sump	Spray alarms	1.6E-008	
CFT hi-press	CF7A alarm	Spray alarms	9.9E-009	
CFT hi-press	CF7A alarm	Cont. sump	4.5E-009	
				1.0E-004

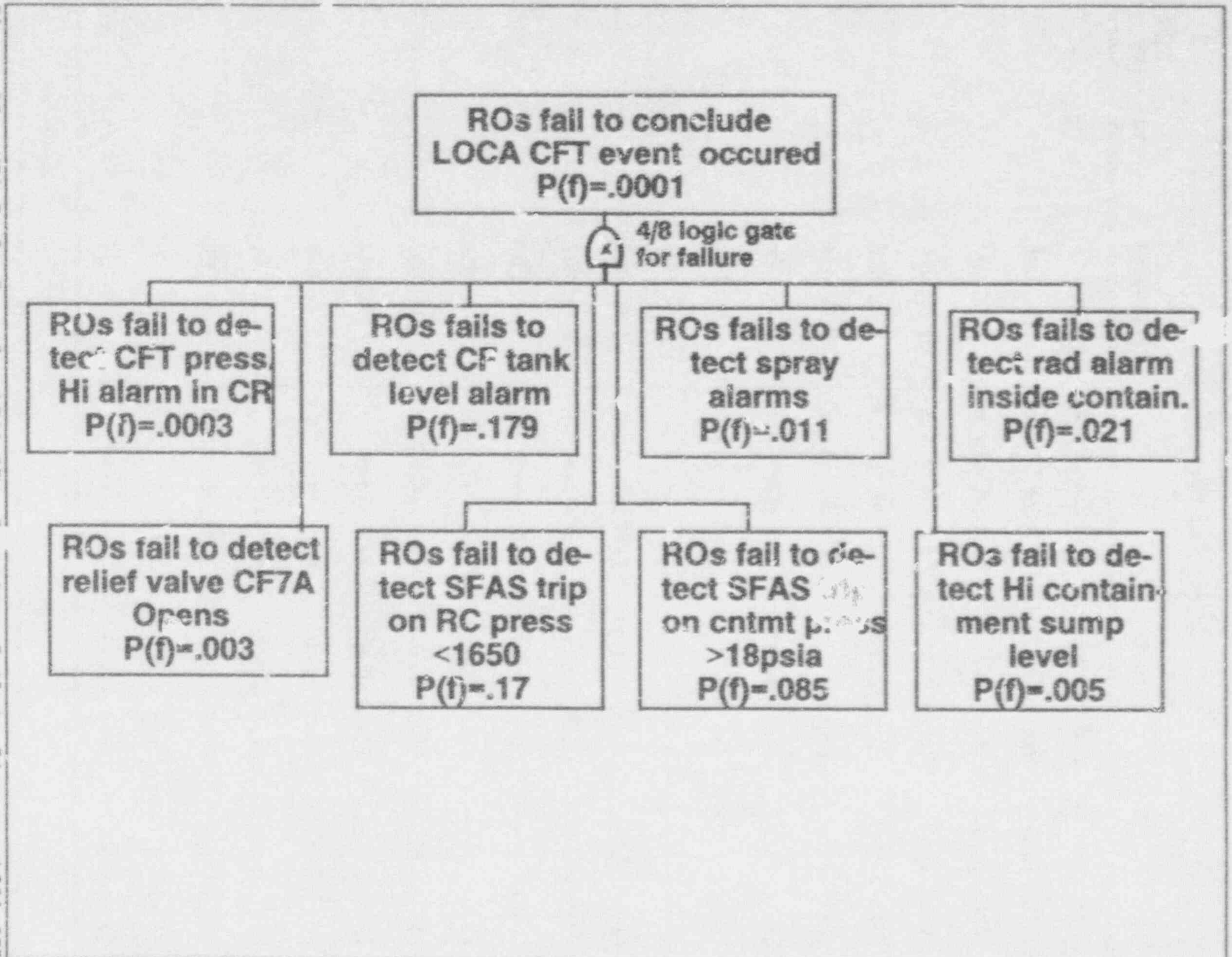


Figure 37: HRA Fault Tree for LDA2-CFT, Operators Fail to Diagnose LOCA CFT (After Rupture)

Table E34: HEPS for LDA2-CFT Operators Fail to Diagnose LOCA CFT (after rupture)

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	ROs fail to detect high containment sump level	0.002	10.0	T20-23 #8	DYN	1	T20-16 #2a	ZD	0.005327	0.005327	10.0
B	ROs fail to detect spray alarms	0.004	10.0	T20-23 #8	DYN	1	T20-16 #2a	ZD	0.010654	0.010654	10.0
C	ROs fail to detect rad alarm inside containment	0.008	10.0	T20-23 #8	DYN	1	T20-16 #2a	ZD	0.021308	0.021308	10.0
D	ROs fail to detect CFT tank level	0.016	10.0	T20-23 #8	DYN	1	T20-16 #2a	MD	0.042616	0.179385	5.9
E	ROs fail to detect CFT pressure high alarm in control room	0.0001	10.0	T20-23 #8	DYN	1	T20-16 #2a	ZD	0.000266	0.000266	10.0
F	ROs fail to detect relief valve CF7A opens	0.001	10.0	T20-23 #8	DYN	1	T20-16 #2a	ZD	0.002663	0.002663	10.0
G	ROs fail to detect SFAS trip on containment pressure >18.4 psia	0.032	10.0	T20-23 #8	DYN	1	T20-16 #2a	ZD	0.085232	0.085232	10.0
H	ROs fail to detect SFAS trip on RC pressure <1650	0.064	10.0	T20-23 #8	DYN	1	T20-16 #2a	ZD	0.170465	0.170465	10.0

Operators fail to close
CF1A(B)
P(f)=.149*
Table 20-7 #2

*Represents the actions only, cogniton related to "where one isolates" is represented as part of the failure rate for diagnosis. Values modified for stress (T20-16 #4) and moderate dependence with preceeding actions.

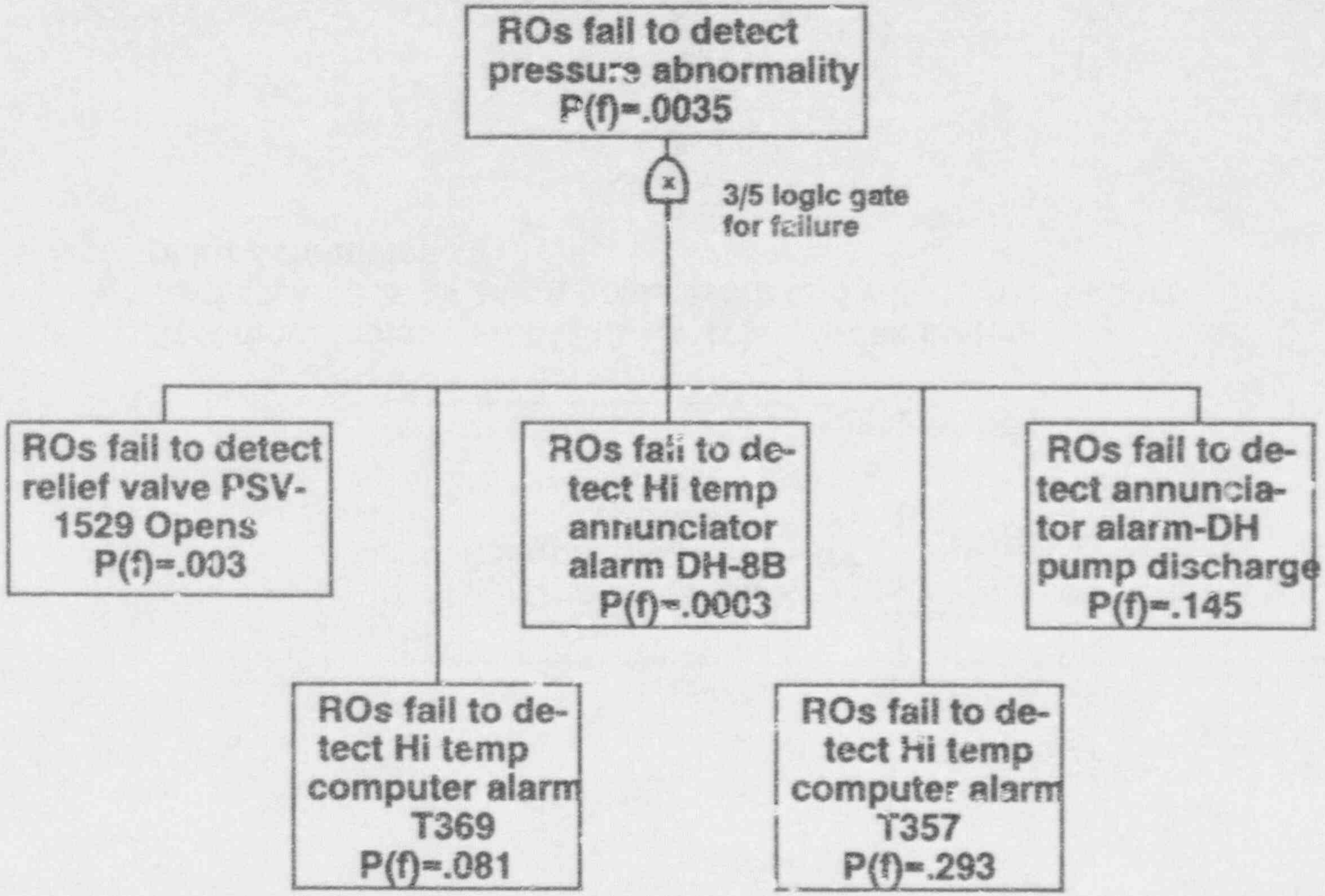
Figure 38: HRA Fault Tree for LI2-CFT, Operators Fail to Isolate

Operators fail to mitigate (i.e., sends the release)
P(f)=1.0*

*Not modeled as part of this exercise, break inside containment is a design basis event and will be handled by plant automatics.

Figure 39: HRA Fault Tree for LMI-CFT, Operator Fails to Mitigate

Figure 40: HRA Fault Tree for LD2-LP, Operators Fail to Detect Abnormality



*Represents assumption that detection of any ? indications is sufficient for detection of the event signature. Table E35 presents combinations of potential failure and their associated probability; value generated using IRRAS 4.0 (1992 version).

Table E35: HEPS for LD2-LP Operators Fail to Detect Pressure Abnormality

	Human Action / Error	Basic Median HEP	Error Factor	Source/ THERP Table #	Step-by-Step or Dynamic	Modifier for PSFs	Modifier Source	THERP Dependency	Basic Mean HEP	Nominal Mean HEP	Error Factor
A	Operators fail to detect relief valve PSV-1529 opens	0.001	10.0	T20-23 #2	DYN	1	T20-16 #2a	ZD	0.002663	0.902663	10.0
B	Operators fail to detect high temp computer alarm T369	0.05	5.0	T20-25 #1	DYN	1	T20-16 #2a	ZE	0.080692	0.080692	5.0
C	Operaters fail to detect high temp annunciator alarm DH-8B	0.0001	10.0	T20-23 #2	DYN	[illegible]	T20-16 #2a	ZD	0.000266	0.000266	10.0
D	Operators fail to detect high temp computer alarm T357	0.05	5.0	T20-25 #1	DYN	1	T20-16 #2a	HD	0.080692	0.540346	5.0
										* 0.292763	2.5
E	Operators fail to detect annunciator alarm-DH pump discharge	0.001	10.0	T20-23 #2	DYN	1	T20-16 #2a	MD	0.002663	0.14514	5.0

Operators fail to conclude ISLOCA event from past rupture information
P(f)=.01 (10)*

*Value determined by engineering judgement.

Figure 41: HRA Fault Tree for LDA2-LP, Operators Fail to Diagnose ISLOCA (after rupture)

Operators fail to close
DH-1A(B)
P(f)=.148 (5)*
Table 20-12 #3

*Represents the actions only, cognition related to "where one isolates" is represented as part of the failure rate for diagnosis. Values modified for extreme high stress (T20-16 #6) and moderate dependence.

Figure 42: HRA Fault Tree for LI2-LP, Operators Fail to Isolate

Review of HEPs and HRA Failure Rates

HRA was used to model the predominant human errors for each scenario in this ISLOCA PRA. As discussed in Section 2.5, HRA is a methodological tool that involves the quantitative analysis, prediction, and evaluation of work-oriented human performance. The B & W ISLOCA HRA diagnosed those factors within the plant's systems that could lead to less than optimal human performance in the initiation, detection, diagnosis, and mitigation of ISLOCA scenarios. HRA was used as a diagnostic tool to isolate the error rate anticipated for individual tasks and to determine where errors were likely to be most frequent.

Within the context of modeling in the HRA, performance shaping factor information is accounted for in both fault tree and HRA event tree estimates. Many of the fault and event trees have been annotated to provide modeling assumptions regarding the degree of task dependence, amount of stress present, communication requirements between ROs and either EOs or I&C technicians, use of anti-contamination clothing, and perceptual demands such as having to detect computer alarms or notice differences in makeup and letdown flow indication.

Inspection of the data reveals that failure rate probabilities are highest for mitigation, isolation, and errors of commission such as inadvertent valve lineup after test, or faulty decisions such as early entry into DHR cooldown. Diagnoses errors range on the order of 5.9E-1 to 6.0E-3 and, in many cases, reflect the large amount of time available for the crew to reach an opinion on the event. Rates for isolation and mitigation were observed to range from 2.0E-3 tO 1.5E-1, respectively, and reflect the lack of resources available to crews. These resources, which if present, would have decreased the failure rate estimates, include an ISLOCA procedure, training on ISLOCA, instrumentation, and a procedure for computer alarm acknowledgement. Without these items, crews could be forced to operate in a knowledge-based realm during an ISLOCA.

Table E36 presents latent errors identified during conduct of the HRA. Each of the errors is preceded by the event sequence number and is followed by the nominal (detailed) HEP value. Description of the error and the sequence is presented in Appendix D.

Table E36: Latent Errors

Sequence	Event Description	Mean Hep
HV1-MU	HP vent line left open	0.0013
HV1-HP	HP vent line left open	0.0013
DM1-SU	MCVs DH 11 & 12 left open	0.0002
DM2-SU	Local valves DH 21 & 23 left open	0.0002

Only one error of commission was identified as an initiator (i.e, DM1-SD): operators open DH11/12 too soon in the shut down cycle. Latent errors involving vent line configuration shown in Table 4.6 can be of either the omission or commission type. The low failure rate for DM2-SU reflects the double verification for these valves as called out by both the SP--0130 and OP--00008 procedures.

Detailed Breakdown of Human Error Actions

The following table represents the distribution of errors modeled in support of ISLOCA evaluation at a B&W plant. The tabled values include all the errors modeled in the supporting fault trees and HRA event trees.

Table E37: Distribution of Errors from Supporting Analyses

	Omission	Simple Commission	Decision-based
Frequency	100	17	13
Percent (%)	77%	13%	10%

As these data indicate, the majority of HEP data used in the present analysis fall into the omission category. This is in keeping with contemporary PRA. What is unique about the ISLOCA HRA for B&W is that some 20% of the total errors modeled are from commission and complex commission decision based sources. Although caution should be taken when extrapolating from one plant's data, these results *do* indicate that PRAs may under-represent human contribution to systems failure by some 10 to 20%.

Decision-based Errors

The rates for decision-based errors presented in Table 4.6 were derived using THERP and engineering judgement techniques. *While these failure rates apply to those decision-based errors identified and quantified in the B & W ISLOCA analysis, they are not limited to instances where the action is the top-level action in an event sequence.* To learn more about where a particular decision-based failure fits within an action flow, the sequence identifier, task description, failure rate and error factor (EF) have been presented in the preceding tables and are explained in detail in Appendix D.

TABLE E38: Decision-based Errors (either Task or Subtask values)

Identifier	Description	HEP
HDA2-MU, HP*	ROs fail to conclude ISLOCA(from prior tasks)	.006
HI2-MU, HP	ROs fail to isolate HP2A, undo what was just done	.002
DM1-SD	ROs decide on early entry into DHR(jumpering OK)	.00066
DDA1-SD*	ROs fail to conclude ISLOCA from event signature	.006
DI2-SD*	Crew fails to send I&C to remove jumpers (total HEP=9.0 x 10E-5)	.008
DM2-SU	ROs fail to close DH21 & 23	.0002
DA1-SU-A	ROs fail to recognize ISLOCA from event signature (local valves open; relief valve opens)	.52
DA1-SU-B	ROs fail to recognize ISLOCA from event signature (local valves open; relief valve fails closed)	.59#
DA1-SU-C	ROs fail to recognize ISLOCA from event signature (MOVs open; relief valve opens)	.29#
DA1-SU-D	ROs fail to recognize ISLOCA from event signature (MOVs open; relief valve fails closed)	.43#
LDA2-CFT*	ROs fail to conclude ISLOCA-core after rupture	.0001
LDA2-LP	ROs fail to conclude ISLOCA from past rupture information	.01

NOTE. * Indicates subtask values;
# Indicates engineering judgement used to estimate HEP.

Sensitivity Analysis

A sens' 'vity analysis was conducted using steps #7 and #8 from the IEEE P1082 standard, i.e., update plant model and review results. The failure paths and HEP's from the detailed analysis were reviewed to determine if modifications to the human-machine system would result in significant gains in operator performance and a corresponding reduction in the nominal HEP. This re-analysis was limited to actions which would prevent initiation of an ISLOCA sequence. The following actions were identified as ways the operators' performance could be easily optimized:

1. Procedures for startup, shutdown, ·· quarterly stroke test being upgraded to reflect the appropriate operator actions, cautions, notes, warnings, or checklists. These revised procedures would adopt current industry standards for being symptom-based and would be used to extensively train plant personnel to recognize the potential for ISLOCA.
2. Instrumentation - hardware changes, such as including th presence of a valve status board in the control room, would tend ower operator error. However, a simpler and more efficacious approach would be to train operators to recognize direct and indirect indications of an ISLOCA.
3. Training was improved by training control room and EO personnel in a formal ISLOCA procedure and by having training and procedures for the handling of computerized alarms on the control room CRT.
4. Recovery factors are included by having all tasks covered by procedures and having an independent second operator (s' ft supervisor, I&C or maintenance foreman etc.) wh must s off on tasks performed.

The actions which were selected and the resulting HEPs after optimization and shown in Table E39.

In the first sequence (HPI scenario involving quarterly stroke test for 2A, MU&P flow), the tasks comprising the errors HD2-MU, HDA2-MU, and HI2-MU were changed to reflect optimized procedures. Specifically the changes made were to use a well written, symptom-based, ISLOCA emergency operating procedure (short list with less than 10 items) and, within the procedure, to provide for an independent verification with required sign-offs by a second person. No other changes were necessary to achieve the risk reduction.

Table E39: Optimized HEPs from the Sensitivity Analysis

Event Tree Element	PRA HEP	Opt. HEP
1. HPI Scenario Involving Quarterly Stroke Test for 2A, MU&P Flow		
HD2-MU operators fail to detect ISLOCA	.0028	<0.0001
HDA2-MU operators fail to diagnose ISLOCA	.006	<0.0001
HI2-MU operators fail to isolate ISLOCA	.002	<0.0001
2. Shut-down Scenario Involving Premature Opening of DH11 & DH12		
Event Tree Element	**PRA HEP**	**Opt. HEP**
DM1-SD operators open DH11 & DH12 too soon	.00066	<0.0001
DDA1-SD operators fail to diagnose ISLOCA	.006	<0.0001
DI2-SD operators fail to isolate ISLOCA	.008	<0.0001
3. Low Pressure Injection System ISLOCA Scenario		
Event Tree Element	**PRA HEP**	**Opt. HEP**
LI2-CFT operators fail to isolate ISLOCA	.149	<0.0001
LDA2-LP operators fail to diagnose ISLOCA	.01	<0.0001
LI2-LP operators fail to isolate ISLOCA	.148	<0.0001

In the third sequence (Shut/down scenario involving premature opening of DH11 &12) the error DM1-SD, DDA1-SD, and DI2-SD were optimized. In this case a number of improvements were made. For DM1-SD these included: applying proper administrative controls which would disallow the practice of defeating interlocks and the jumpering of DH12; and having a well written procedure with clearly stated RHR system limits and proper cautions about the consequences of early entry into decay heat removal. These actions would eliminate the possibility that an operator would believe it proper to enter decay heat removal prematurely by opening DH-11 & DH-12 in this shutdown scenario. For DDA1-SD and DI2-SD, in the event that an ISLOCA event occurred before RHR temperature and pressure limits were acceptable, the HEP estimates were reduced based on the following assumptions: that a well written, symptom-based ISLOCA abnormal operating procedure was availabl o guide operators in diagnosing ISLOCA and in taking effective actions to isolate after an ISLOCA; that operators had received extensive training on how to recognize direct and secondary indications of ISLOCA during the various stages of plant operations.

In the last sequence (Low pressure injection system ISLOCA Scenario) the errors LI2-CFT, LDA2-LP, and LI2-LP were optimized. These HEPs were reduced due to the fact that all personnel, e.g., the control room operators, equipment operators, and maintenance personnel would now be trained on the potential for ISLOCA and that there would be a well written, symptom-based ISLOCA emergency operating procedure with sign-offs for a second operator's independent verification.

Most of the modifications suggested by the sensitivity analysis are believed to be fairly simple, i.e., the use of procedures with checklists, verification of operator actions, specific training recognizing ISLOCA scenarios, and the inclusion of a valve status board in the control room. The sensitivity analysis results point out the need to specifically address ISLOCA as a possible plant scenario, and underscore the need for plant personnel to be made aware of ISLOCA indications through appropriate modifications to procedures and ISLOCA specific training.

Conclusions

The current analysis indicates that human errors, particularly, errors of commission, are an important contributor to the core damage frequency for ISLOCA sequences. However, it is premature at the present time t say whether, in Reason's terminology[11], "active" errors such as the decision to prematurely enter DHR, or the human contribution to risk due from "latent" errors will be important at other plants. In the present case, both of these types of errors of commission played a significant role in assessing the plant's susceptibility to ISLOCA. If training for ISLOCA had been available at the plant and if personnel had good "ISLOCA" procedures, then the probability for ISLOCA would be reduced. Proceduralizing crew response to computer alarms and providing additional valve status indication would also serve to reduce risk.

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Appendix F

Thermal Hydraulic Calculations

G. E. Gruen

CONTENTS

FIGURES

APPENDIX F
THERMAL HYDRAULIC CALCULATIONS

F.1 Introduction

The appendix presents selected thermal hydraulic results from studies involving the interfacing systems of the B&W reference plant. The thermal hydraulic parameters of the interfacing system loss-of-coolant-accidents were calculated. The parameters that were calculated are the pressure and flow histories within the decay heat removal and low-pressure injection systems piping. These calculations were based on failures of specified valves.

The ISLOCA thermal hydraulic analyses were performed using the RELAP5/MOD2.5 computer code.[F-1] The thermal hydraulic analysis consisted of three complex models describing the; a.) Decay Heat Removal (DHR) piping, b.) Low-Pressure Injection (LPI) piping and, c.) the Make Up and Purification Interface (MU&P) piping. These three complex models were supplemented by three relatively simple models. The input models for RELAP5 were built within a Lotus 1-2-3 spreadsheet. These models were developed using data from in-service inspection isometric drawings. The spreadsheets and RELAP input and output listings are maintained in an Energy and Systems Technology (EAST) group calculation file.

F.2 DHR Analysis

The ISLOCA thermal hydraulic analysis of the DHR system involved modeling 643 ft of piping, five relief valves, and all the elbows, expansions, and contractions. Pipe volume lengths of 4 ft were used for the DHR analysis as well as for the other two detailed analyses. The pipe wall friction factor was 1.5E-4 for all the analyses.[F-2] The friction losses through valves, elbows, and orifices were based on the Crane technical paper.[F-3]

The hardware schematic is shown in Figure F-1 with RELAP pipe component numbers and the relief valves described. The pipe components are keyed to the reference B&W plant component piping specifications. The RELAP5 pipe components 70, 100, 110, 140, 470, 500, 510, and 540 are all dead ended. The Pipe components 30, 90, 130, 490, and 530 connect to the relief valves specified. The reactor vessel is component 200. It was modeled as a constant pressure and temperature source for the piping. (Note: It should be noted that the constant pressure and temperature reactor vessel assumption results in an overly conservative estimate of the time averaged mass flow rate out of the break. This assumption does allow for an adequate method to determine if the interfacing systems will fail during the <u>early</u> rapid pressurization part of the ISLOCA event.)

The ISLOCA thermal hydraulic analysis was initiated by an assumed failure of valve DH-12. The RELAP5 model constructed for the failure of DH-12 opened the valve linearly over 10 s to initiate the transient. The relief valves were also opened linearly. The time scale for the relief valves however was 0.10 seconds. The RELAP5 set points of these relief valves open them at the following pressures:

Valve No.	Setpoint (psia)
PSV 4849	320
PSV 1508	450
PSV 1509	450
PSV 1529	75
PSV 1550	75

The ISLOCA transient thermal hydraulic calculations were done at the following reactor vessel conditions:

Pressure (psia)	Temperature (°F)
2200	600.0
1500	546.2
1000	494.6
500	417.0

The reactor coolant temperatures associated with reactor vessel pressures of 500, 1000, and 1500 psia were predefined. The coolant temperatures were set to the corresponding saturation temperature minus 50°F. The calculated results provided by the RELAP software are shown in Figures F-2 through F-13. Three sets of figures for each reactor vessel pressure have been provided.

The DHR data were correlated with component pressure as a function of reactor vessel pressure. This nearly linear relationship is shown in Figure F-14 for components DH-12, DH-4849, FE-4908, AND DH-2734.

F.3 LPI Analysis

The analysis of the LPI system piping involved the same model that was used for DHR piping. The same 643 ft of piping, 5 relief valves, and all the elbows, expansions, and contractions were used. The origin of high pressure and temperature (the reactor vessel) was connected at valve DH-76 instead of DH-12. Valve DH-76 was used to initiate the transient with a 10 s opening time.

The hardware schematic for this sequence is shown in Figure F-15. This figure describes the RELAP5 pipe component numbers and the relief valves. Note that the reactor vessel (RELAP5 component 200) is shown to connect to valve DH-76. The pipe components have been keyed to the referenced B&W plant's component piping specifications. The RELAP5 pipe components 10, 70, 100, 110, 470, 500, 510, and 540 are all dead ended. The pipe components 30, 90, 130, 490, and 530 all connect to the specified relief valves. The relief valves were opened linearly over 0.1 s. The relief valves used the same setpoints as identified in Section F.2 for the DHR piping.

The flow direction was reversed for the LPI analysis. A check valve prevented reverse mass flow through the DHR system pump. The mass flow was diverted through a 2-in. line by passing the pump. (The 2-in. line is modeled by RELAP5 component 120 in Figure F-15.) The 2-in. line contained a 0.657-in. orifice that severely restricted the reverse mass flow. For that reason, the pressure downstream of the orifice is much lower than upstream. This pressure behavior is shown in Figure F-16.

Figure F-16 also shows the pressure within the piping as a function of node position. The node position is keyed to hardware components. Note the reactor vessel is node 161 in this analysis. These results make it readily apparent that components upstream of the orifice (closer to the reactor vessel) are likely to fail while those components downstream of the orifice are likely to survive.

F.4 Makeup & Purification Interface

Three different RELAP5 input models were used in calculations to determine the behavior of the MU&P system piping during its postulated interfacing LOCA transient.

F.4.1 Simplified Model

A simplified RELAP5 input model was prepared for the MU&P system. As shown in Figure F-17, the reactor vessel was modeled as a tank at 2200 psia and 600°F. The tank was connected to 10 pipe volumes of 3-in. schedule 40 pipe each for a total pipe length of 200 ft.

Figure F-17 shows a diagram of the hardware as modeled with RELAP5. Figures F-18 and F-19 show the pressure distribution at steady state and the exit mass flow rate history. The steady state mass flow rate and mass flux are 281.7 lb/s and 5487 lb/ft^2-s.

F.4.2 Detailed Model Without Orifice

A detailed RELAP5 input model was prepared that included the various pipe lengths, diameters, elbows, and valves, but without an orifice. Results for this model are shown in Figures F-20 through F-23, Figures F-20 and F-21 show the pressure history at the various pipe locations. Figure F-22 shows the exit mass flow rate history, and Figure F-23 shows the steady state pressure as a function of location along the pipe.

F.4.3 Detail Model With Orifice

The detailed RELAP5 input model was modified to include an orifice area of 0.0045 ft^2. The results for this model are shown in Figures F-24 through F-27. Figures F-24 and Figure-25 show the pressure history at the various pipe locations. Figure F-26 shows the exit mass flow rate history and Figure F-27 shows the steady state pressure as a function of location along the pipe.

.5 One-inch Pipe

Pipe sizes that could be involved in an ISLOCA were determined by RELAP5 calculations. These calculations were performed to eliminate pipe sizes below a certain diameter from inclusion in ISLOCA sequences. The results of these pipe calculations indicate that pipe sizes below one inch can be eliminated from ISLOCA considerations.

The one inch pipe diameter calculations are described next. The 1-in. diameter pipe modeled with the RELAP computer code is shown in Figure F-28. The pipe is 50 ft long with four gate valves spaced equally along the pipe. The upstream component represents a reactor vessel at 2200 psia and 550°F and the downstream boundary represents atmospheric conditions. The RELAP5 analysis indicated that volumetric flow rates are about 207 gpm when the gate valves were assumed to have the same flow area as the pipe and 203 gpm when the gate valves were assumed to have 80% of the pipe area. These flow rates are similar to the make up flow to the BWST. It is possible to eliminate pipe sizes less than 1 inch from the ISLOCA analysis, since the failure of a one inch line is not able to drain the BWST.

F.6 LPI - Opening Time and Break Size Effect

A simplified model representing the piping for the LPI system was used for a sensitivity study of break opening time and opening size. A diagram of the model is shown in Figure F-29. Break opening times of 0.1 and 1.0 s and break sizes of 0.05, 0.1, 0.2, and 0.5% were calculated. The plotted results are shown in Figures F-30 and F-31.

F.7 References

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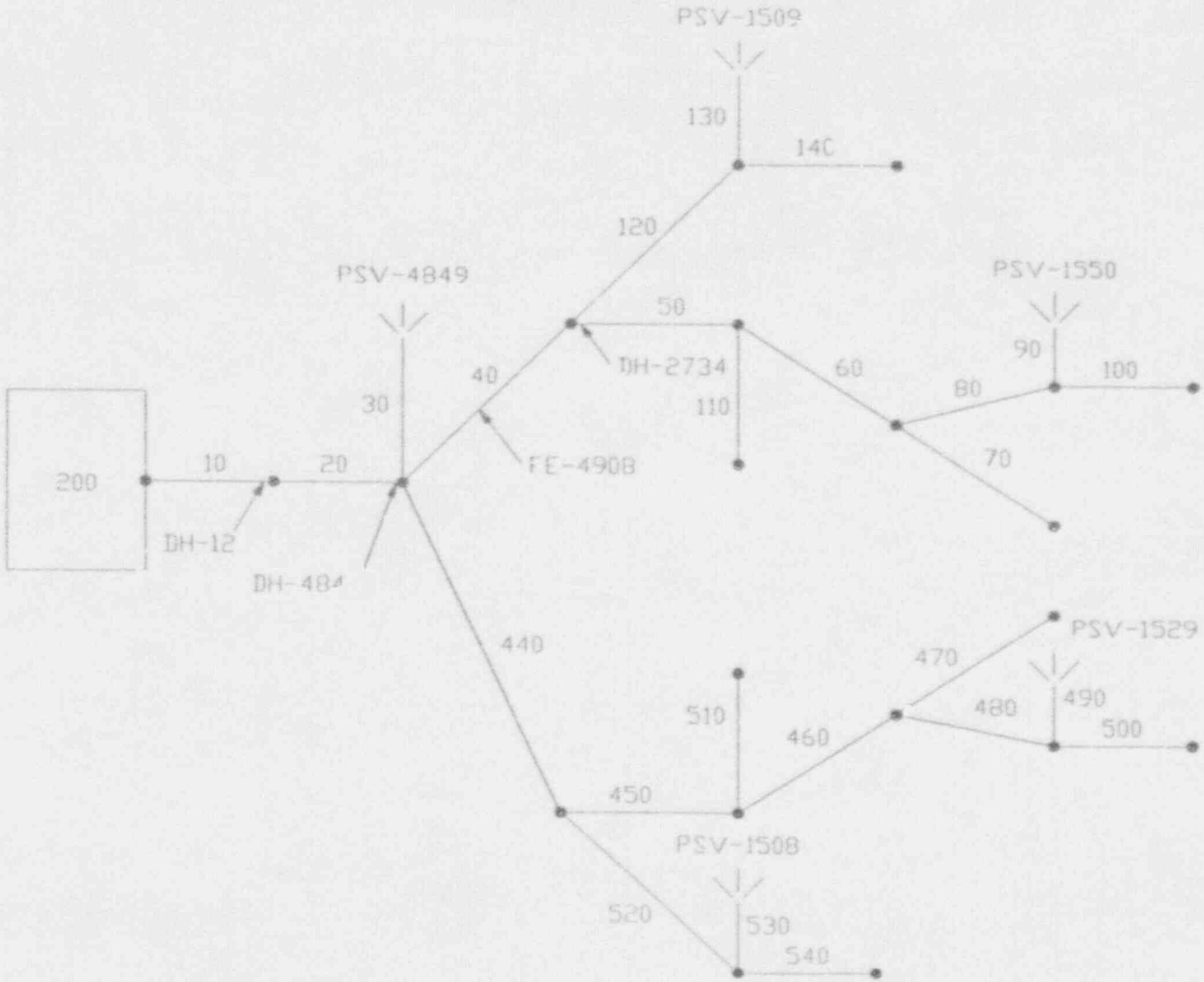


DHR	
RELAP PIPE	DAVIS-BESSE COMPONENT
10	12in-CCA-4
20	12in-CCA-4
	12in-GCB-7
30	4in-GCB-7
40 & 440	12in-GCB-7
	2.5in-GCB-7
50 & 450	18in-GCB-8
60 & 460	18in-HCB-1
70 & 470	18in-HCB-1
80 & 480	18in-HCB-1
90 & 490	18in-HCB-1
100 & 500	18in-HCB-1
	14in-HCB-1
110 & 510	10in-HCB-3
120 & 520	18in-GCB-8
	12in-GCB-8
	10in-GCB-1
	10in-GCB-10
130 & 530	1.5in-10S
140 & 540	10in-GCB-10
	10in-CCB-6
200	Reactor Vessel

Figure F-1 Diagram for RELAP model of DHR system piping.

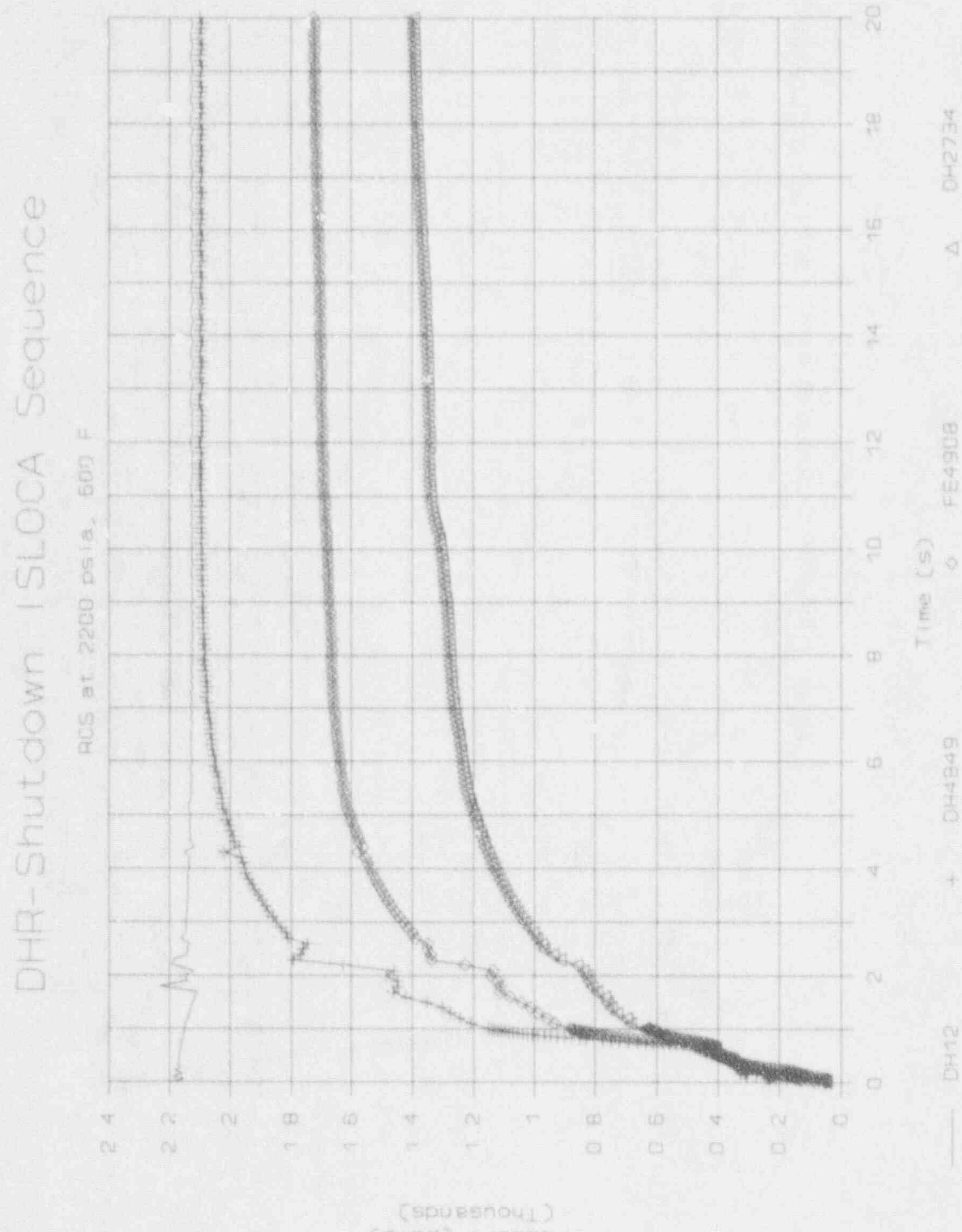


Figure F-2 Valve pressure history in DHR system piping - 2200 psia.

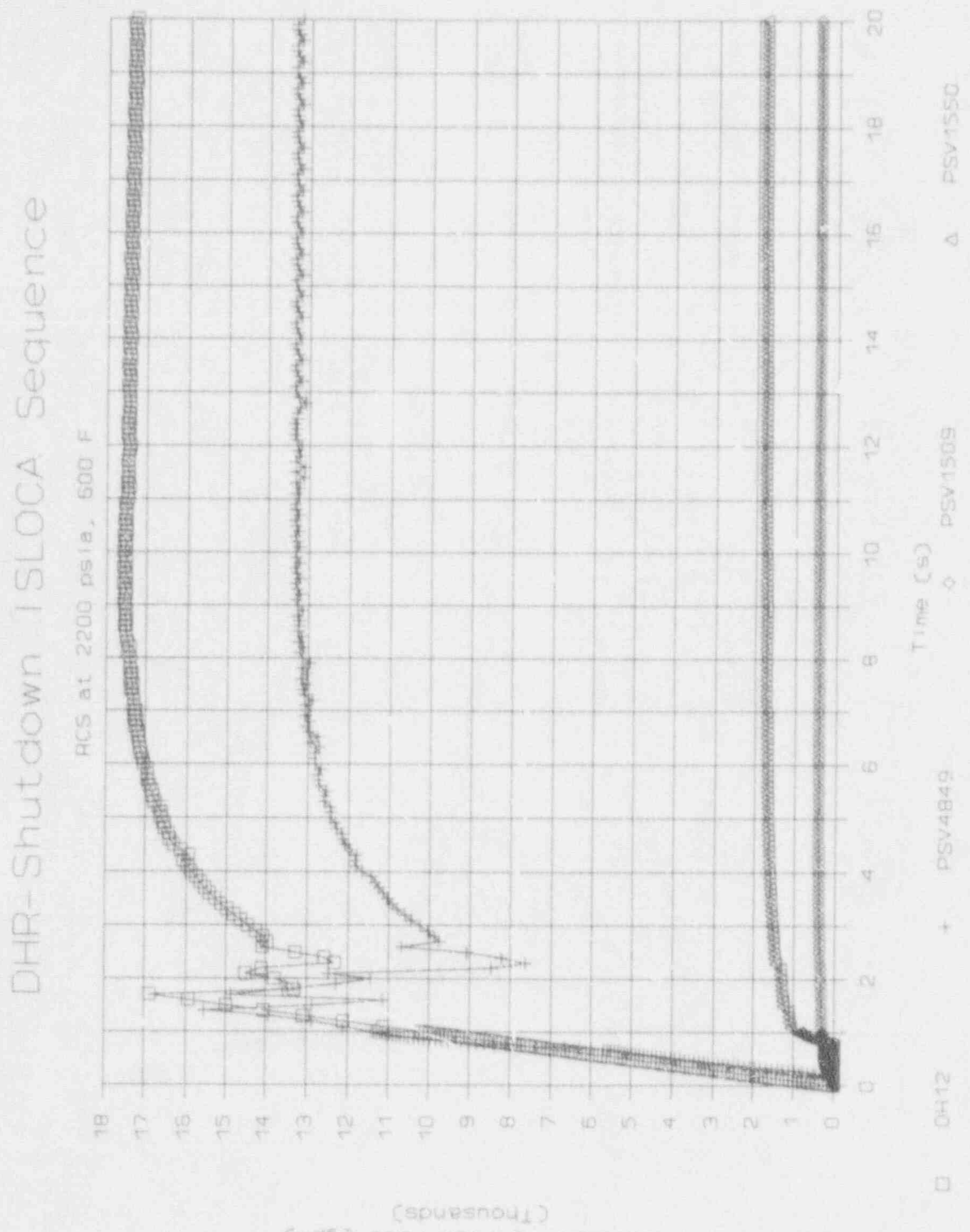


Figure F-3 Relief valve flow from DHR piping - 2200 psia.

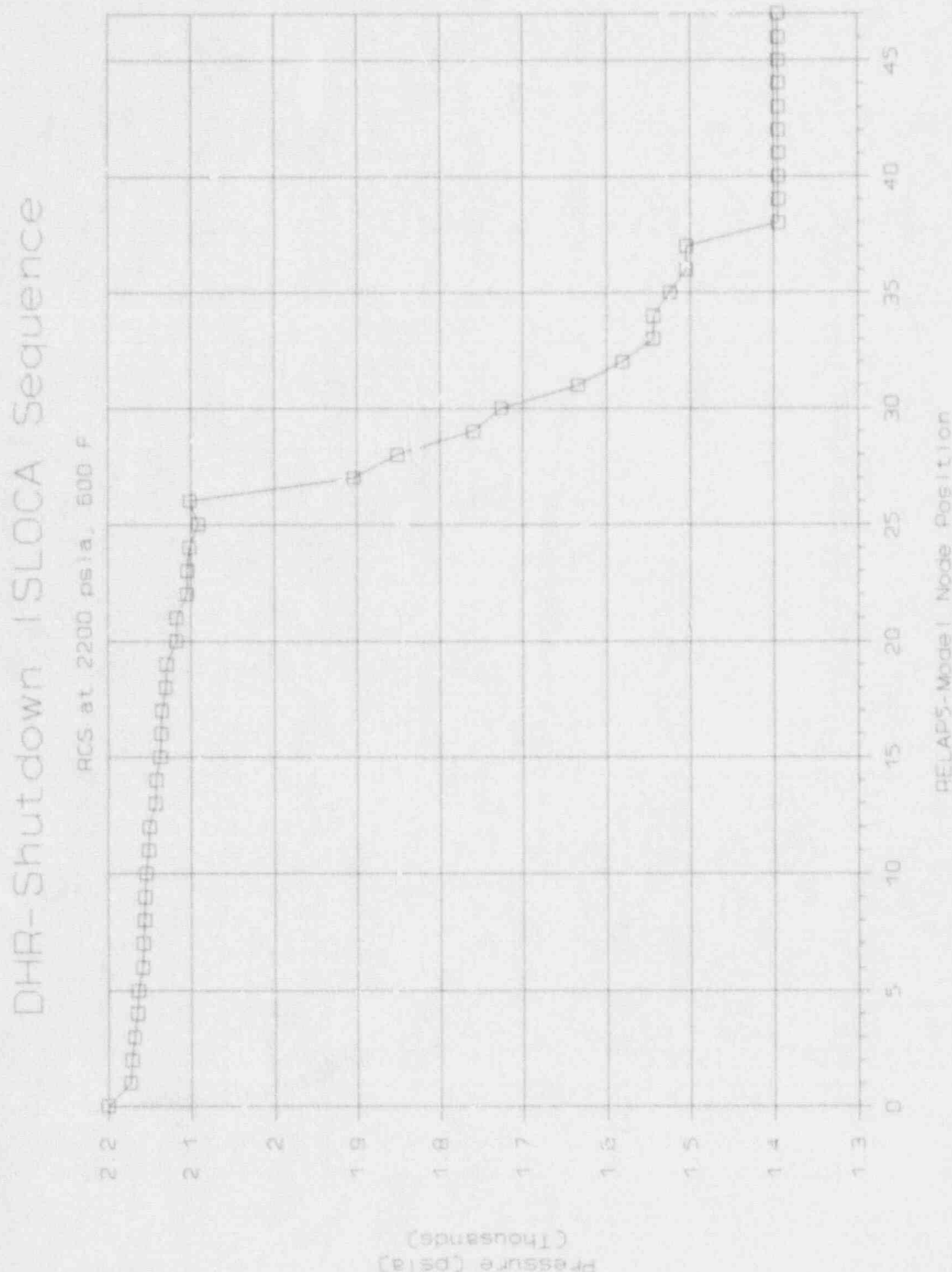


Figure F-4 Steady state pressure in DHR system piping - 2200 psia.

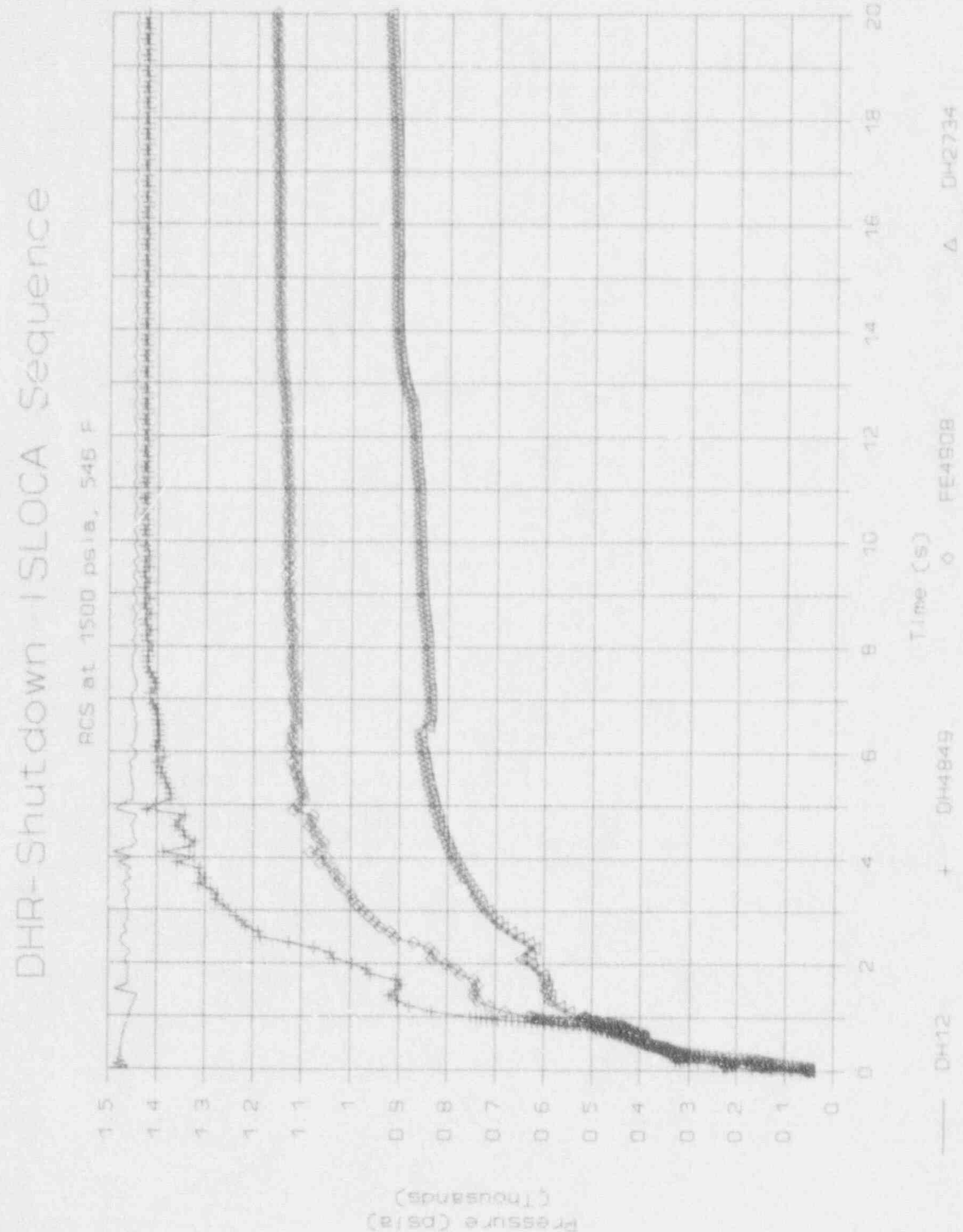


Figure F-5 Valve pressure history in DHR system piping - 1500 psia.

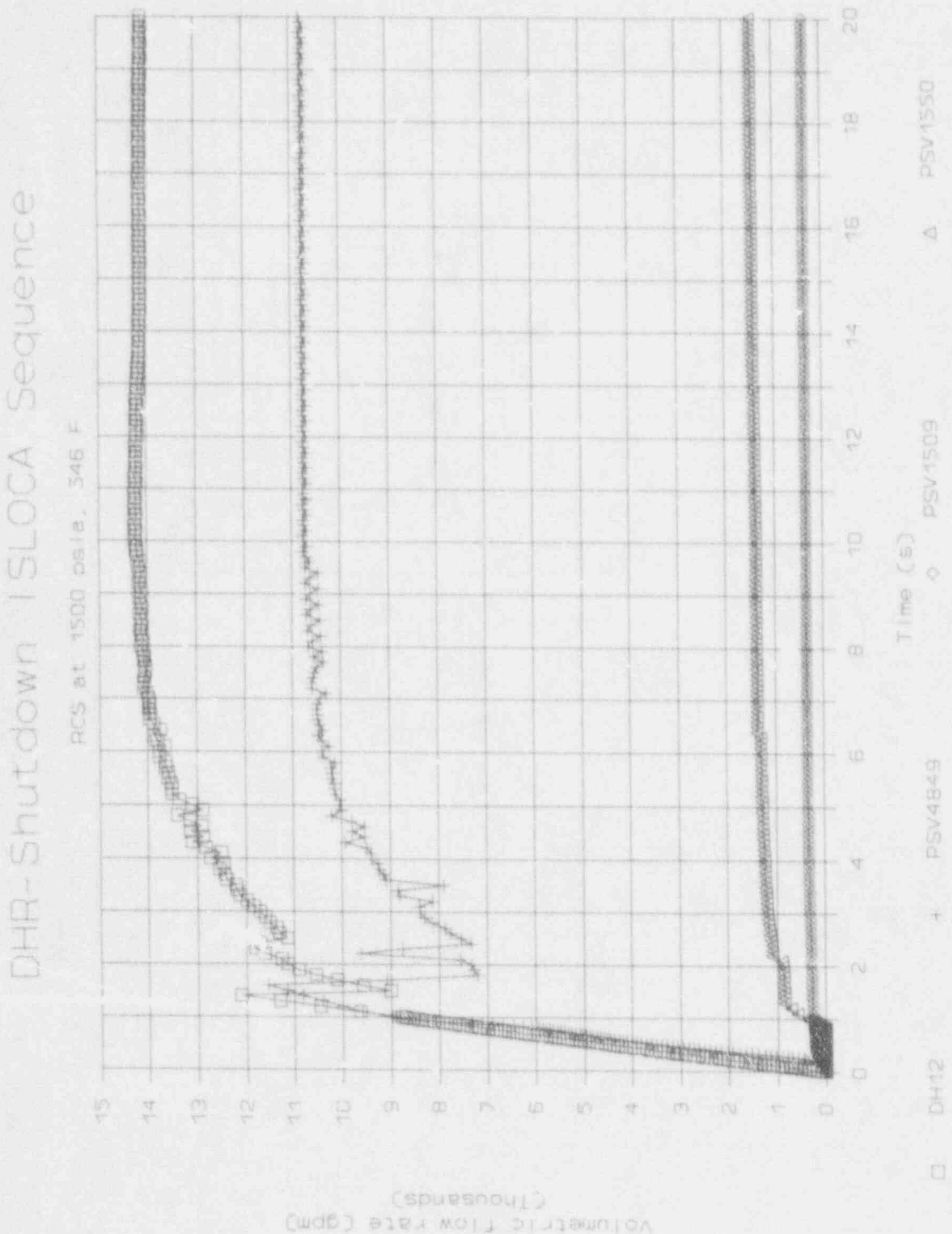


Figure F-6 Relief Valve flow from DHR piping - 1500 psia.

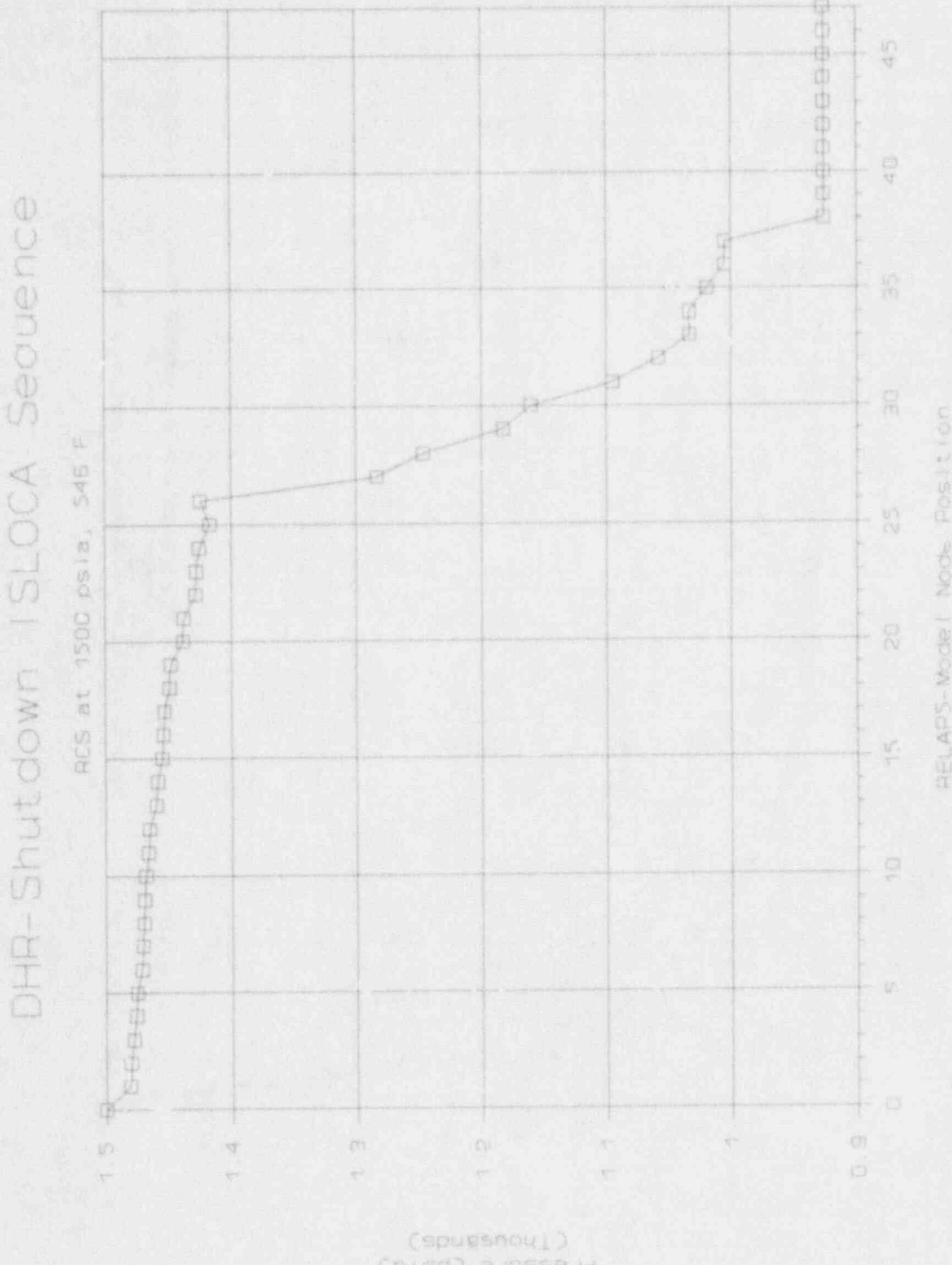


Figure F-7 Steady state pressure in DHR system piping - 1500 psia.

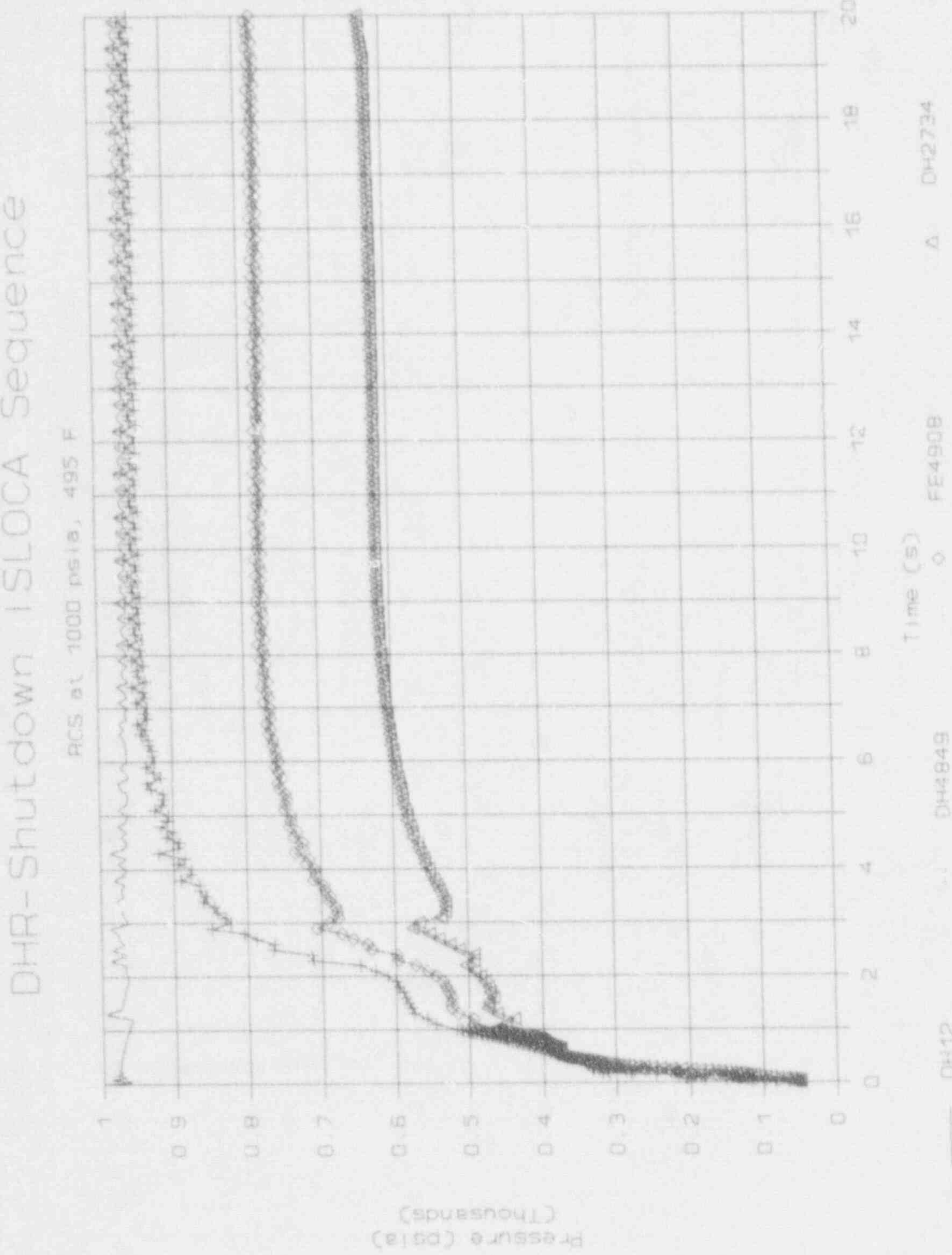


Figure F-8 Valve pressure history in DHR system piping - 1000 psia.

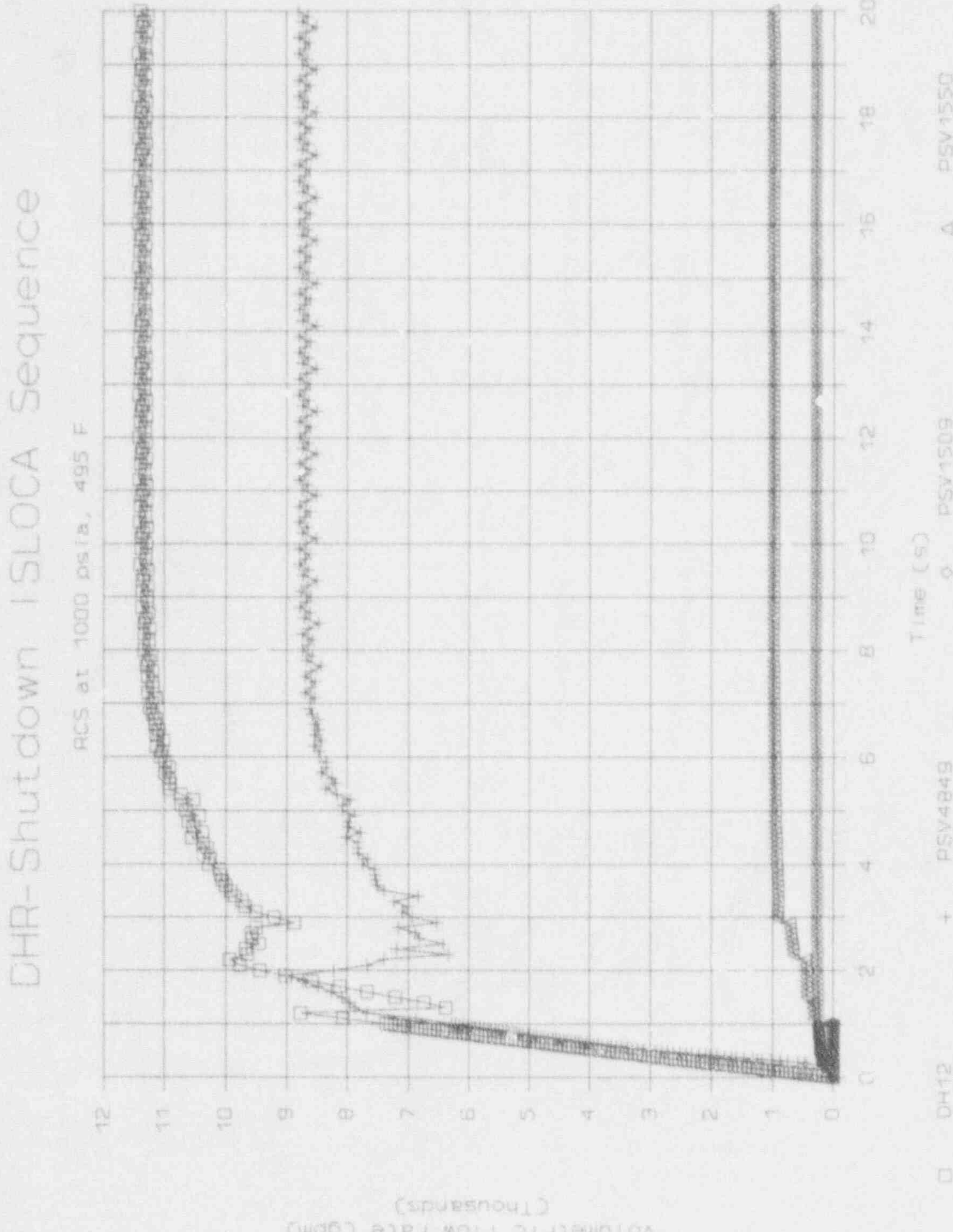


Figure F-9 Relief valve flow from DHR piping - 1000 psia.

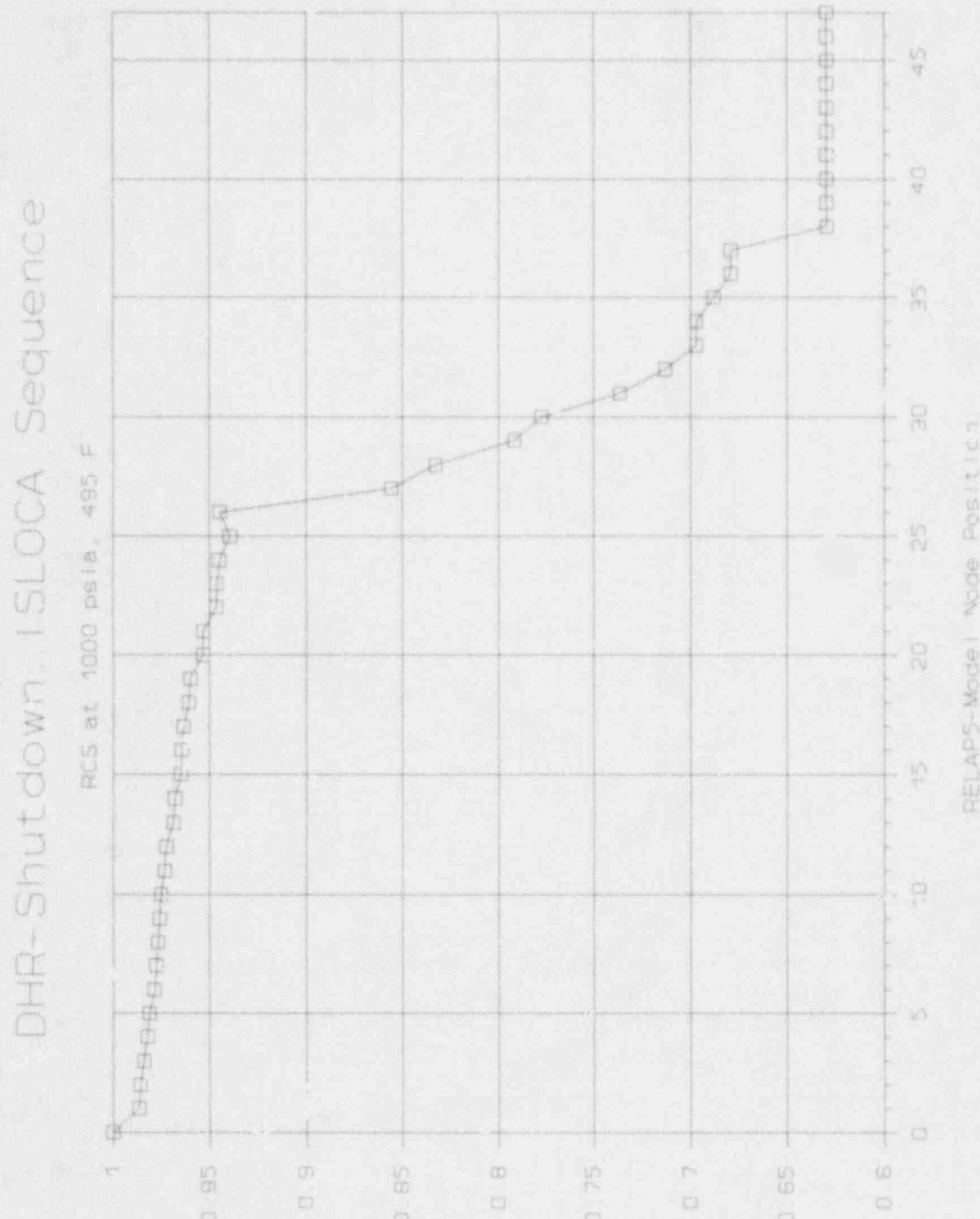


Figure F-10 Steady state pressure in DHR system piping.

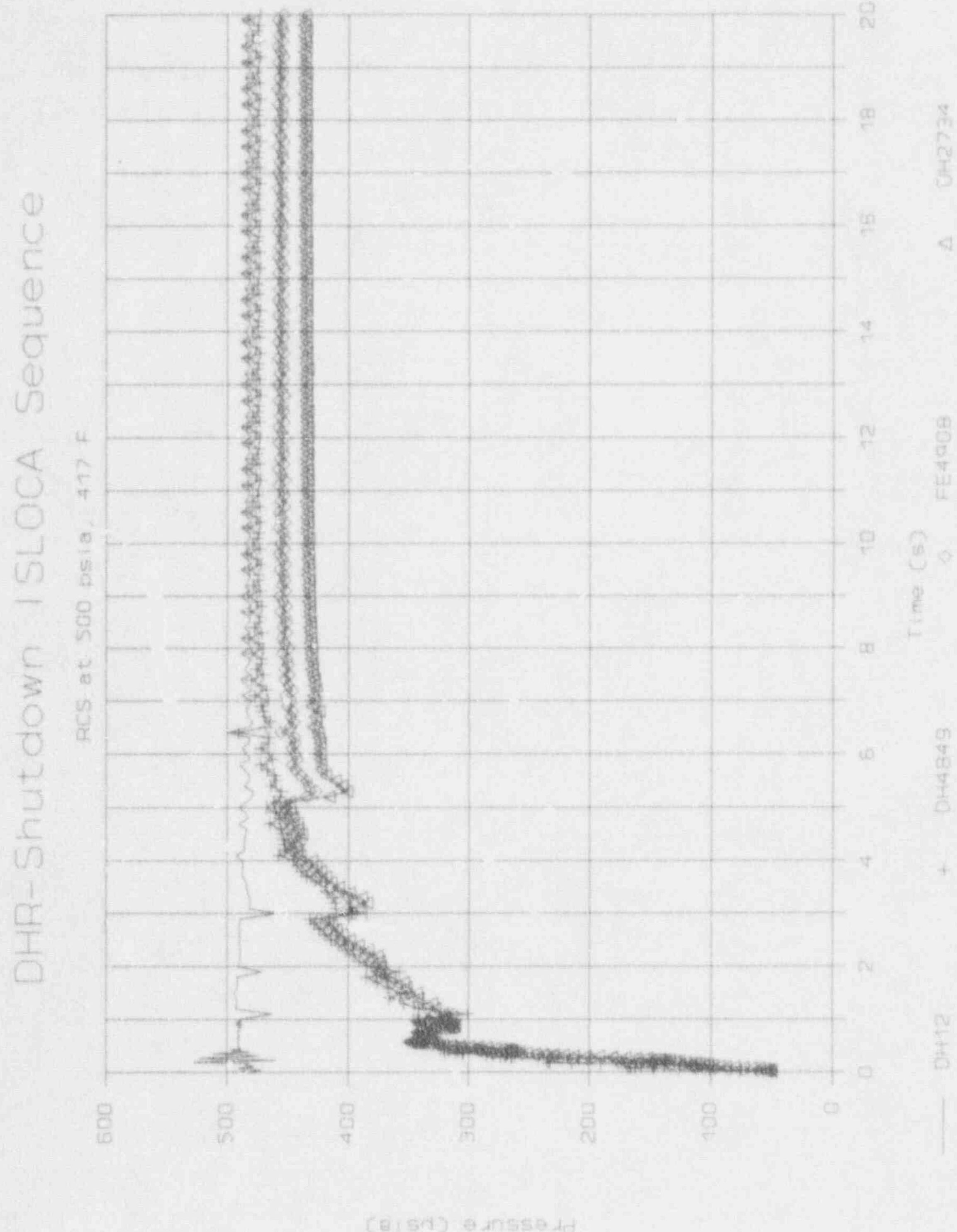


Figure F-11 Valve pressure history in DHR system piping - 500 psia.

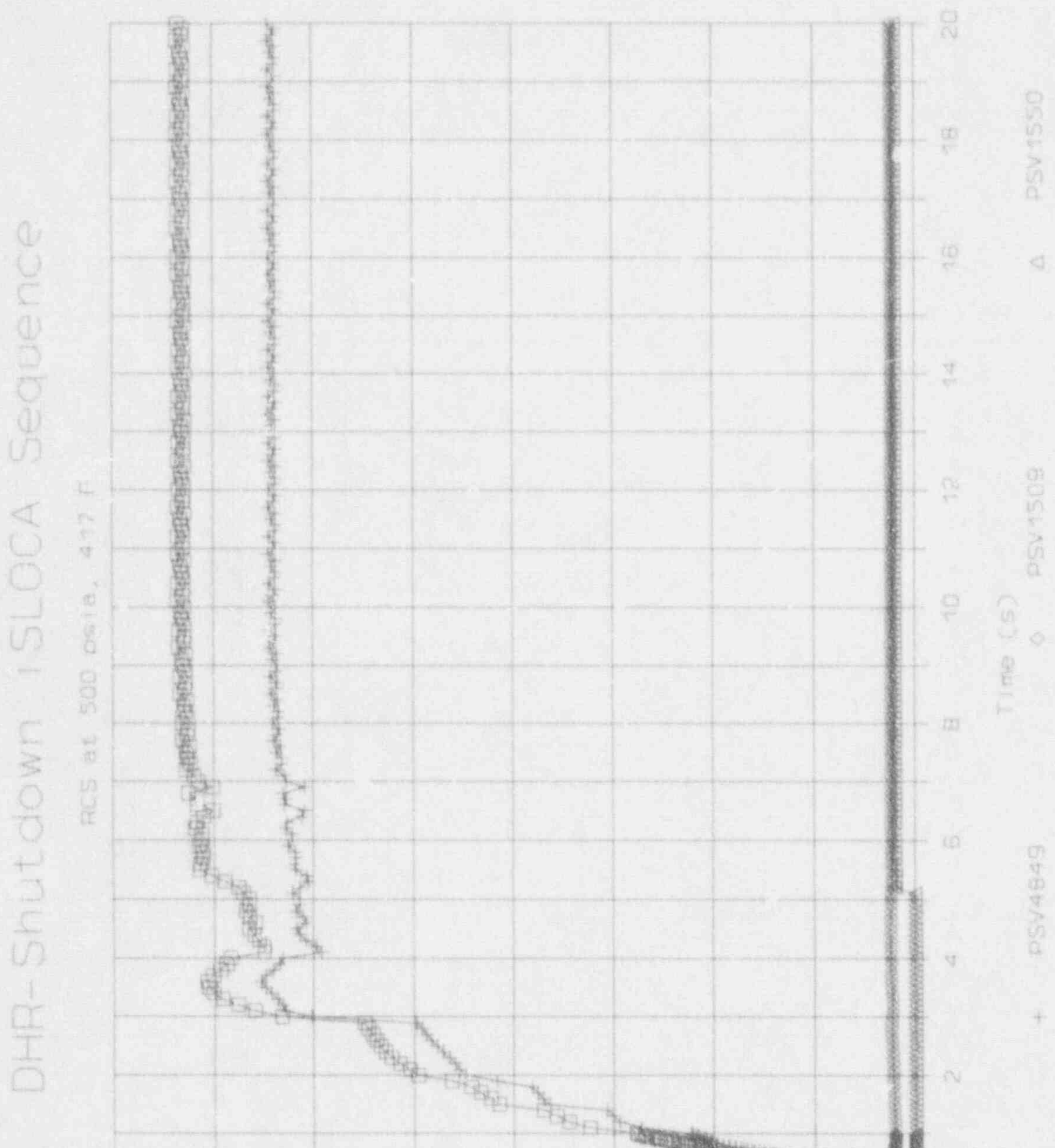


Figure F-12 Relief valve flow from DHR piping - 500 psia.

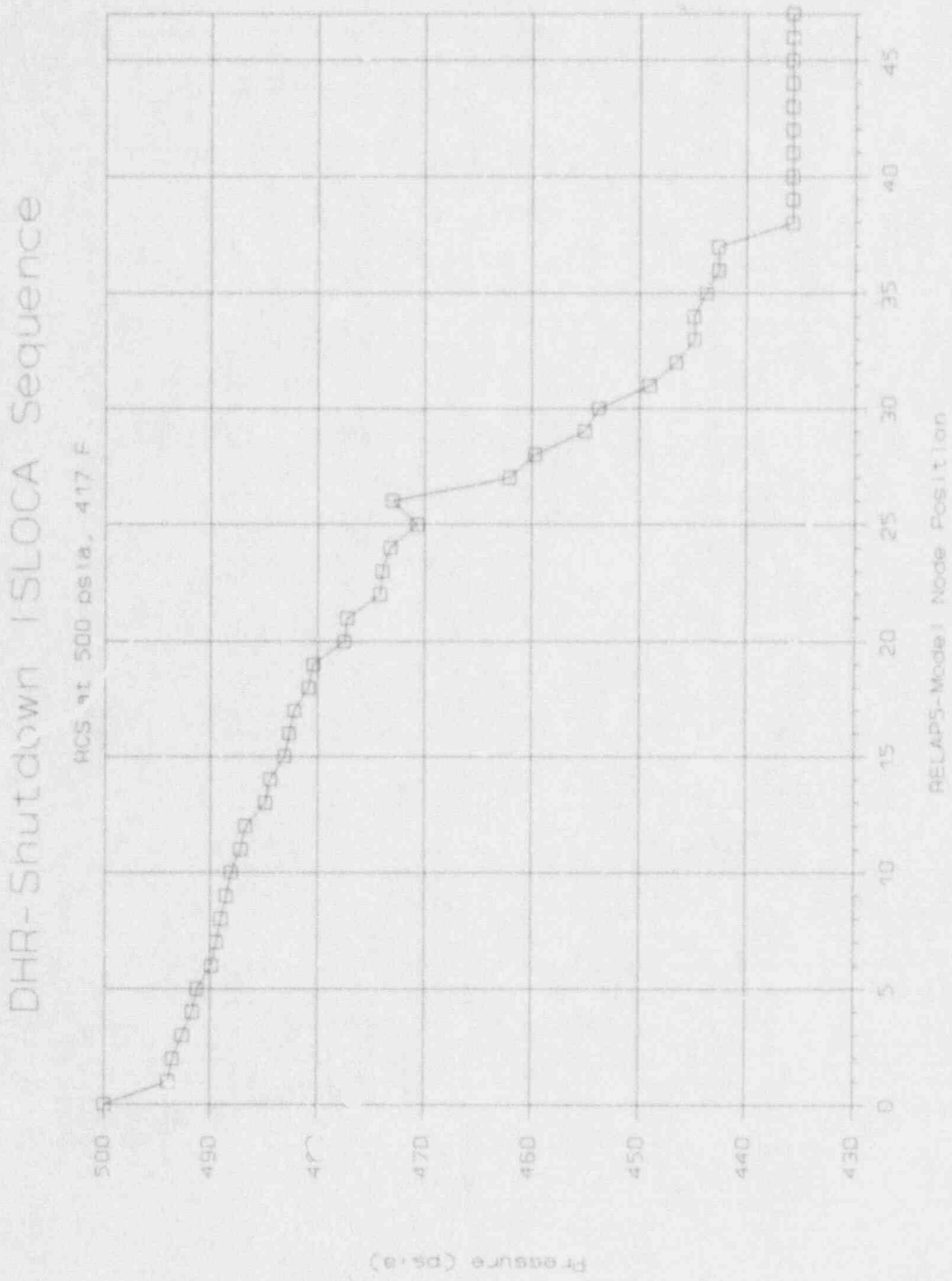


Figure F-13 Steady state pressure in DHR system piping - 500 psia.

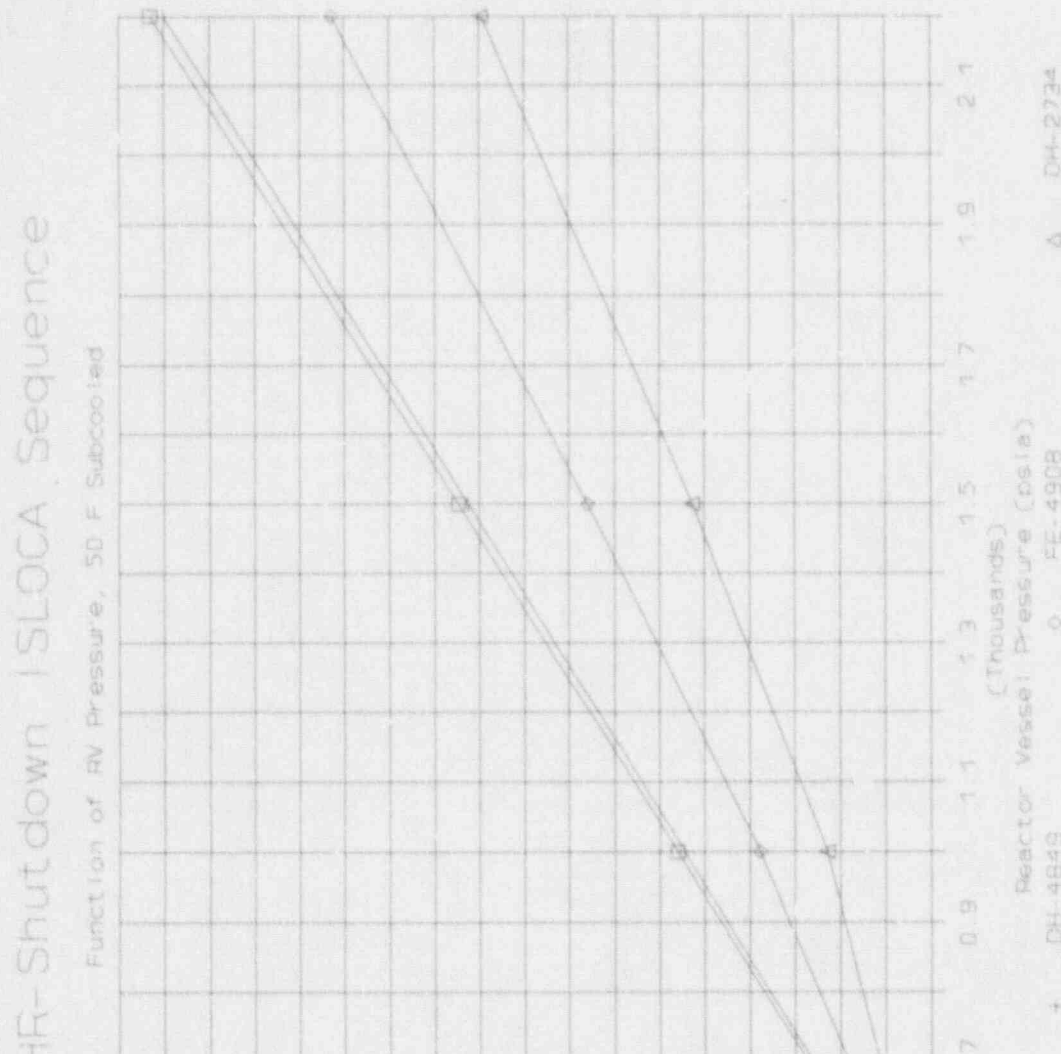
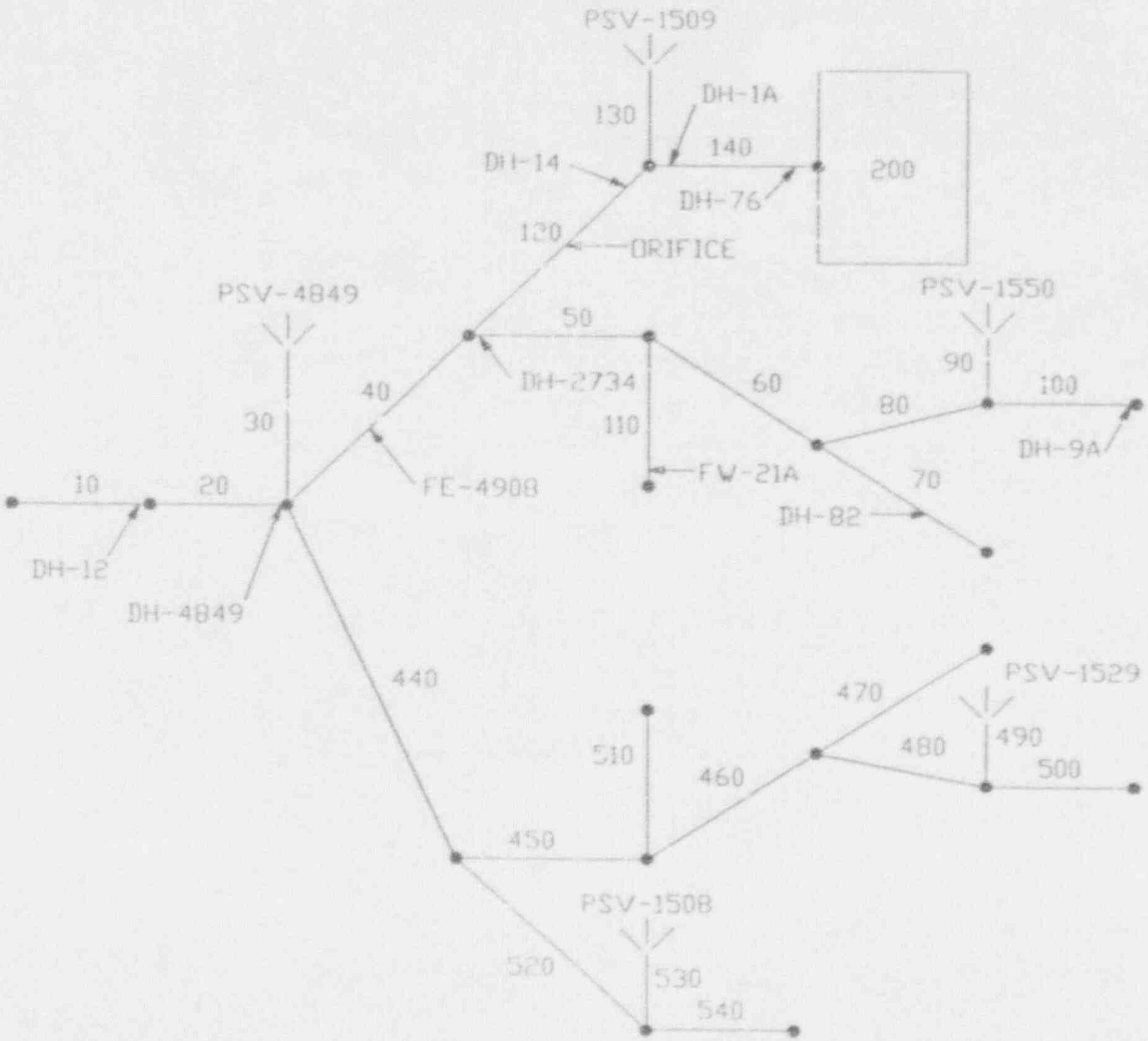


Figure F-14 Component pressure versus reactor pressure for DHR.



DHR	
RELAP PIPE	DAVIS-BESSE COMPONENT
10	12in-CCA-4
20	12in-CCA-4
	12in-GCB-7
30	4in-GCB-7
40 & 440	12in-GCB-7
	2.5in-GCB-7
50 & 450	18in-GCB-8
60 & 460	18in-HCB-1
70 & 470	18in-HCB-1
80 & 480	18in-HCB-1
90 & 490	18in-HCB-1
100 & 500	18in-HCB-1
	14in-HCB-1
110 & 510	10in-HCB-3
120	18in-GCB-8
	2in-GCB
	0.657 orif
	10in-GCB-1
	10in-GCB-10
520	18in-GCB-8
	12in-GCB-8
	10in-GCB-1
	10in-GCB-10
130 & 530	1.5in-10S
140 & 540	10in-GCB-10
	10in-CCB-6
200	Reactor Vessel

Figure F-15 Diagram for RELAP model of LPI system piping.

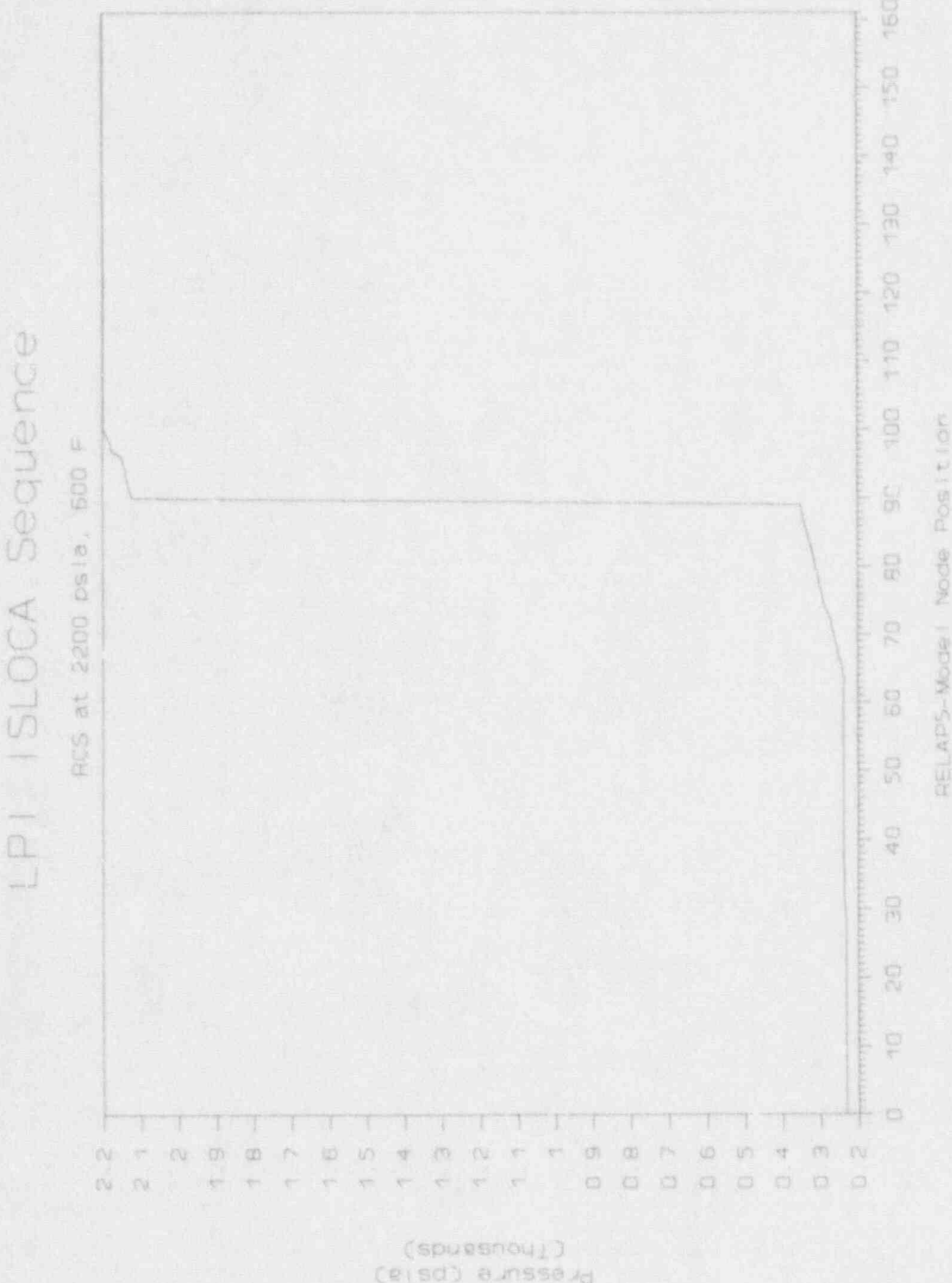
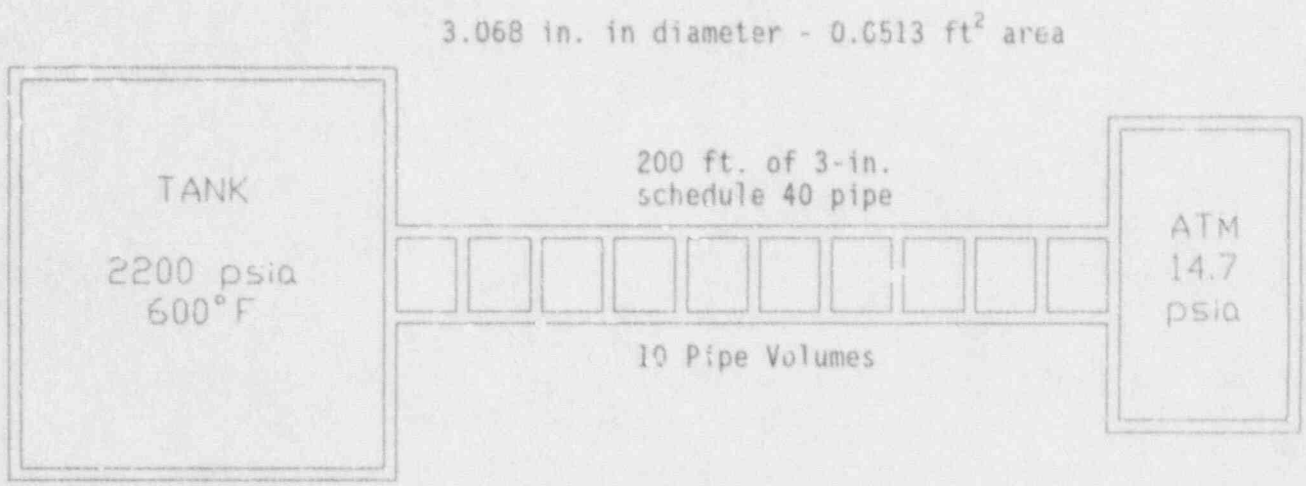


Figure F-16 Steady state pressure in LPI system piping - 2200 psia.



Steady state m = 281.71 lb/s

Steady state reached at 12.7 s

Mass flow = 281.71 lb/0.0513 ft^2s = 5487.0 lb/ft^2s

Figure F-17 MU&P diagram for RELAP simple input model.

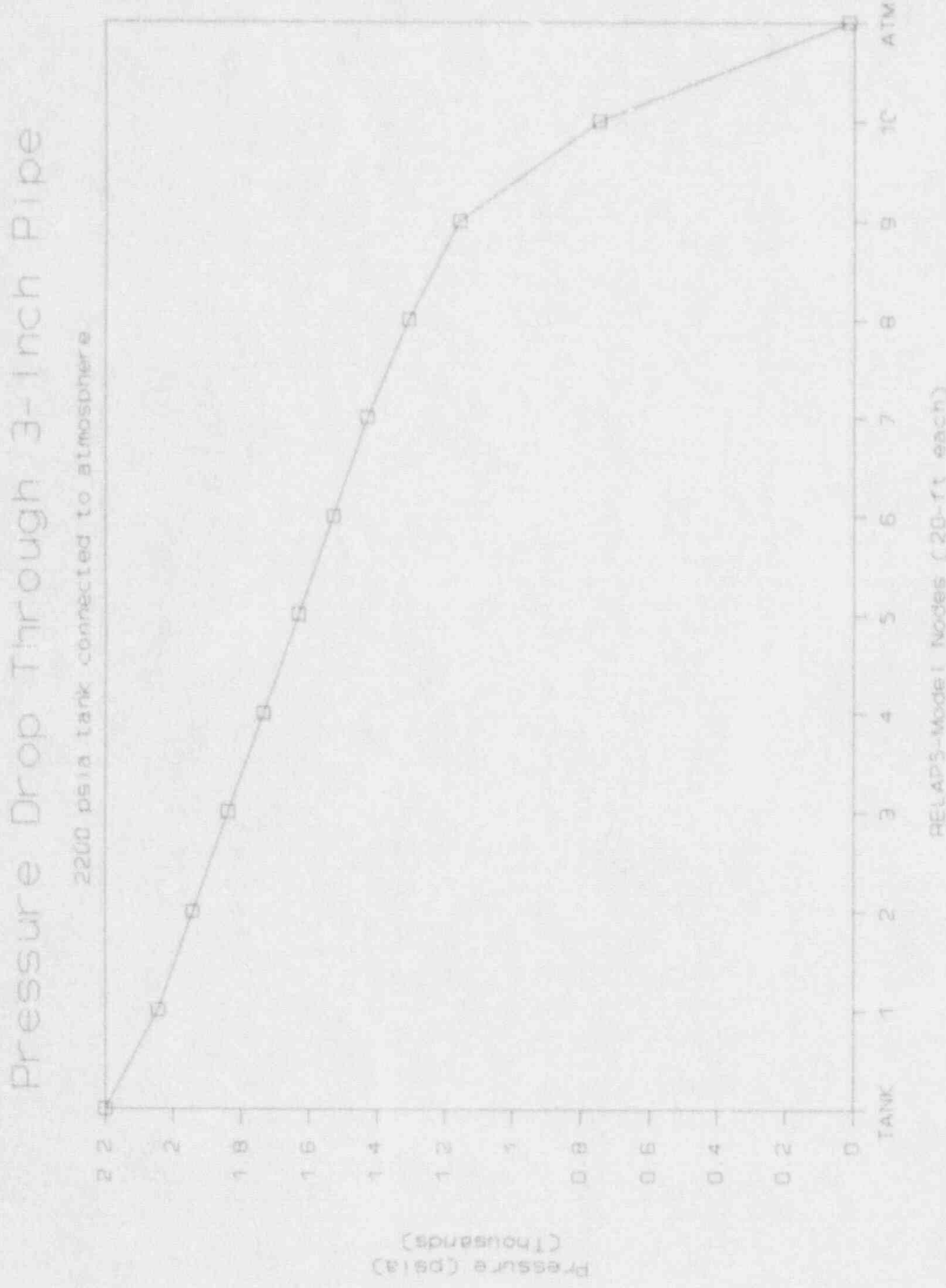


Figure F-18 MU&P pipe pressure versus pipe position - simple model.

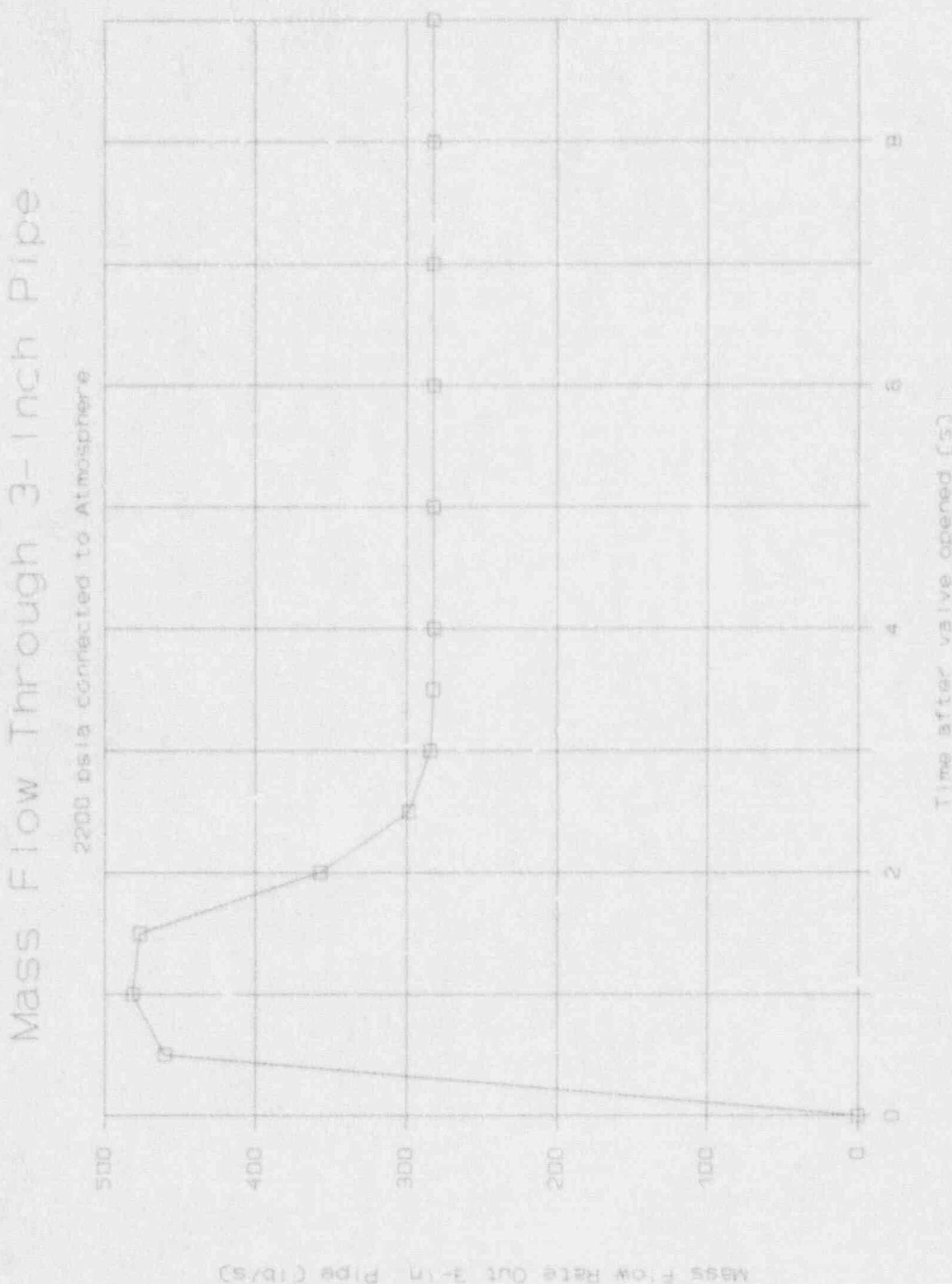


Figure F-19 MU&P exit mass flow rate versus time after valve is opened - simple model.

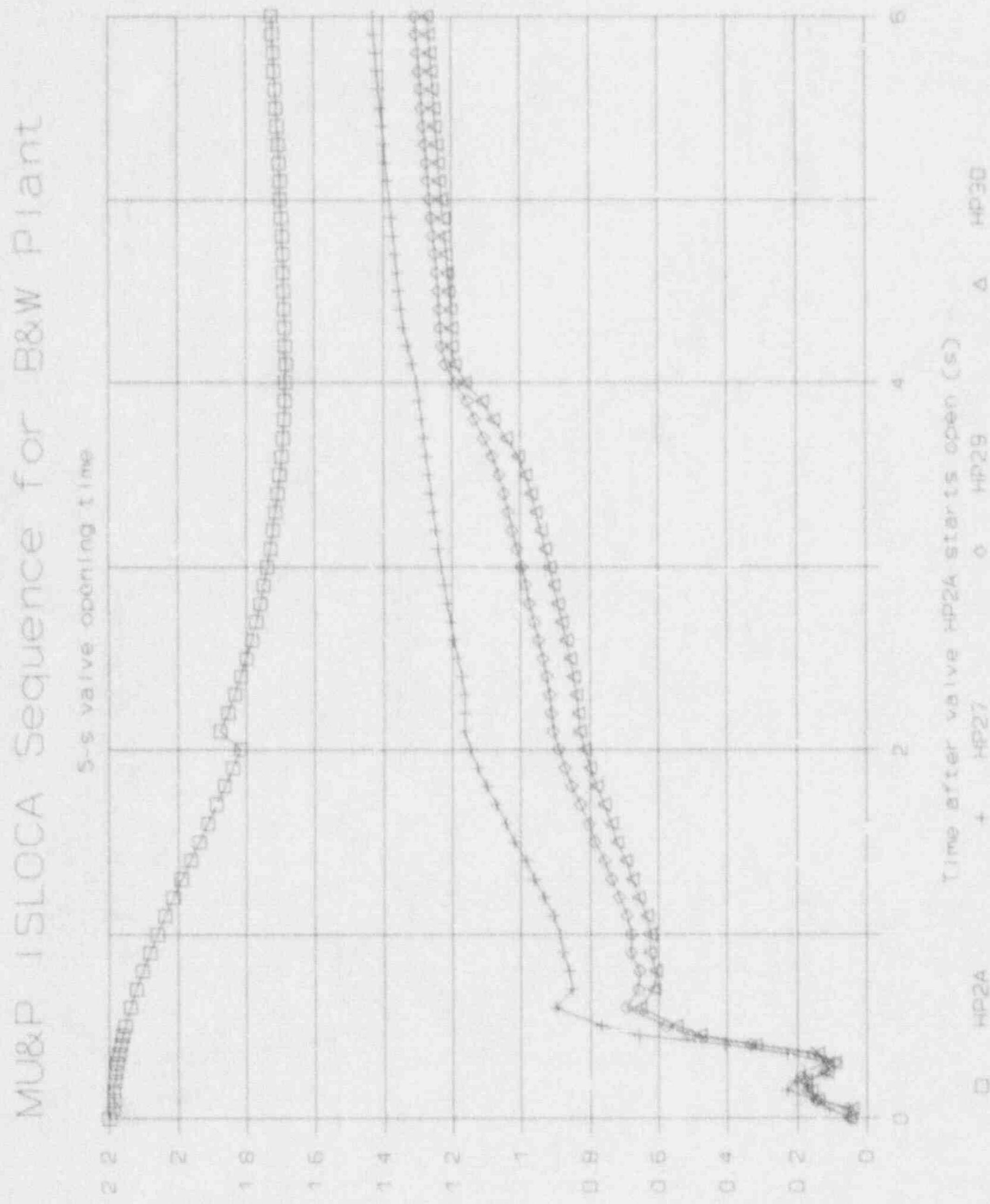


Figure F-20 MU&P mass flow rate histories, detailed model, without orifice, first group.

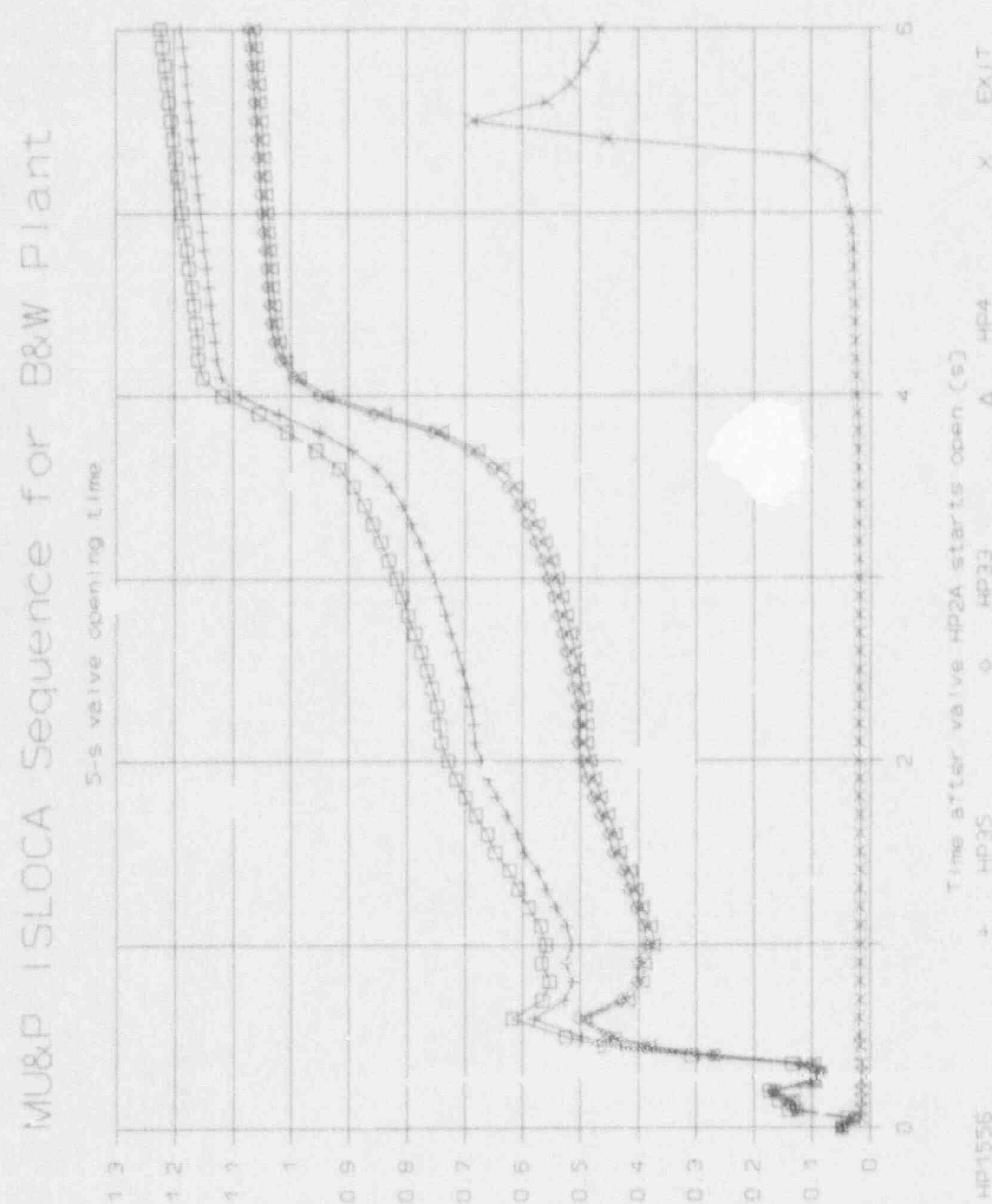


Figure F-21 MU&P mass flow rate histories, detailed model, without orifice, second group.

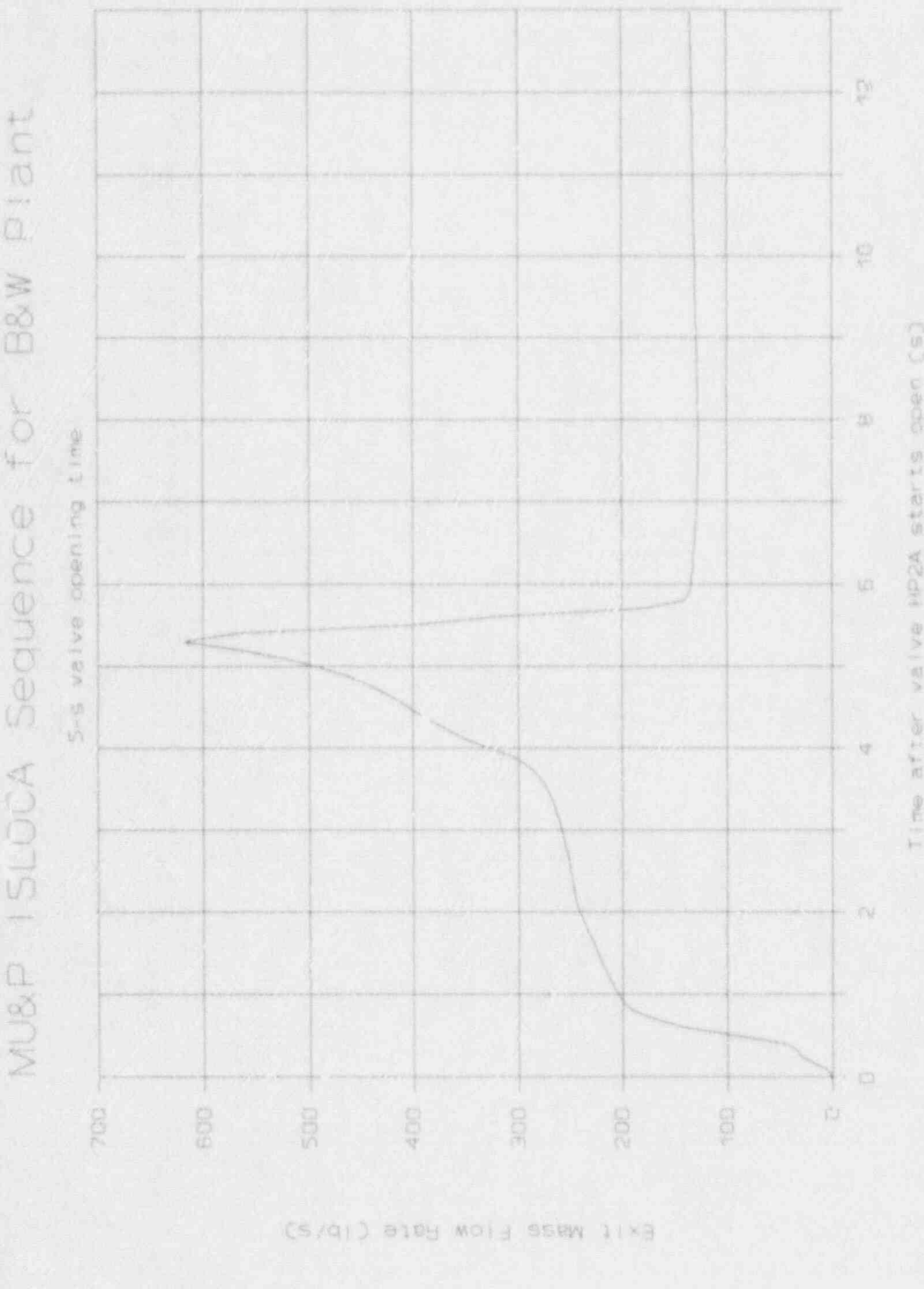


Figure F-22 MU&P exit mass flow rate, detailed model, without orifice.

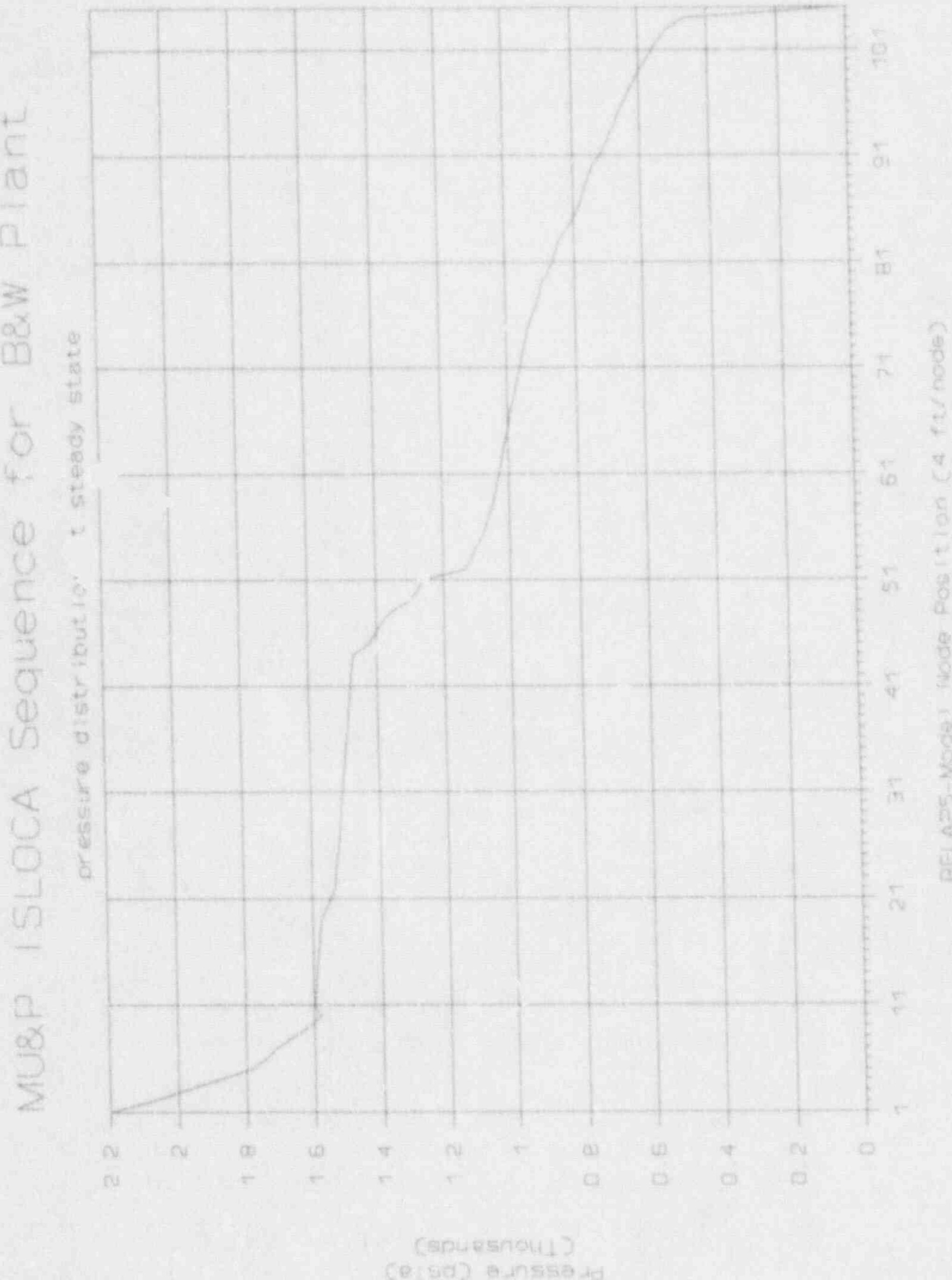


Figure F-23 MU&P pipe pressure versus pipe position, ʼetailed ᵐodel, without orifice.

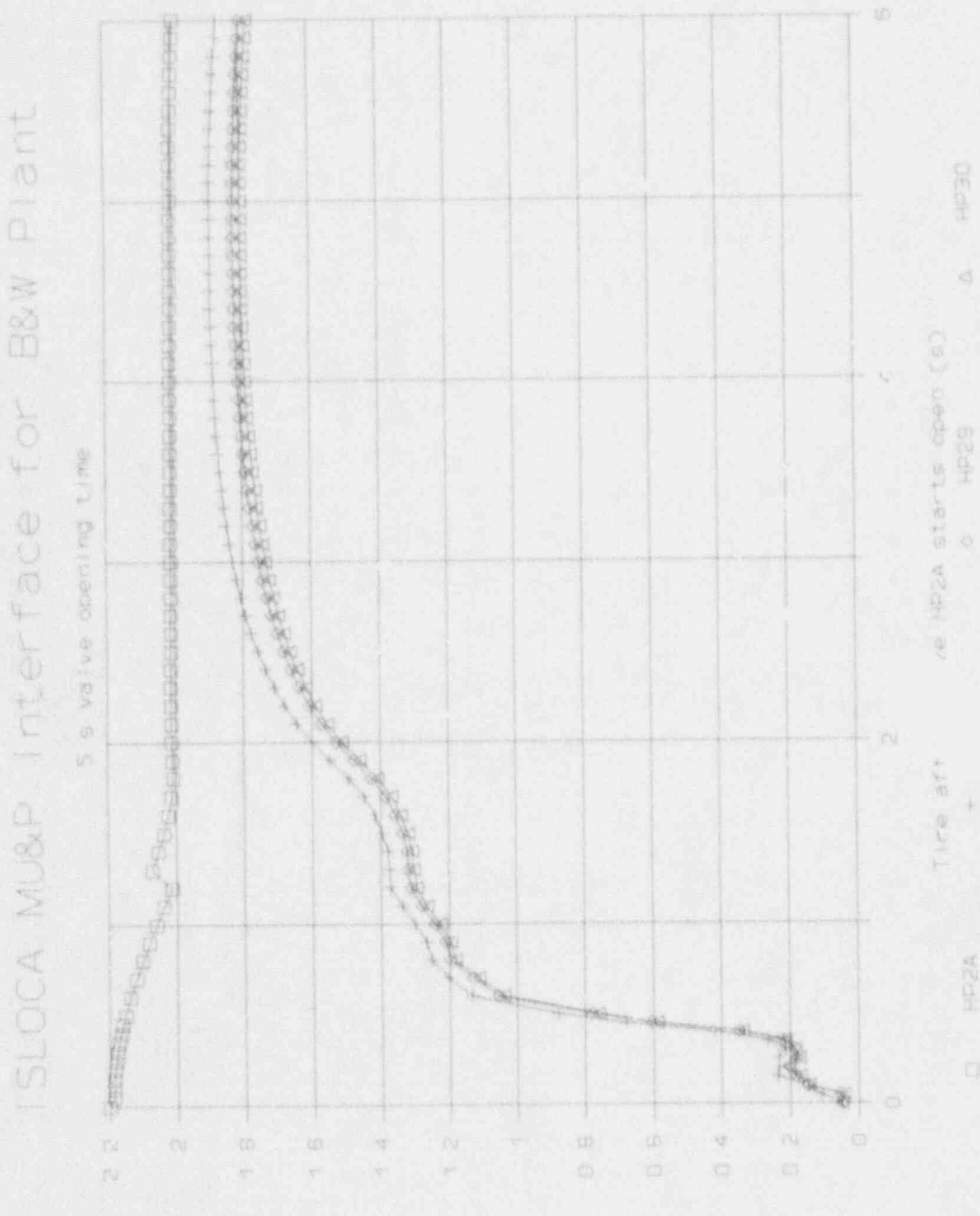


Figure F-24 MU&P mass flow rate histories, detailed model, with orifice, first group.

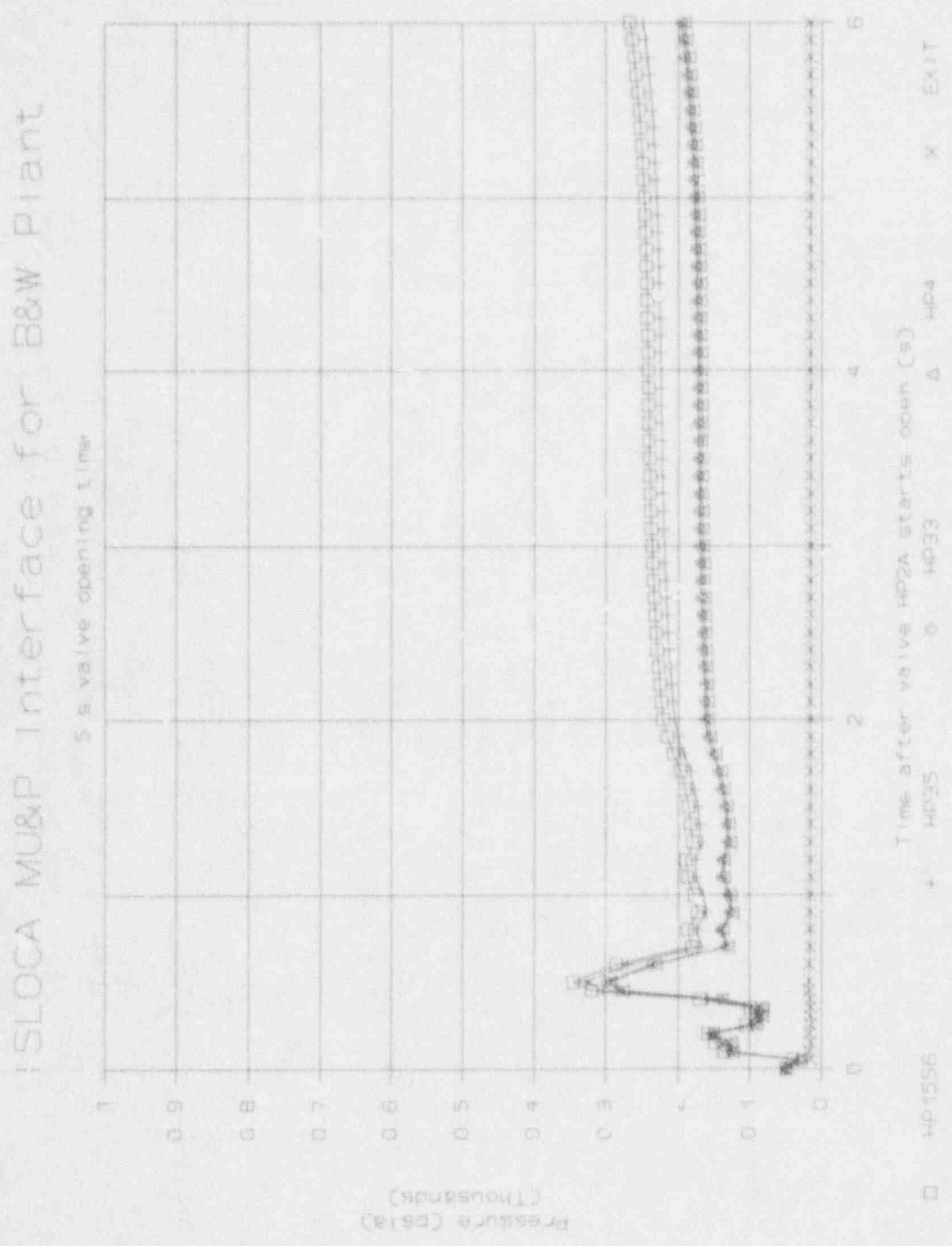


Figure F-25 MU&P mass flow rate histories, detailed model, with orifice, second group.

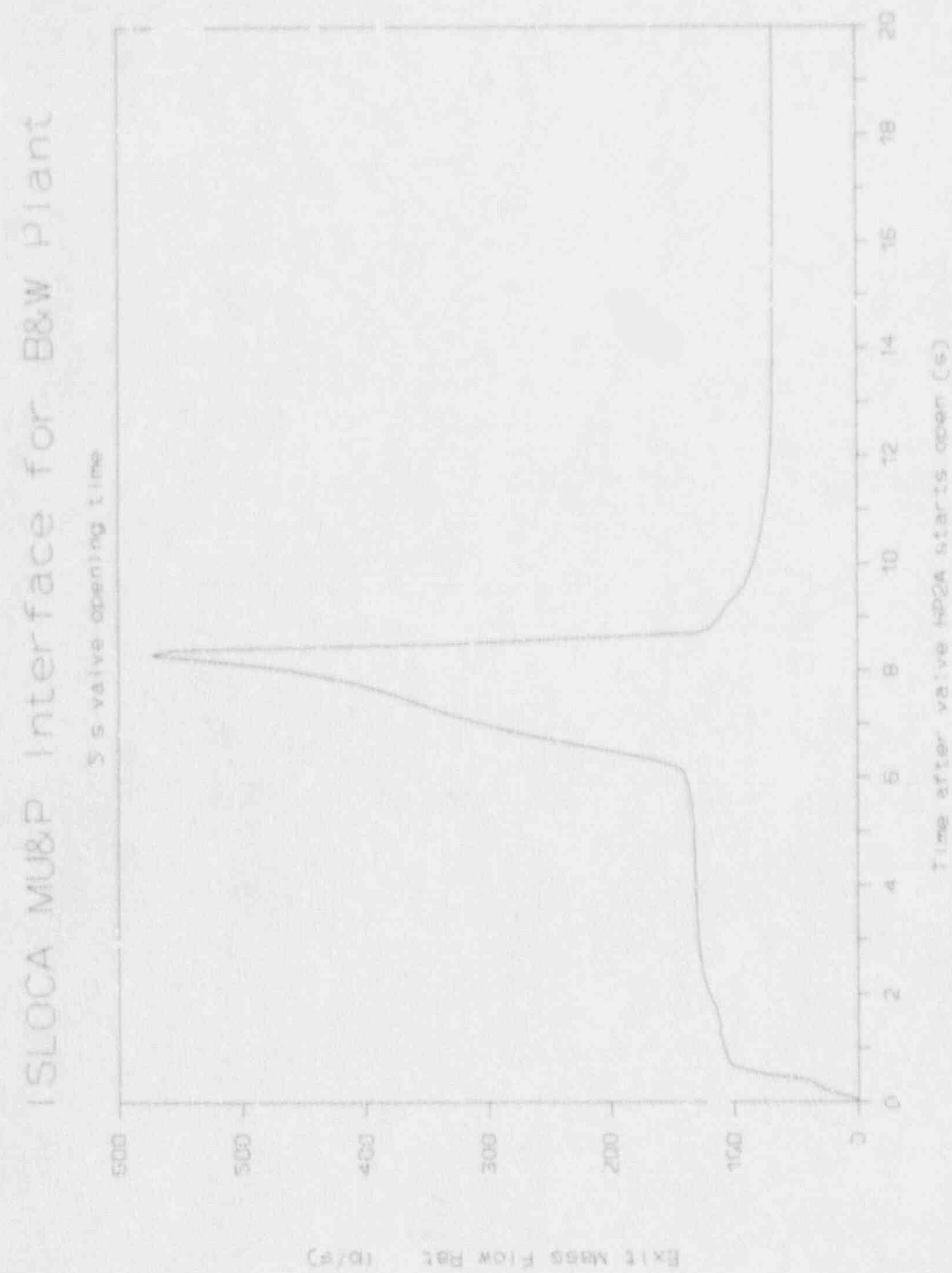


Figure F-26 MU&P exit mass flow rate, detailed model, with orifice.

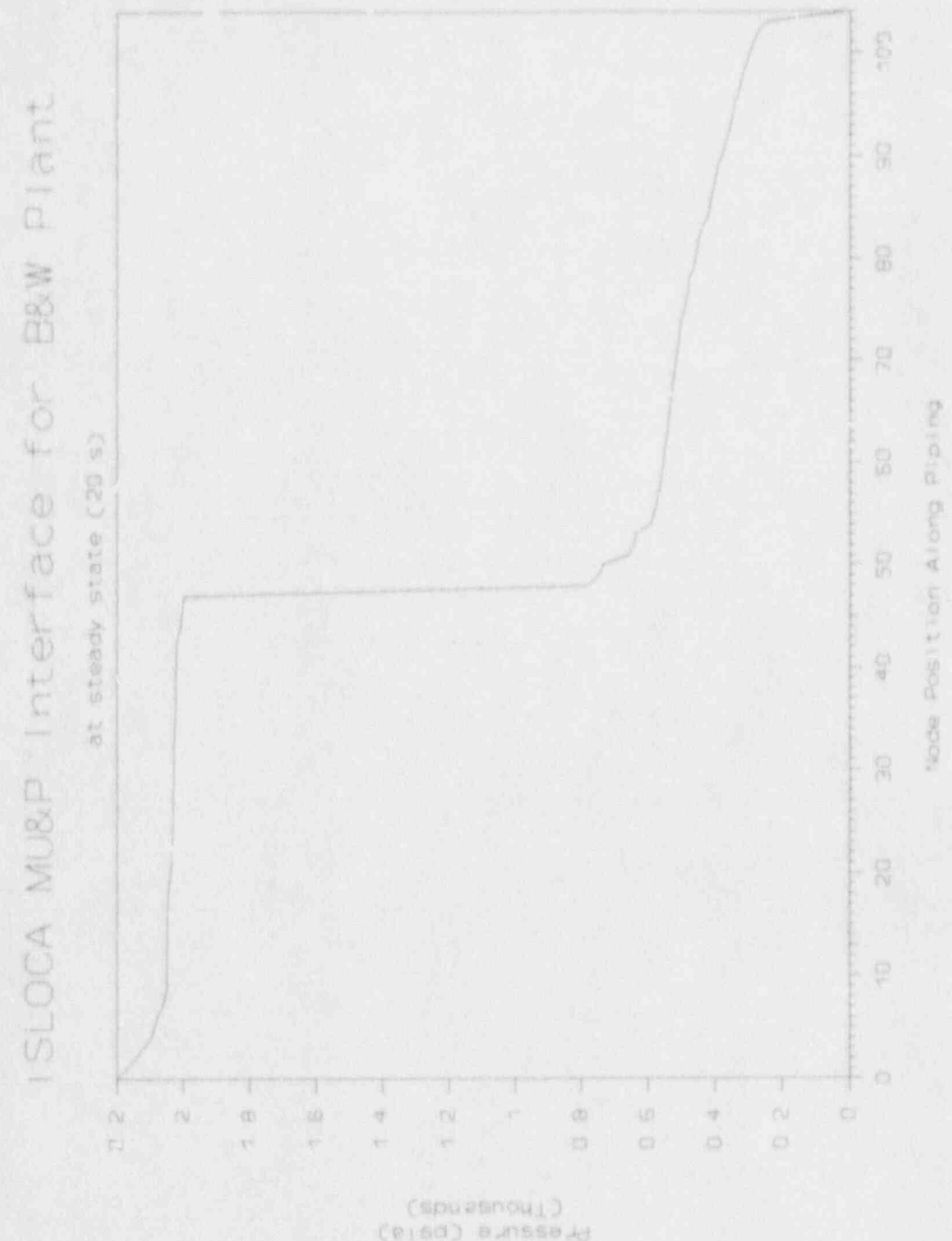
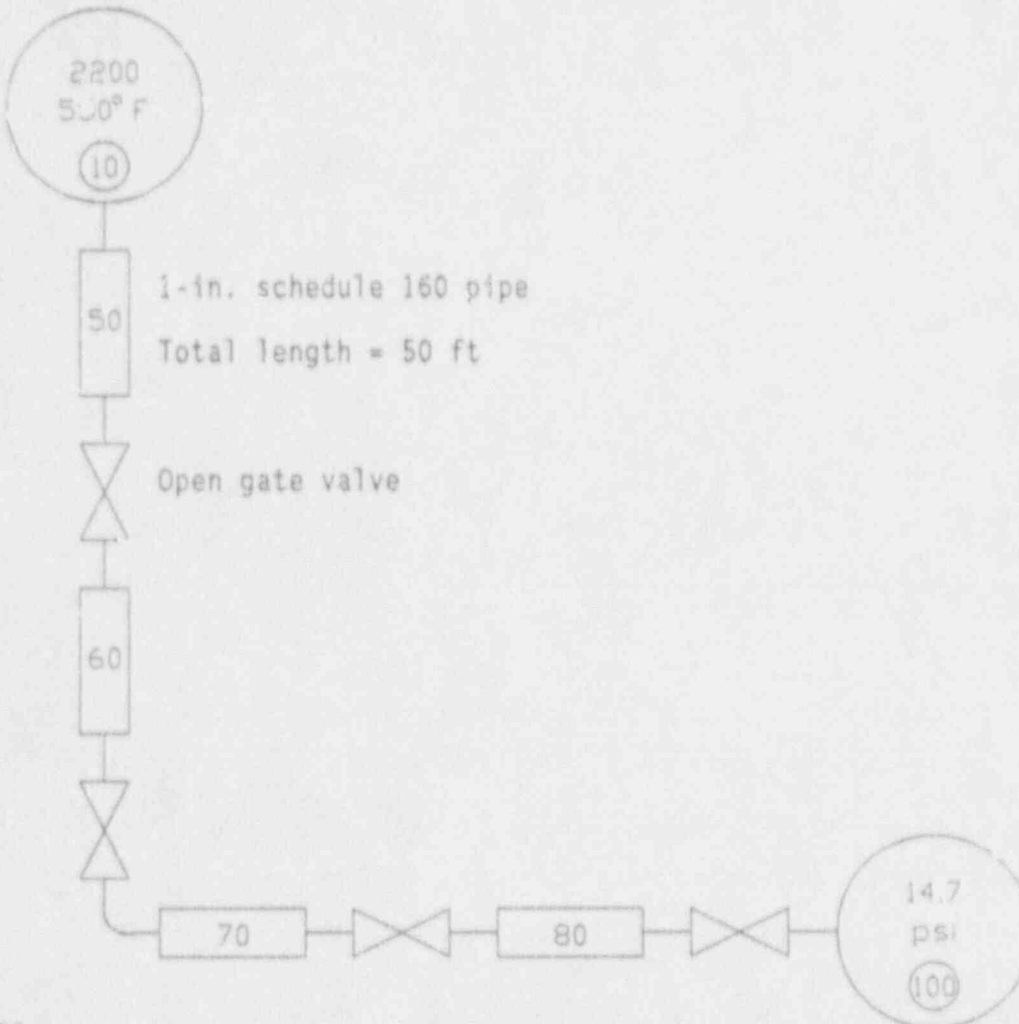


Figure F-27 MU&P pipe pressure versus pipe posit on, detailid model, with orifice.



Area

$$\frac{0.815^2\pi}{4} \times \frac{1\ ft^2}{144\ in^2} = 0.00362\ ft^2$$

Roughness = $1.5E^{-4}$ ft

RELAP Calcs 21.3 lb/s

$$\frac{21.3\ lb}{5} \times \frac{ft^3}{46\ lb} \times \frac{7.45\ gal}{ft^3} \times \frac{60\ s}{min} = \underline{207\ gpm}$$

Assume 0.8 area factor on 3 gate valves

RELAP Calcs 20.9 lb/s

= 203 gpm

Figure F-28 One-inch pipe diagram for RELAP.

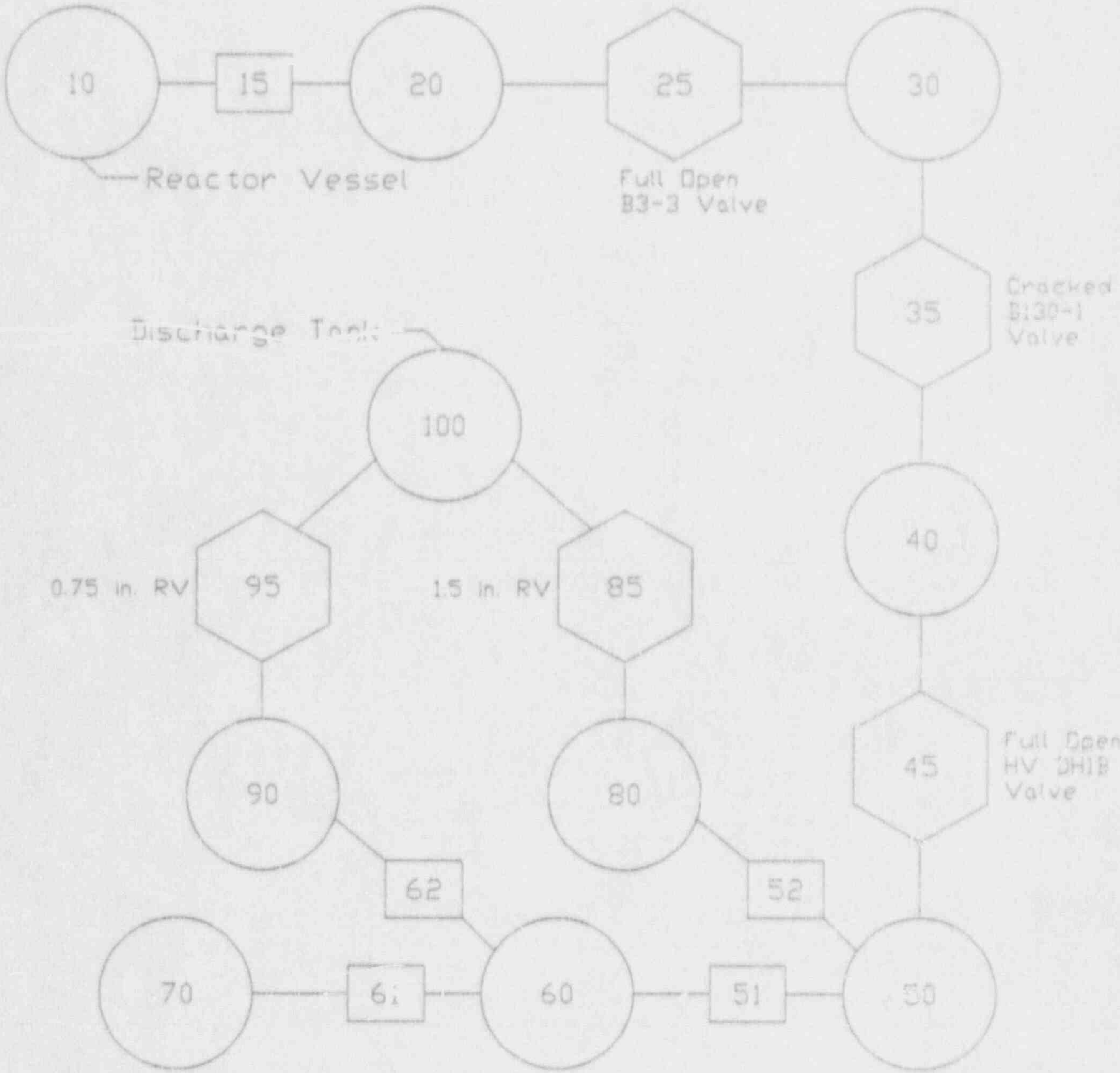


Figure F-29 LPI opening time and break size study diagram for RELAP.

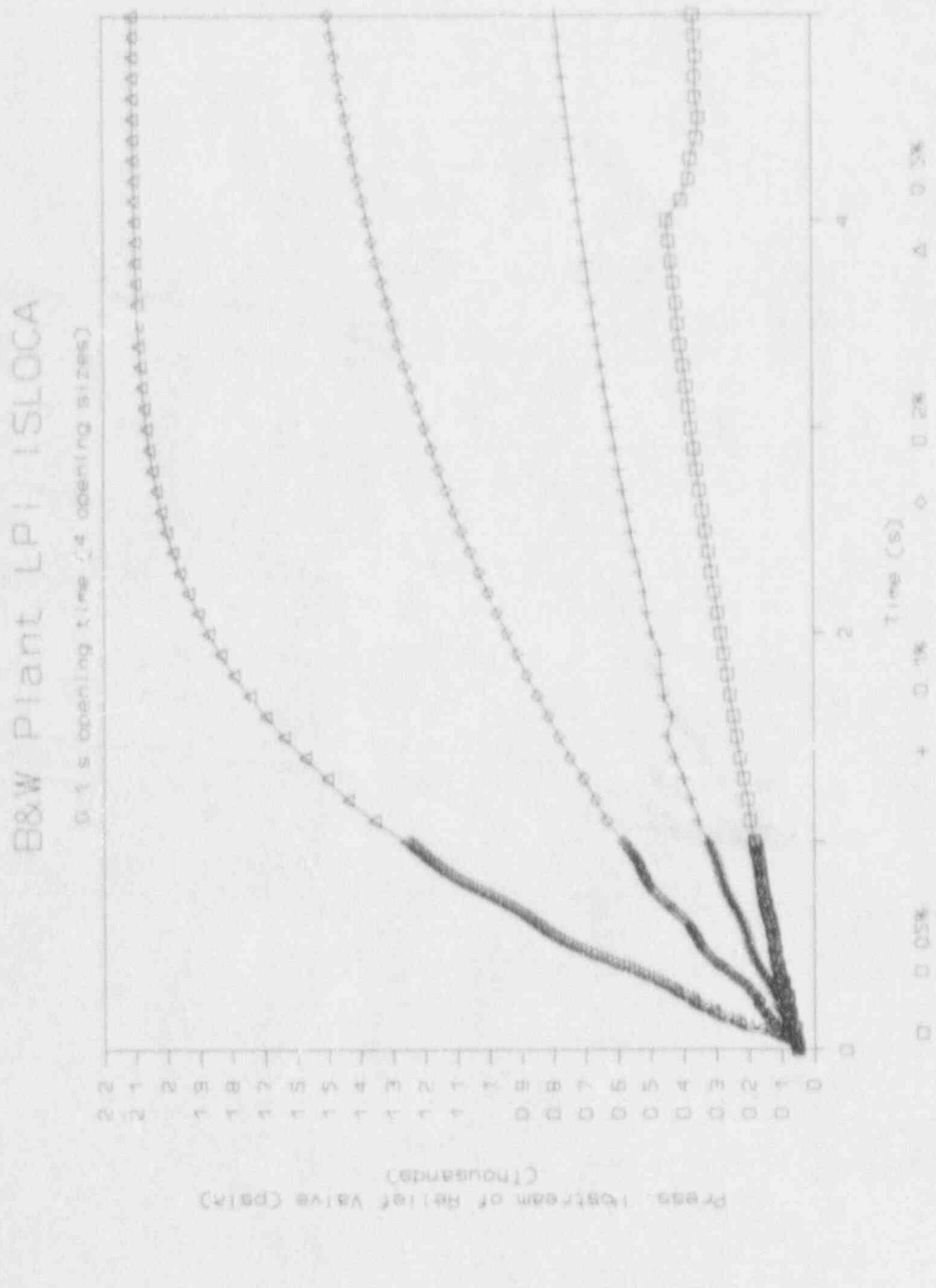


Figure F-30 Pressure at relief valve versus time after valve is opened, for 0.1 second opening time and 0.05 and 0.5% opening area.

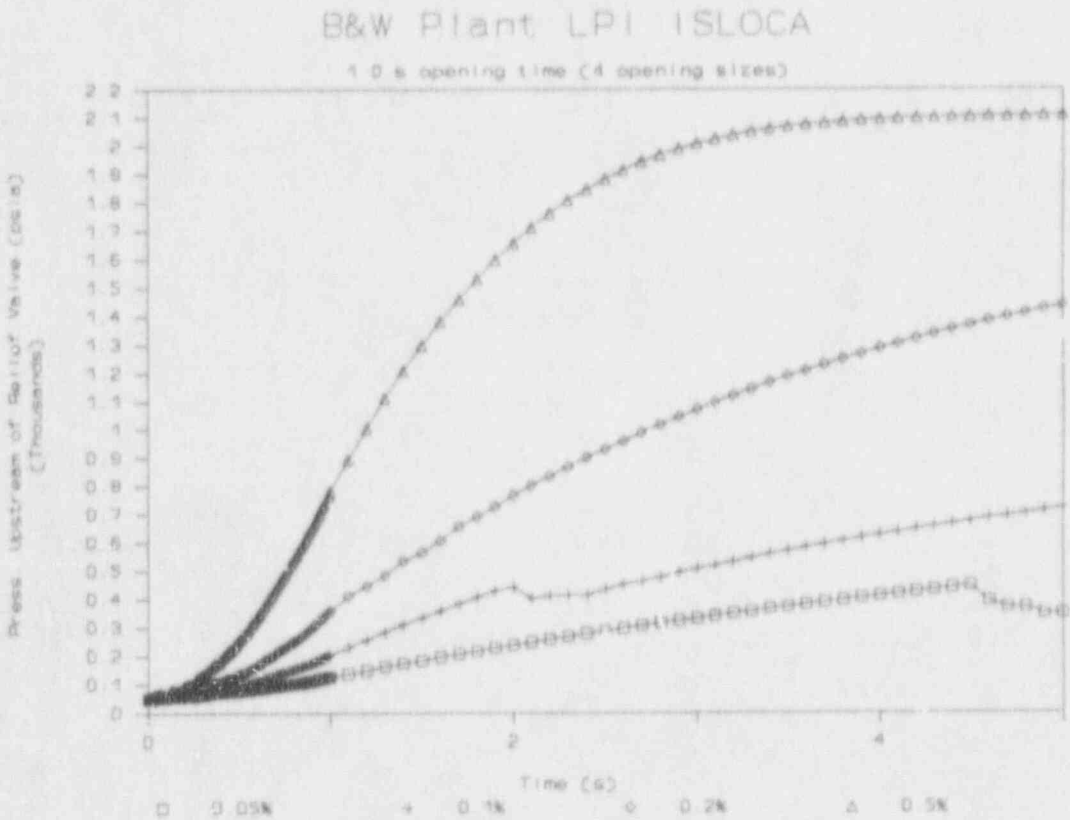


Figure F-31 Pressure at relief valve versus time after valve is opened, for 1.0 second opening time and 0.05 and 0.5% opening area.

Appendix G

Bounding Core Uncovery Time Calculations

C. M. Kullberg

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FIGURES

TABLES

APPENDIX G
CORE UNCOVERY TIME BOUNDING CALCULATIONS

G.0 INTRODUCTION

This appendix provides an estimate of the time required for the core of the B&W reference plant to uncover. This conservative core uncovery time is associated with the interfacing system loss of coolant accidents. The appendix provides uncovery time estimates for both a high-pressure injection (HPI) line break and the low-pressure injection (LPI) line break. Both of these accidents were assumed to be initiated from a full power conditicn.

The core uncovery time estimates are based on a number of conservative assumptions. These assumptions are detailed in Sections G.1 and G.2 of this appendix. The results of these sections provide a lower bound on the drainage time for the refueling water storage tank (RWST). The calculations also provide an estimate of the time subsequent to core uncovery. The estimated core uncovery ti[illegible]s are based on the time at which the vessel's collapsed liquid level has reached the top of the [illegible]ve fuel. A summary of these times is provided in Table G.1.

There are a wide range of uncovery times possi[illegible] This variation is possible if consideration is taken for the various ISLOCA sequences occurring at different initiating pressures and the large number of plants states possible due to operator actions. The analysis provided in this section presents a <u>minimum</u> core uncovery time. These times were used to estimate the Human Error Probabilities for the HPI & LPI ISLOCA sequences.

Table G-1. Summary of ISLOC[illegible] times to the onset of core boil off

	2.5 in. HPI ISLOCA	10 in. LPI ISLOCA
Time to empty RWST (hr)[a]	2.9	.1
Time to onset of core boil off (hr)	4.0	1.9

a. All times referenced to the beginning of the ISLOCA.

G.1 BOUNDING CALCULATIONS FOR THE SMALL BREAK ISLOCA

G.1.1 Introduction

This section documents the core uncovery time calculations for a small break interfacing systems LOCA. These calculations are used to estimate the minimum time required for the onset of core uncovery. The core uncovery time estimate includes a boil down time after ECCS pump suction was lost from the RWST & CFT volumes.

The core uncovery time is defined in this analysis as the time at which the collapsed vessel liquid level drops below the top of the core's active fuel. This definition is not the true core uncovery time due to the presence of a void fraction distribution in the core. The use of the collapsed water level at the top of the core in these assessments provides a conservative indication of the operator response times available before core damage occurs.

G.1.2 Time to Empty the RWST & CFT

The ISLOCA small break was assumed to occur outside of the containment. The break occurs on one of the HPI injection lines. The HPI lines are 2.5-in. pipe immediately outside of the containment. As a consequence, the break was modeled as a 2.5-in. diameter leak. No credit was given for form losses, wall friction, or other pressure drop effects that may reduce the break mass flow rate. These break assumptions were incorporated into a simplified five volume RELAP5 model. This model was used to estimate the time required to empty the RWST & CFT. The primary system's ECCS and the break were explicitly modeled. The assumed break configuration leads to a shorter and a more conservative time estimate of when the RWST & CFT volumes empty.

Several additional assumptions were incorporated into the RELAP5 model to ensure a conservative time estimate to empty the RWST:

- The ECCS water supply was limited to 400,000 gal.
- Auxiliary feedwater was available and it was assumed that the steam generators were depressurized to enhance primary to secondary heat transfer. Thus it was assumed all core decay energy is removed by the steam generators. Or equivalently, no

stored energy or decay heat was included in the RELAP5 model. This resulted in a lower primary system pressure and a higher ECCS mass flow rate.

- It is assumed that the break flow rate will stabilize near the LPI shutoff head and the time averaged break mass flow rate will balance with the time averaged ECCS mass flow rate. This is typical of many small break LOCA's. The steady-state ECCS mass flow rate was modeled by equating it with the steady-state choked break mass flow rate at the LPI activation pressure. This time averaged balance accounted for short intermittent periods when the LPI would activate. Continuous LPI operation was not possible because the total LPI mass flow rate was significantly larger than the break mass flow rate.

- The CTF's are available and the drain out time will be estimated by dividing the total break mass flow rate by the total volume of the CTF liquid inventory. The CTF's were not in the RELAP5 model and this time estimate was done separately.

- It is assumed that the transient is initiated at 100% power conditions and that there is no significant delay for reactor scram.

- One HPI, 2-LPI, and 2-charging trains were modeled to refill the primary system.

Figures G-1 and G-2 describe the primary pressure response and the ECCS/break flow rates for the simplified RELAP5 simulation. These figures indicate that after about one hour the primary system pressure will stabilize near 200 psia. The ECCS/break flow rates will be approximately 330 lbs/sec. This mass flow rate of 330 lbs/sec will deplete the RWST's volume of 400,000 gallons in about 2.8 hours.

The time required to deplete the CFT's volume is modeled in the same manner as the depletion of the RWST's volume. The total volume of the CFTs equals 2080 ft^3. Once the CFTs are activated it is assumed that they empty at an average rate equal to the ECCS/break flow. The ECCS/break mass flow rate of 330 lbs/second provides the time estimate for the draining of the CFTs. The CFT's tank volume of 2080 cubic feet is then emptied in about 0.11 hours with this break flow. The total time to drain the RWST's and CFT's coolant inventory is then about 2.9 hours.

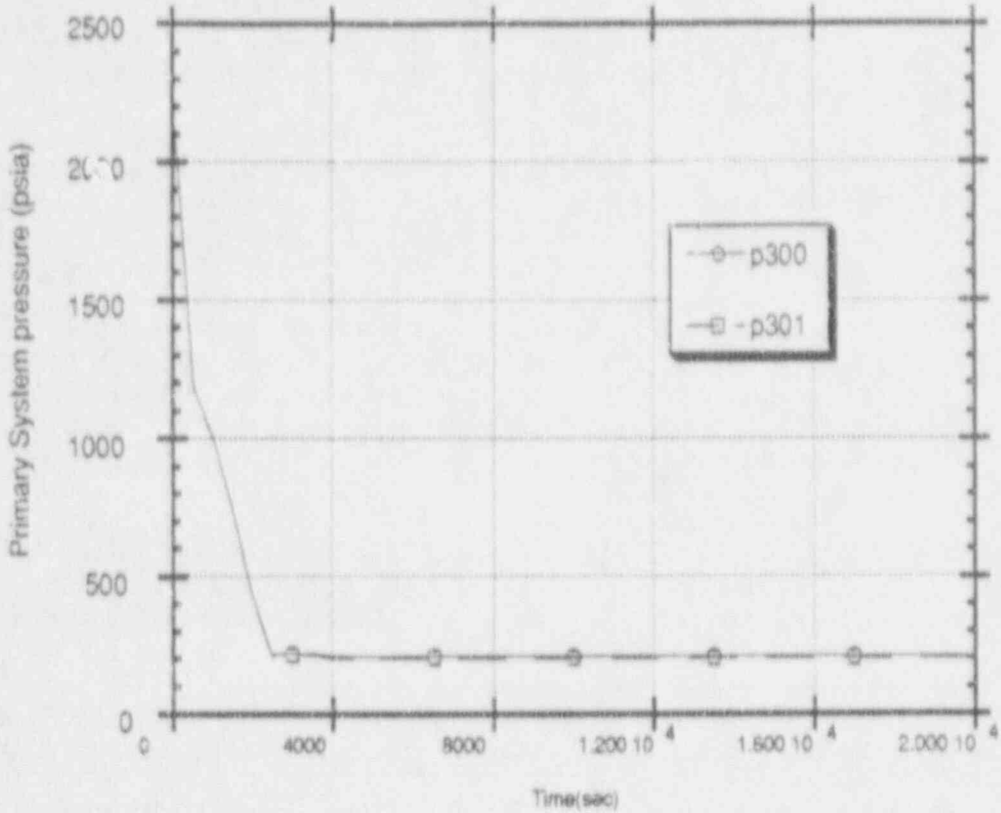

Figure G-1. Small break ISLOCA showing primary system pressure

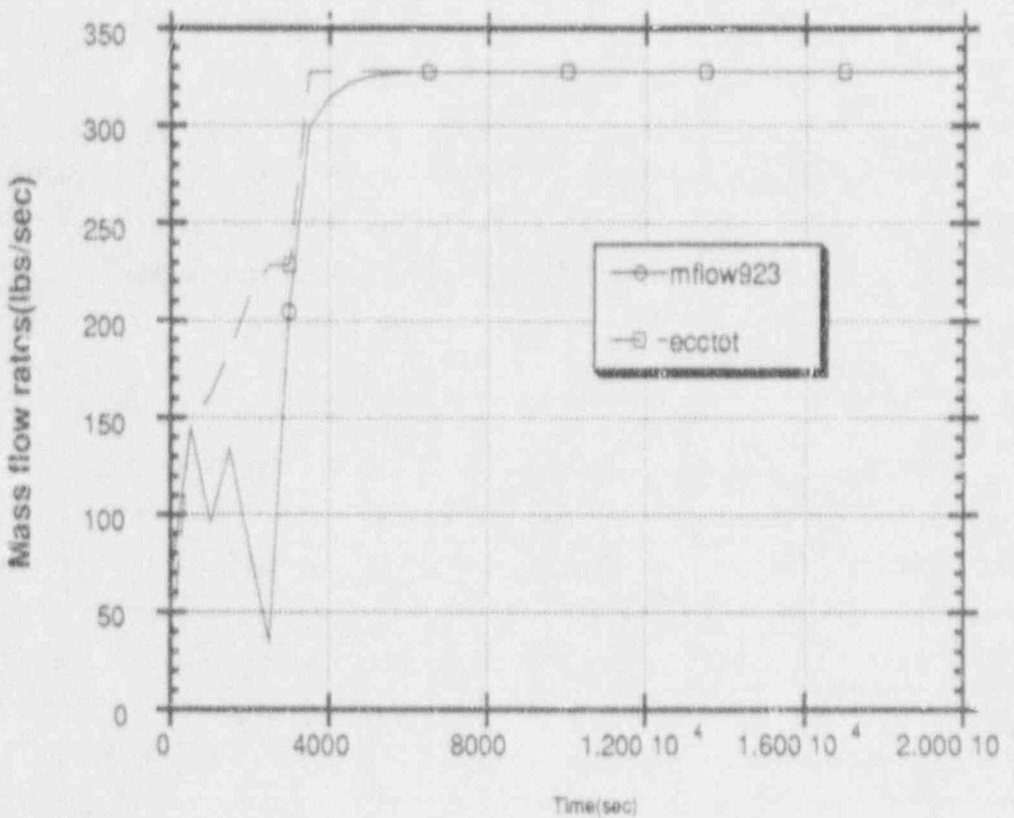

Figure G-2. Small break ISLOCA showing mass flow rates

stored energy or dec. heat was included in the RELAP5 model. This resulted in a lower primary system pressure and a higher ECCS mass flow rate.

- It is assumed that the break flow rate will stabilize near the LPI shutoff head and the time averaged break mass flow rate will balance with the time averaged ECCS mass flow rate. This is typical of many small break LOCA's. The steady-state ECCS mass flow rate was modeled by equating it with the steady-state choked break mass flow rate at the LPI activation pressure. This time averaged balance accounted for short intermittent periods when the LPI would activate. Continuous LPI operation was not possible because the total LPI mass flow rate was significantly larger than the break mass flow rate.

- The CTF's are available and the drain out time will be estimated by dividing the total break mass flow rate by the total volume of the CTF liquid inventory. The CTF's were not in the RELAP5 model and this time estimate was done separately.

- It is assumed that the transient is initiated at 100% power conditions and that there is no significant delay for reactor scram.

- One HPI, 2-LPI, and 2-charging trains were modeled to refill the primary system.

Figures G-1 and G-2 describe the primary pressure response and the ECCS/break flow rates for the simplified RELAP5 simulation. These figures indicate that after about one hour the primary system pressure will stabilize near 200 psia. The ECCS/break flow rates will be approximately 330 lbs/sec. This mass flow rate of 330 lbs/sec will deplete the RWST's volume of 400,000 gallons in about 2.8 hours.

The time required to deplete the CFT's volume is modeled in the same manner as the depletion of the RWST's volume. The total volume of the CFTs equals 2080 ft^3. Once the CFTs are activated it is assumed that they empty at an average rate equal to the ECCS/break flow. The ECCS/break mass flow rate of 330 lbs/second provides the time estimate for the draining of the CFTs. The CFT's tank volume of 2080 cubic feet is then emptied in about 0.11 hours with this break flow. The total time to drain the RWST's and CFT's coolant inventory is then about 2.9 hours.

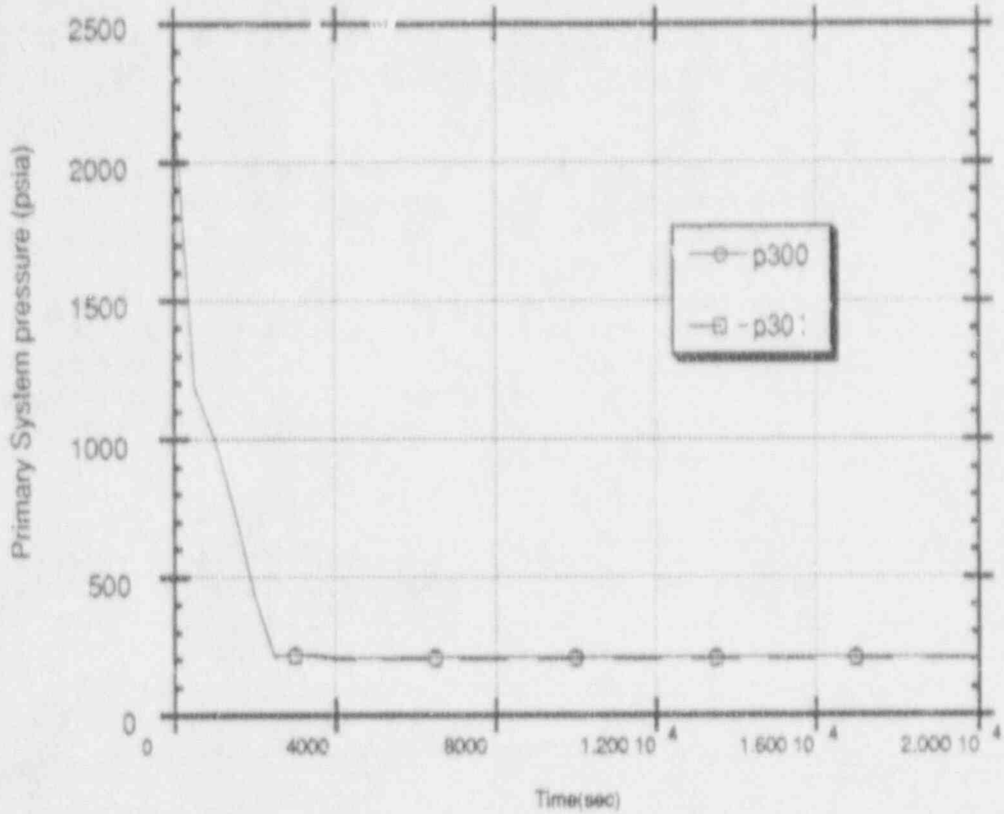

Figure G-1. Small break ISLOCA showing primary system pressure

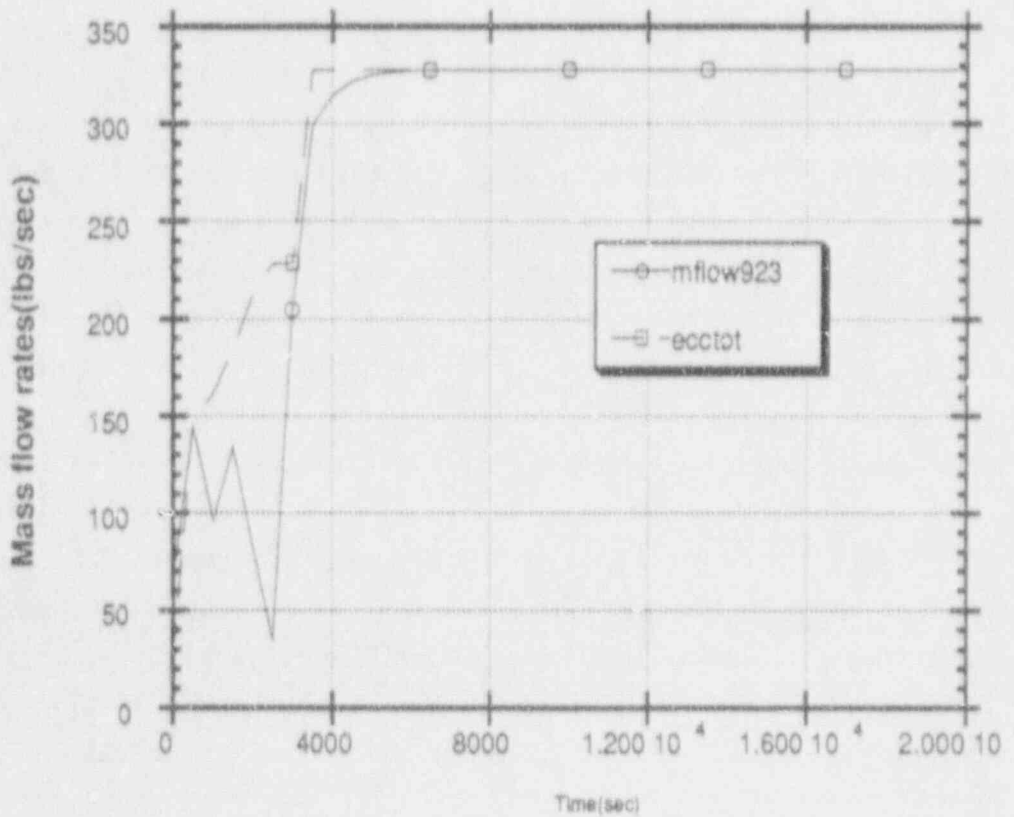

Figure G-2. Small break ISLOCA showing mass flow rates

The are several sources of uncertainty in the estimate of the time for the RWST and CFT to empty. These uncertainties are listed as follows:

- The fact that the primary system pressure is a nonconstant nonlinear function of time.
- The HPI and charging mass flow rates are nonlinear functions of the primary system pressure.
- Primary to secondary heat transfer may greatly alter outcome depending whether the operator decided to depressurize the steam generators.
- Flooding out the auxiliary building may disrupt or destroy the ECCS pumping equipment before the RWST tank is empty.
- Intrinsic uncertainties in the RELAP5 critical flow model.

In a small break ISLOCA simulation that is more typical of actual conditions the secondary pressure would be above 1000 psia for several hours. At these secondary pressures the primary system will be maintained above the LPI shutoff head. The proper modeling of the primary system pressure would necessitate modeling primary to secondary heat transfer as well as stored energy and core decay heat in RELAP5. These RELAP5 models were not incorporated in the analysis presented in this section. These models were not incorporated for the expressed purpose of minimizing the time to empty the RWST and CFT.

G.1.3 Core's Upper Plenum Boil Down Time Calculation

This section will detail the procedure used to estimate the time required for the core's upper plenum coolant to boil off. This calculation is based on the assumption that no additional ECCS coolant is injected into the primary system after 2.9 hr.

To estimate the time for the core's upper coolant plenum to boil off the following assumptions were made:

1. No liquid in the loop regions is available to be heated by the core. The liquid in the loop regions is assumed to either exit the break or reside in the loop seals. This assumption is

conservative because after the RWST has drained, some liquid that is in the primary loops will drain back into the vessel region. It is not possible to make an estimate of how much liquid would actually drain into the vessel unless a full-systems calculation is completed.

2. It was assumed that the vessel upper head is completely drained by the time the RWST has emptied. This assumption is reasonable since some small break PWR LOCA scenarios lead to vessel upper head voiding.

3. It was assumed that any sensible or latent heat added to the vessel liquid will result in no significant repressurization of the primary system and therefore the total integrated core decay power goes to initiate core boil off.

4. It was assumed that the remaining liquid in the vessel available for boil off to be at the bulk subcooled conditions of 100°F at a pressure of 200 psia. This assumption is not based on rigorous quantitative arguments.

Two parameters must now be calculated. These parameters are required to determine the time interval required for the collapsed coolant level to reach the top of the active fuel. The two parameters are the total energy required to reduce the coolant level to the top of the active fuel and the integral of the power generation rate of the core. The time interval is determined by integration of the core's power level until the core's energy output equals the energy required to reduce the coolant level.

The total energy required for the reduction of the coolant level is composed of two contributions. The first contribution is the energy required to raise the temperature of the reactor vessel's total coolant inventory to saturation at 200 psig. The second contribution is the energy required to vaporize the coolant remaining in the reactor vessel above the active fuel region. The calculation of the minimum energy required to raise the coolant's temperature to saturation requires the following information:

- The enthalpy of the subcooled vessel liquid at 200 psia & 100°F,
- The saturation enthalpy of the vessel liquid at 200 psia.
- The bulk density of the liquid at 100°F.
- The coolant volume of the reactor vessel minus the upper head volume.

The appropriate coolant volume values for the B&W reference plant were utilized in the calculation. The energy required to raise the coolant remaining in the vessel after RWST & CFT depletion to its saturation temperature is 5.47E10 joules.

The calculation of the second energy contribution required to estimate the time for core uncovery is described next. The calculation begins with the liquid in the reactor vessel at bulk boiling conditions. The energy added to the vessel's liquid is assumed to result in liquid vaporization. This is based on the assumption that no liquid in the loop regions will drain into the vessel. All the liquid above the core is assumed to be turned into steam. The energy needed to vaporize the liquid region above the active core region is equal to energy required to vaporize the saturated liquid in the vessel upper plenum. The calculation of the vaporization energy requires the following information:

1. The upper plenum's volume of saturated liquid,
2. The latent heat of vaporization,
3. The saturation density of the liquid.

With this information a total energy of 4.85E10 joules is required to vaporize the upper plenum water. The vaporization of this water results in the collapsed water level dropping to the top of the active fuel.

The total energy required to reduce the reactor vessel's coolant level to the top of the active fuel is the contribution of the sum of the subcooled and vaporization energies. These two energy values sum to 1.0E11 joules for the B&W reference plant.

The minimum time required for the core to uncover is determined by integrating the reactor's decay heat power curve from the time the RWST & CFT emptied to the point in where the integral equals 1.0E11 joules. It was assumed in this analysis that the reactor scram started at the same time the ECCS flow was initiated by the ISLOCA transient. This assumption is conservative since a delay in the core scram allows the primary system to remain pressurized and as a result prevents or reduces the ECCS coolant flow.

The normalized ANS core decay power curve was used to estimate the reactor's decay heat as a function of time. This decay heat curve was fit to a quadratic polynomial by the Mathematica curve fitting routines. This curve fit was then integrated between the time the RWST & CFT coolant volumes were depleted and the unknown core uncovery time. This polynomial equation was normalized to the initial core power. The time required to achieved a collapsed water level equal to the top of the active fuel was determined by using the Mathematica algebraic/numerical routines.

The data used to develop the decay heat curve is the normalized decay curve from the standard ANS decay model. The data points in the following table were used to develop the curve fit used in the analysis.

The integral of core's decay heat curve from the time the RWST & CFT empties is required to estimate the time when the core's coolant temperature reaches saturation and the time the core "uncovers". A regression fit to the

Table G-2. ANS Normalized Decay Heat Curve for the H.B. Robinson Plant (From RELAP5)

Time Seconds	Normalized Power
100.	0.0331
400.	0.0235
800.	0.0196
1000.	0.0185
2000.	0.0157
4000	0.0128
8000.	0.0105
10000.	0.00965
20000.	0.00795
60000.	0.00566
100000.	0.00475

H.B. Robinson decay heat curve was utilized to provide the decay heat values required. The regression fit of the ANS data gives:

$$Pd(t) = 0.012 - 1.939 * 10^{-7} * t + 1.215 * 10^{-12} * t^2$$

This decay heat curve fit was then integrated using Mathematica. The Mathematica routines employed a quadratic curve fit to the ANS data. The integral of this quadratic expression is:

$$Eth(t) = -118.04 + (0.0122)t - (9.69 \times 10^{-8})t^2 + (4.05 \times 10^{-13})t^3$$

This integral when multiplied by the reference plant's nominal power is the energy released to the coolant from the time the accident was initiated. When this expression is equated to the total energy required to uncover the core plus the energy released from the time required to deplete the CFT & RWST it can be solved for the core "uncovery" time. The results of these manipulations is that the core "uncovery" time is about 4 hours.

G.1.4 Summary

The results of this section provides a <u>very conservative estimate</u> of the time required for core "uncovery". The estimate provided by this section indicates that the operators will have more than 4 hours in order to identify and isolate a small ISLOCA HPI break.

The inclusion of more realistic calculations in an ISLOCA evaluation may be necessary for proper quantification of the event trees. The calculations are required if it is determined that the core uncovery times strongly influence the quantification of the event trees. The refined calculations should be performed for the dominant sequences identified in the event tree quantification using the reference plant's LOCA procedures. These supporting calculations could take the form of detailed RELAP5 calculations, MELCOR or simulator trials. These refinements to the calculational methodology will provide a better estimate of the time available for operator actions to occur.

There are two assumptions used in the ISLOCA HPI analysis that make a significant difference in the calculated core uncovery times. These two assumptions are in the use of a 2.5 inch hole in the RCS HPI injection line and the inventory of water assumed in the RWST.

The first assumption, (i.e. the 2.5 in. hole size), over predicts the flow rate. This flow rate over prediction occurs since the HPI piping of the reference plant has thermal sleeves in the lines that are 1.5 inches in diameter. These sleeves can limit the coolant flow rate to less than that predicted in this analysis. The coolant flow is also limited by the pressure drop through the HPI piping to the location of the break. The second major assumption that influences the core uncovery time was the amount of coolant in the RWST.

The core "uncovery" time for this ISLOCA HPI sequence is extended to 11 hours when:

a). the reference plant's technical specifications are used for the RWST inventory and,
b.) the thermal sleeves in the HPI nozzles are included.

The above scoping results indicate that the operations crew have a significant time period to identify and isolate an ISLOCA HPI sequence. It also appears that more refined calculations will provide a substantial time margin for the operations crew to isolate the failure before the onset of core damage occurs.

G.2 BOUNDING CALCULATIONS FOR THE LARGE BREAK ISLOCA

G.2.1 Introduction

This section documents the core uncovery time calculations for a large break interfacing systems LOCA. These calculations are used to estimate the minimum time required for the onset of core uncovery. The core uncovery time estimate includes a boil down time after ECCS pump suction was lost from the RWST & CFT volumes.

The core uncovery time is defined in this analysis as the time at which the collapsed vessel liquid level drops below the top of the core's active fuel. This definition is not the true core uncovery time due to the presence of a void fraction distribution in the core. The use of the collapsed water level at the top of the core in these assessments provides a conservative indication of the operator response times available before core damage occurs.

G.2.2 Time to Empty the RWST & CFT

The ISLOCA large break was assumed to occur outside of the containment. The break occurs on one of the LPI injection lines. The LPI lines are 10.-in. pipe immediately outside of the containment. As a consequence, the break was modeled as a 10.-in. diameter leak. No credit was given for form losses, wall friction, or other pressure drop effects that may reduce the break mass flow rate. These break assumptions were incorporated into a simplified five volume RELAP5 model. This model was used to estimate the time required to empty the RWST & CFT. The primary system's ECCS and the break were explicitly modeled. The assumed break configuration leads to a shorter and more conservative time estimate of when the RWST & CFT empty.

Several additional assumptions were incorporated into the RELAP5 model to ensure a conservative time estimate to empty the RWST:

- The ECCS water supply was limited to 400,000 gal.
- Auxiliary feedwater was available and it was assumed that the steam generators were depressurized to enhance primary to secondary heat transfer. Thus it was assumed all core decay energy is removed by the steam generators. Or equivalently,

no stored energy or decay heat was included in the RELAP5 model. This resulted in a lower primary system pressure and a higher ECCS mass flow rate.

- It is assumed that the break flow rate will stabilize near the LPI shutoff head and the time averaged break mass flow rate will balance with the time averaged ECCS mass flow rate. This is typical of many small break LOCA's. The steady-state ECCS mass flow rate was modeled by equating it with the steady-state choked break mass flow rate at the LPI activation pressure. This time averaged balance accounted for short intermittent periods when the LPI would activate. Continuous LPI operation was not possible because the total LPI mass flow rate was significantly larger than the break mass flow rate.

- The CTF's are available and the drain out time will be estimated by dividing the total break mass flow rate by the total volume of the CTF liquid inventory. The CTF's were not in the RELAP5 model and this time estimate was done separately.

- It is assumed that the transient is initiated at 100% power conditions and that there is no significant delay for reactor scram.

- Two HPI, 1-LPI, and 2-charging trains were modeled to refill the primary system.

Figures G-3 and G-4 describe the primary pressure response and the ECCS/break flow rates for the simplified RELAP5 simulation. These figures indicate that after about one hour the primary system pressure will stabilize near 20 psia. The ECCS/break flow rates will be approximately 850 lbs/sec. Under these conditions the break mass flow is not chocked. This mass flow rate of 850 lbs/sec will deplete the RWST volume of 400,000 gallons in about 1.1 hours.

Once the CFTs are activated it is assumed that they empty at an average rate equal to the ECCS/break flow. The ECCS/break mass flow rate of 850 lbs/second provides the time estimate for the draining of the CFTs. The CFT's volume is then emptied in about 0.04 hours with this break flow. The total time to drain the RWST's and CFT's coolant inventory is then about 1.1 hours.

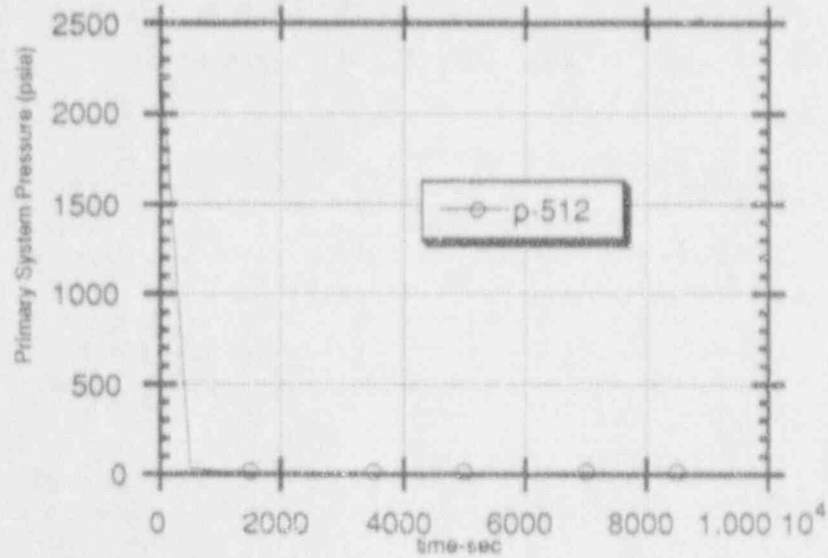

Figure G-3. Small break ISLOCA showing primary system pressure

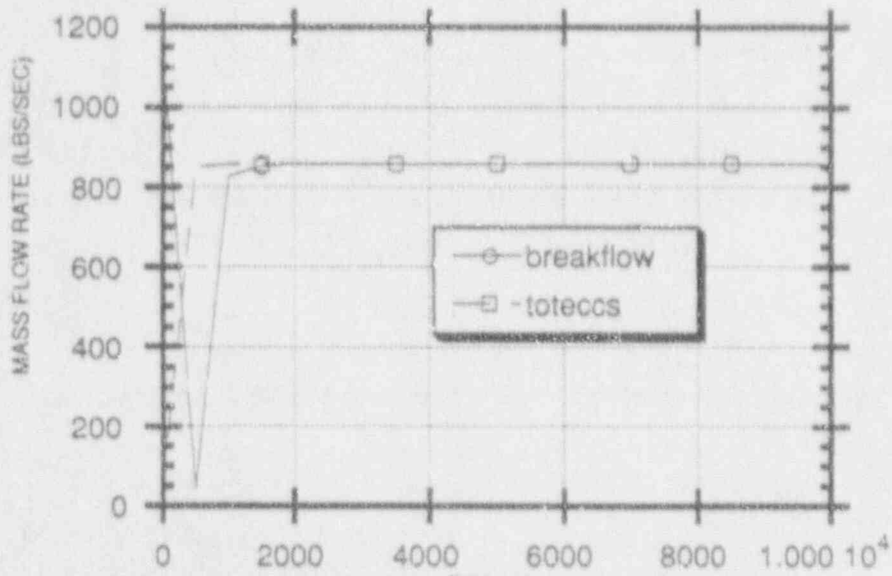

Figure G-4. Small break ISLOCA showing mass flow rates

The are several sources of uncertainty in the estimate of the time for the RWST and CFT to empty. These uncertainties are listed as follows:

- The fact that the primary system pressure is a nonconstant nonlinear function of time.
- The HPI and charging mass flow rates are nonlinear functions of the primary system pressure.
- Primary to secondary heat transfer may greatly alter outcome depending whether the operator decided to depressurize the steam generators.
- Flooding out the auxiliary building may disrupt or destroy the ECCS pumping equipment before the RWST tank is empty.
- Intrinsic uncertainties in the RELAP5 critical flow model.

The proper modeling of the primary system pressure would necessitate modeling primary to secondary heat transfer as well as stored energy and core decay heat in RELAP5. These RELAP5 models were not incorporated in the analysis presented in this section. These models were not incorporated for the expressed purpose of minimizing the time to empty the RWST and CFT.

G.2.3 Core's Upper Plenum Boil Down Time Calculation

This section will detail the procedure used to estimate the time required for the core's upper plenum coolant to boil off. This calculation is based on the assumption that no additional ECCS coolant is injected into the primary system after 1.1 hr.

To estimate the time for the core's upper coolant plenum to boil off the following assumptions were made:

1. No liquid in the loop regions is available to be heated by the core. The liquid in the loop regions is assumed to either exit the break or reside in the loop seals. This assumption is conservative because after the RWST has drained, some liquid that is in the primary loops will drain back into the vessel region. It is not possible to make an estimate of how much liquid would actually drain into the vessel unless a full-systems calculation is completed.

2. It was assumed that the vessel upper head is completely drained by the time the RWST has emptied. This assumption is reasonable since some small break PWR LOCA scenarios lead to vessel upper head voiding.

3. It was assumed that any sensible or latent heat added to the vessel liquid will result in no significant repressurization of the primary system and therefore the total integrated core decay power goes to initiate core boil off.

4. It was assumed that the remaining liquid in the vessel available for boil off to be at the bulk subcooled conditions of 100°F at a pressure of 20 psia. This assumption is not based on rigorous quantitative arguments.

Two parameters must now be calculated. These parameters are required to determine the time interval required for the collapsed coolant level to reach the top of the active fuel. The two parameters are the total energy required to reduce the coolant level to the top of the active fuel and the integral of the power generation rate of the core. The time interval is determined by integration of the core's power level until the core's energy output equals the energy required to reduce the coolant level.

The total energy required for the reduction of the coolant level is composed of two contributions. The first contribution is the energy required to raise the temperature of the reactor vessel's total coolant inventory to saturation at 20 psig. The second contribution is the energy required to vaporize the coolant remaining in the reactor vessel above the active fuel region. The calculation of the minimum energy required to raise the coolant's temperature to saturation requires the following information:

- The enthalpy of the subcooled vessel liquid at 20 psia & 100°F,
- The saturation enthalpy of the vessel liquid at 20 psia.
- The bulk density of the liquid at 100°F.
- The coolant volume of the reactor vessel minus the upper head volume.

The appropriate coolant volume values for the B&W reference plant were utilized in the calculation. The energy required to raise the coolant remaining in the vessel after RWST & CFT depletion to saturation is 2.45E10 joules.

The calculation of the second energy contribution required to estimate the time for core uncovery is described next. The calculation begins with the liquid in the reactor vessel at bulk boiling conditions. The energy added to the vessel's liquid is assumed to result in liquid vaporization. This is based on the assumption that no liquid in the loop regions will drain into the vessel. All the liquid above the core is assumed to be turned into steam. The energy needed to vaporize the liquid region above the active core region is equal to energy required to vaporize the saturated liquid in the vessel upper plenum. The calculation of the vaporization energy requires the following information:

1. The upper plenum's volume of saturated liquid,
2. The latent heat of vaporization,
3. The saturation density of the liquid.

With this information a total energy of 6.03E10 joules is required to vaporize the upper plenum water. The vaporization of this water results in the collapsed water level dropping to the top of the active fuel.

The total energy required to reduce the reactor vessel's coolant level to the top of the active fuel is the contribution of the sum of the subcooled and vaporization energies. These two energy values sum to 8.48E10 joules for the B&W reference plant.

The minimum time required for the core to uncover is determined by integrating the reactor's decay heat power curve from the time the RWST & CFT emptied to the point in where the integral equals 8.48E10 joules. It was assumed in this analysis that the reactor scram started at the same time the ECLS flow was initiated by the ISLOCA transient. This assumption is conservative since a delay in the core scram allows the primary system to remain pressurized and as a result prevents or reduces the ECCS coolant flow.

The normalized ANS core decay power curve was used to estimate the reactor's decay heat as a function of time. This decay heat curve was fit to a quadratic polynomial by the Mathematica curve fitting routines. This curve fit was then integrated between the time the RWST & CFT coolant volumes were depleted and the unknown core uncovery time. This polynomial equation was normalized to the initial core power. The time required to achieved a collapsed water level equal to the top of the active fuel was determined by using the Mathematica algebraic/numerical routines.

The data used to develop the decay heat curve is the normalized decay curve from the standard ANS decay model. The data points in the following table were used to develop the curve fit used in the analysis.

The integral of core's decay heat curve from the time the RWST & CFT empties is required to estimate the time when the core's coolant temperature reaches saturation and the time the core "uncovers". A regression fit to the

Table G-3. ANS Normalized Decay Heat Curve for the H.B. Robinson Plant
(From RELAP5)

Time Seconds	Normalized Power
100.	0.0331
400.	0.0235
800.	0.0196
1000.	0.0185
2000.	0.0157
4000	0.0128
8000.	0.0105
10000.	0.00965
20000.	0.00795
60000.	0.00566
100000.	0.00475

H.B. Robinson decay heat curve was utilized to provide the decay heat values required. The regression fit of the H.B. Robinson's ANS data gives:

$$Pd(t) = 0.012 - 1.939 * 10^{-7} * t + 1.215 * 10 \quad * t^2$$

This decay heat curve fit was then integrated using Mathematica. The Mathematica routines employed a quadratic curve fit to the ANS data. The integral of this quadratic expression is:

$$Eth(t) = -118.04 + (0.0122)t - (9.69 \times 10^{-8})t^2 + (4.05 \times 10^{-13})t^3$$

This integral when multiplied by the reference plant's nominal power is the energy released to the coolant from the time the accident was initiated. When this expression is equated to the total energy required to uncover the core plus the energy released from the time required to deplete the CFT & RWST it can be solved for the core "uncovery" time. The results of these manipulations is that the core uncovery time is about 1.9 hours.

G.2.4 Summary

The results of this section provide a <u>very conservative estimate</u> of the time for core "uncovery". The estimate provided by this section indicates that the operators will have more than 2 hours in order to identify and isolate a large ISLOCA LPI break.

The inclusion of more realistic calculations in an ISLOCA evaluation may be necessary for proper quantification of the event trees. The calculations are required if it is determined that the core uncovery times strongly influence the quantification of the event trees. The refined calculations should be performed for the dominant sequences identified in the event tree quantification using the reference plant's LOCA procedures. These supporting calculations could take the form of detailed RELAP5 calculations, MELCOR or simulator trials. These refinements to the calculational methodology will provide a better estimate of the time available for operator actions to occur.

There are two assumptions used in the large break ISLOCA LPI analysis that make a significant difference in the calculated core uncovery times.

These two assumptions are in the use of a 10 inch hole in the RCS LPI injection line and the inventory of water assumed in the RWST.

The first assumption, (i.e. the 10 in. hole size), over predicts the flow rate. This over prediction occurs since the LPI piping of the reference plant has an internal diameter of 8.5 inches. The coolant flow is also limited by the pressure drop through the LPI piping to the location of the break. The second major assumption that influences the core uncovery time was the amount of coolant in the RWST.

The core "uncovery" time for this ISLOCA LPI sequence is extended to 3.3 hours when:

a). the reference plant's technical specifications are used for the RWST inventory and,
b.) the correct flow diameter is used for the LPI lines.

The above scoping results indicate that the operations crew have a significant time period to identify and isolate a large break ISLOCA LPI sequence. It also appears that more refined calculations will provide a substantial time margin for the operations crew to isolate the failure before the onset of core damage occurs.

Appendix H

System Rupture Probability Calculations Using EVENTRE

J. A. Schroeder

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FIGURES

TABLES

LISTINGS

APPENDIX H

SYSTEM RUPTURE PROBABILITY CALCULATIONS USING EVNTRE

This appendix describes the quantification of the interfacing system failure events in the event trees described in Appendix D. The failure events occur in the Makeup & Purification System tree (Figure D-2), the High-Pressure Injection System tree (Figure D-4), and the DHR Letdown tree (Figure D-6). The Makeup & Purification System and the High-Pressure Injection System failure events are identical. In these two trees the failure probabilities and the components affected are conditional on failure of valve HP-23. If HP-23 does not fail, the resulting sequences are identified as HP2A-1. If HP-23 fails, they are identified as HP2A-2.

In the discussions that follow, the method used to obtain the system failure probabilities is described in general terms. Then the method is applied to both the high-pressure system scenarios (the HP2A scenarios) and the DHR Letdown Scenario. In each discussion, system pressure capacities and resulting system failure mode probabilities are presented.

H.1. Modeling Approach

The basic approach for simulating the performance of low-pressure rated equipment exposed to a high-pressure internal environment consists of building an event tree model that questions component failure, and failure mode. These event tree models are built for each piece of equipment in the low-pressure rated system. The system pressure is assumed to be either normally distributed about a specified mean value having a specified standard deviation (for the HP2A-1 and HP2A-2 scenarios) or is uniformly distributed between specified values (DHR Letdown scenario). The component failure probabilities are described with lognormal distributions having specified median failure pressures and standard deviations.[H-1] Each question in the event tree is answered by comparing a random sample from the component failure pressure distribution with a random sample from the system pressure distribution. If the sampled system pressure is greater than the sampled component failure pressure, the component fails. Each component in the low-pressure rated system is evaluated in this manner until all components have been examined. The failure mode of each component is evaluated based on the ratio of the system pressure to component failure pressure. This process is repeated for 10,000 samples (or observations) in the Monte Carlo simulation. Once the simulation is completed, the results are binned and estimates regarding the relative frequency of equipment failures can be made.

These calculations were performed using the EVNTRE generalized event tree processor,[H-2] and its associated post processor, PSTEVNT.[H-3] EVNTRE allows the user to define parameters that can be manipulated by the code, or by user defined functions. In this case, the parameters were the log of the system pressure and the log component failure pressures. A user function was developed that assigned failure probabilities (either 1.0 or 0.0) and calculated pressure ratios based on these parameters for each sample evaluation of the tree. The results were binned, written to a post processor file, and then aggregate failure probabilities were determined using PSTEVNT.

The failure pressures used in the simulation were developed in an independent structural analysis by Impell Corporation. Not only were failure pressures calculated, but also leak rates and areas as well. In this respect, flanges exhibit somewhat unique behavior because there are actually two failure pressures of interest. First, is the estimated Gross Leak Pressure (GLP), which is the pressure where a measurable leak area appears. At lower pressures, leakage around the gasket is possible but at very small rates (measured in mg/sec). Once the GLP is exceeded, the bolts in the flange begin to stretch (elastically) and the flange surfaces begin to separate. At some higher pressure P_o, the bolts begin to yield plastically. At this point, large leak areas begin to appear with corresponding large leak rates. These three regimes, (below GLP, between GLP and P_o, and greater than P_o) are associated with three sizes of leaks, namely spray leaks, small leaks and large leaks. Each of these regimes was evaluated in the event tree model by determining the ratio of system pressure to GLP, and assigning an appropriate failure mode (no leak, spray, small leak, large leak) for each component in the system.

The binning scheme used to identify the system failure modes requires some explanation. The binning scheme assumes that if a large leak occurs in any path through the tree, the end state for that path is included in the large leak bin. The large leak bin is therefore the union of all large break events in the tree. It will also include end states that have small leak, spray, and no leak events. The total leak area associated with the bin could therefore be many times that of a single large leak.

In collecting the end states for the small leak bin, pathways that include large leaks are excluded, and the union is formed of all remaining end states that include at least one small leak event. Some of these paths will also contain spray and no leak events.

In collecting end states for the spray bin, an end state will only be assigned to the bin if it contains no small leaks, and no large leaks. Once these end states are excluded, the spray bin will consist of the union of the remaining spray leakage events.

The no leak bin is collected, on a system basis, as the union of all no leak pathways through the tree.

H.2. MOV HP-2A Scenarios

If the RCS pressure isolation boundary is breached, high-pressure reactor coolant is allowed to enter the low-pressure rated systems. There are two opportunities for overpressurizing and rupturing a portion of the low-pressure rated pipe. The first, which is labeled "HP2A-1," involves the backflow of high-pressure water through a 3-in. recirculation line to the borated water storage tank (BWST). Two factors exert a large influence on the pressure imposed on the low-pressure rated pipe: (a) the presence of a restricting orifice and (b) the fact that the pipe empties into the BWST and is not dead-ended. These two factors influence the maximum pressure generated in the low-pressure pipe, which is approximately 650 psi (estimated from RELAP code calculations). However, for the second overpressure scenario, called "HP2A-2," the flow does not pass through the restricting orifice and is dead-ended. This scenario requires the additional failure of the HPI pump discharge check valve (HP-23 or HP-22), to allow backflow through the pump and into the pump suction line. In this case, the pressure in the pipe would likely reach 2,000 psi.

HP2A-1

The HP2A-1 scenario involves the backflow of high-pressure water through a 3-in. recirculation line to the BWST. The pressure in the system will be reduced by a flow restricting orifice. The result is a system pressure of approximately 650 psia. The data in Table H-1 were used in the event tree. The component failure distributions are all lognormal.

Because the analysis of this scenario was performed by Monte Carlo evaluation of the system event tree, sample distributions were created based on the component failure data in Table H-1. Sample vectors for this scenario were created with the program provided in Listing 1. The program is not complete in that the essential subroutines are not listed, however, the complete listing of all subroutines is provided with the data for the DHR Letdown Scenario.

The event tree for this scenario will be evaluated using the EVNTRE computer code. EVNTRE requires a number of data files to describe the event

Table H-1. HP2A-1 scenario component failure data

Component	Description	Med. Fail Press	Ln Mean	Ln Std Dev	A P=.001	B P=.999
3"-HCC-91	Pipe, sch. 10S	2712	7.905	0.360	888.4	8278.7
FE-HP4	3" 150 psi flow el.	955	6.862	0.040	843.6	1081.1
HP-33	3" swing check valve	5507	8.614			

Mean initial pressure = 650, std. dev. = 50 psi

tree model. The first is a keyword file (see Listing 2). Following the keyword file is the event tree description (Listing 3), the sample definition file (Listing 4), and the binning input data (Listing 5). The EVNTRE output for the base case (see Table H-1) is provided in Listing 6. A sensitivity study with the pipe failure log standard deviation decreased from 0.36 to 0.10 produced the results in Listing 7.

To check the validity of the data used in the Monte Carlo evaluation, a Latin Hypercube evaluation was also performed. The data for this evaluation were produced using the LHS program from Sandia National Laboratory.[H-4] Because of the greater computational efficiency of the LHS method, only 1000 samples were generated, as opposed to 10,000 for the Monte Carlo evaluation. The results from the LHS evaluation of the tree are provided in Listing 8 and agree reasonably well with the Monte Carlo results.

HP2A-2

This scenario is different from the HP2A-1 scenario in that the flow does not pass through a restricting orifice and is dead-ended. This scenario is likely to result in system pressures as high as 2000 psia. The data in Table H-2 were used in the event tree. The component failure distributions are all lognormal.

Table H-2. HP2A-2 scenario component failure data

Component	Description	Med. Fail Press	Ln Mean	Ln Std Dev	A P=.001	B P=.999
P58-2	HPI Pump 1-2	2250	7.719	0.250	1036.6	4883.8
6"-GCB-4	Pipe, sch. 10S	1644	7.405	0.360	538.6	5018.5
6GCB4a	6" 300 psi flange-a	2362	7.767	0.120	1628.3	3426.4
6GCB4b	6" 300 psi flange-b	2362	7.767	0.120	1628.3	3426.4
HP-13	6" 300 psi LM valve	2170	7.682	0.250	999.7	4710.2
4"-GCB-2	Pipe, sch. 10S	2075	7.638	0.360	679.7	6334.2
4"-GCB-11	Pipe, sch. 10S	2075	7.638	0.360	679.7	6334.2

Mean initial pressure = 2000, std. dev. = 50 psi

The EVNTRE output was saved in a post-processor file for later evaluation with PSTEVNT. The keyword file and rebin data are provided in Listings 20 and 21. The output from PSTEVNT for the base case and sensitivity case is provided in Listings 22 and 23. Table H-3 summarizes the component and system failure probabilities for the HP2A-2 scenario.

Table H-3. HP2A-2 failure mode probabilities

		Failure Mode Probability			
Component	Description	NoLeak	Spray	Small	Large
P58-2	HPI Pump 1-2	2.45E-01	4.35E-01	3.20E-01	
6"-GCB-4	Pipe, sch. 10S	2.90E-01			7.10E-01
6GCB4a	6" 300 psi flange-a	9.13E-01			8.67E-02
6GCB4b	6" 300 psi flange-b	9.15E-01			8.53E-02
HP-13	6" 300 psi LM valve	6.31E-01		3.69E-01	
4"-GCB-2	Pipe, sch. 10S	5.44E-01			4.56E-01
4"-GCB-11	Pipe, sch. 10S	5.49E-01			4.51E-01
Total		1.18E-02	2.43E-02	4.01E-02	9.23E-01

Mean initial pressure = 2000, std. dev. = 50 psi

H.3. DHR Letdown Scenarios

When the plant operates in a shutdown mode (i.e., modes 4 or 5), the DHR system is used for removing core decay heat. It operates via a 12-in. pipe connected to one of the RCS hot legs and is isolated by two 12-in. motor-operated gate valves in series (DH-12 and DH-11). There is also an 8-in. line that bypasses DH-11 and DH-12 that has two locally-manually operated gate valves in series.

There are two scenarios that relate to possible ISLOCA sequences: (a) the premature opening of the DHR letdown line while the plant is in the process of shutting down but not yet in the operating range of the DHR system (i.e., RCS above approximately 300 psi and 300°F) and (b) a plant startup with the DHR letdown line left open while the RCS heats up above the operating range of the DHR system. In both situations, the DHR system is exposed to high-pressure reactor coolant that could possibly result in the rupture of some low-pressure rated components.

The component failure data (illustrated in Table H-5) were treated the same way in this scenario as in the previous HP2A scenarios. However, the reactor system pressure varies over a wide range (2000 psia down to 300 psia) during the course of the shutdown. Also, pressures at various components in the system were shown (using RELAP calculations) to vary significantly from the reactor system pressures. To treat these factors two system pressure parameters were used in evaluating the system event tree. The pressure distributions were derived by assuming the RCS pressure is uniformly distributed over the range of 300 psia to 2000 psia, and that the pressure at any point in the system could be obtained as a simple function of RCS pressure.

Because the analysis of this scenario was performed by Monte Carlo evaluation of the system event tree, sample distributions were created based on the component failure data in Table H-4. Sample vectors for this scenario were created with the program provided in Listing 24.

The event tree for this scenario was evaluated using EVNTRE. EVNTRE requires a number of data files to describe the event tree model. The first

Table H-4. DHR letdown scenario component failure data

Component	Description	Med. Fail Press	Ln Mean	Ln Std Dev	A P=.001	B P=.999
DH-4849						
12"-GCB-7	Pipe, sch. 20	1660	7.415	0.360	543.8	5067.3
DH-2734						
DH-1517	12" MOGV, 300 psi	1704	7.441	0.200	916.7	3167.6
18"-GCB-8	Pipe, sch. 20	1488	7.305	0.360	487.5	4542.3
DH-2733	18" MOGV, 300 psi	2277	7.731	0.200	1224.9	4232.8
18"-HCB-1	Pipe, sch. 10S	843	6.737	0.360	276.2	2573.4
14"-HCB-1	Pipe, sch. 10S	1090	6.994	0.360	357.1	3327.4
DH-81	14" SwCV, 150 psi	1445	7.276	0.200	777.3	2686.2
12-GCB-8	Pipe, sch. 20	1660	7.415	0.360	543.8	5067.3
12GCBa	Flange, 300 psi	2250	7.719	0.120	1551.0	3263.9
12GCBb	Flange, 300 psi	2250	7.719	0.120	1551.0	3263.9
12GCBc	Flange, 300 psi	2250	7.719	0.120	1551.0	3263.9
P42-1	DHR pump 1-1	2250	7.719	0.200	1210.4	4182.6
10"-GCB-1	Pipe, sch. 20	1984	7.593	0.360	649.9	6056.4
10GCB1a	10" flange, 300 psi	2485	7.818	0.120	1713.0	3604.8
DH-43	10" SwCV, 300 psi	2016	7.609	0.200	1084.5	3747.6
DH-45	10" HWGV, 300 psi	2170	7.682	0.200	1167.3	4033.9
E271T	DHR Hx tube sht	432	6.068	0.120	297.8	626.7
E271P	DHR Hx plastic col	1030	6.937	0.230	504.9	2101.3
E271C	DHR Hx cyl. rupt.	1630	7.396	0.270	705.8	3764.4
E271A	DHR Hx asym. hd. bkl	2030	7.616	0.230	995.0	4141.9
E271a	10" out-f, 300 psi	2485	7.818	0.120	1713.0	3604.8
E271b	10" in-f, 300 psi	2485	7.818	0.120	1713.0	3604.8
6"-GCB-10	Pipe, sch. 10S	1585	7.368	0.360	519.2	4838.4
10"-GCB-10	Pipe, sch. 20	1984	7.593	0.360	649.9	6056.4
8"-GCB-10	Pipe, sch. 20	2503	7.825	0.360	820.0	7640.7
DH-128	8" SwCV, 300 psi	1242	7.124	0.200	668.1	2308.8
4"-GCB-2	Pipe, sch. 10S	2075	7.638	0.360	679.7	6334.2
FE-DH2B	10" FE, 300 psi	2485	7.818	0.120	1713.0	3604.8

Median initial RCS pressure = 1250 (uniform between 300 and 2200 psi)
Median system pressure at DH-4849 = 1188.
Median system pressure at DH-2734 = 818.

is a keyword file (see Listing 25). Following the keyword file is the event tree description (Listing 26), the sample definition file (Listing 27), and the binning input data (Listing 28). The EVNTRE output for the base case (see Table H-4) is provided in Listing 29.

The EVNTRE output was saved in a post-processor file for later evaluation by PSTEVNT. The keyword file and rebin data are provided in Listings 30 and 31. The output from PSTEVNT for the base case (pipe failure standard deviation of 0.36) and sensitivity case (pipe failure standard deviation of 0.10) is provided in Listings 32 and 33. Table H-5 summarizes the component and system failure mode probabilities from these files.

Cumulative system rupture mode distributions were also obtained. These were produced by making separate fixed pressure Monte Carlo evaluations of the system event trees over the range of system pressures expected during cooldown or startup. The data resulting from these runs are summarized by the binning reports provided in Listings 34 and 35. Listing 34 provides the result for the base case, and Listing 35 provides the result with the narrower pipe failure distribution. The results from these two studies are shown in Figures H-1 and H-2.

Table H-5. DHR letdown failure moda probabilities

Component	Description	Failure Mode Probability: NoLeak	Spray	Small	Large
DH-4849					
12"-GCB-7	Pipe, sch. 20	7.45E-01			2.55E-01
DH-2734					
DH-1517	12" MOGV, 300 psi	9.87E-01		1.30E-02	
18"-GCB-8	Pipe, sch. 20	8.93E-01			1.07E-01
DH-2733	18" MOGV, 300 psi	9.99E-01		5.00E-04	
18"-HCB-1	Pipe, sch. 10S	5.53E-01			4 [illegible]7E-01
14"-HCB-1	Pipe, sch. 10S	7.31E-01			2.70E-01
DH-81	14" SwCV, 150 psi	9.33E-01		6.75E-02	
12-GCB-8	Pipe, sch. 20	9.29E-02			7.12E-02
12GCBa	Flange, 300 psi	1.00E-00			
12GCBb	Flange, 300 psi	1.00E-00			
12GCBc	Flange, 300 psi	1.00E-00			
P42-1	DHR pump 1-1	9.99E-01		3.00E-04	
10"-GCB-1	Pipe, sch. 20	9.69E-01		3.15E-02	
10GCB1a	10" flange, 300 psi	1.00E-00			
DH-43	10" SwCV, 300 psi	9.98E-01		2.50E-03	
DH-45	10" HWGV, 300 psi	9.99E-01		9.00E-04	
E271T	DHR Hx tube sht	1.45E-01		4.27E-01	4.27E-01
E271P	DHR Hx plastic col	9.40E-01			5.99E-02
E271C	DHR Hx cyl. rupt.	9.55E-01			4.48E-02
E271A	DHR Hx asym. hd. bkl	9.99E-01		9.20E-04	
E271a	10" out-f, 300 psi	1.00E-00			
E271b	10" in-f, 300 psi	1.00E-00			
6"-GCB-10	Pipe, sch. 10S	9.18E-01		8.22E-02	
10"-GCB-10	Pipe, sch. 20	9.71E-01			2.95E-02
8"-GCB-10	Pipe, sch. 20	9.93E-01			7.30E-03
DH-128	8" SwCV, 300 psi	8.58E-01		1.42E-01	
4"-GCB-2	Pipe, sch. 10S	9.78E-01			2.20E-02
FE-DH2B	10" FE, 300 psi	1.00E-00			
Total		1.42E-01		2.51E-01	6.06E-01

Median initial RCS pressure = 1250 (uniform between 300 and 2200 psi)
Median system pressure at DH-4849 = 1188.
Median system pressure at DH-2734 = 318.

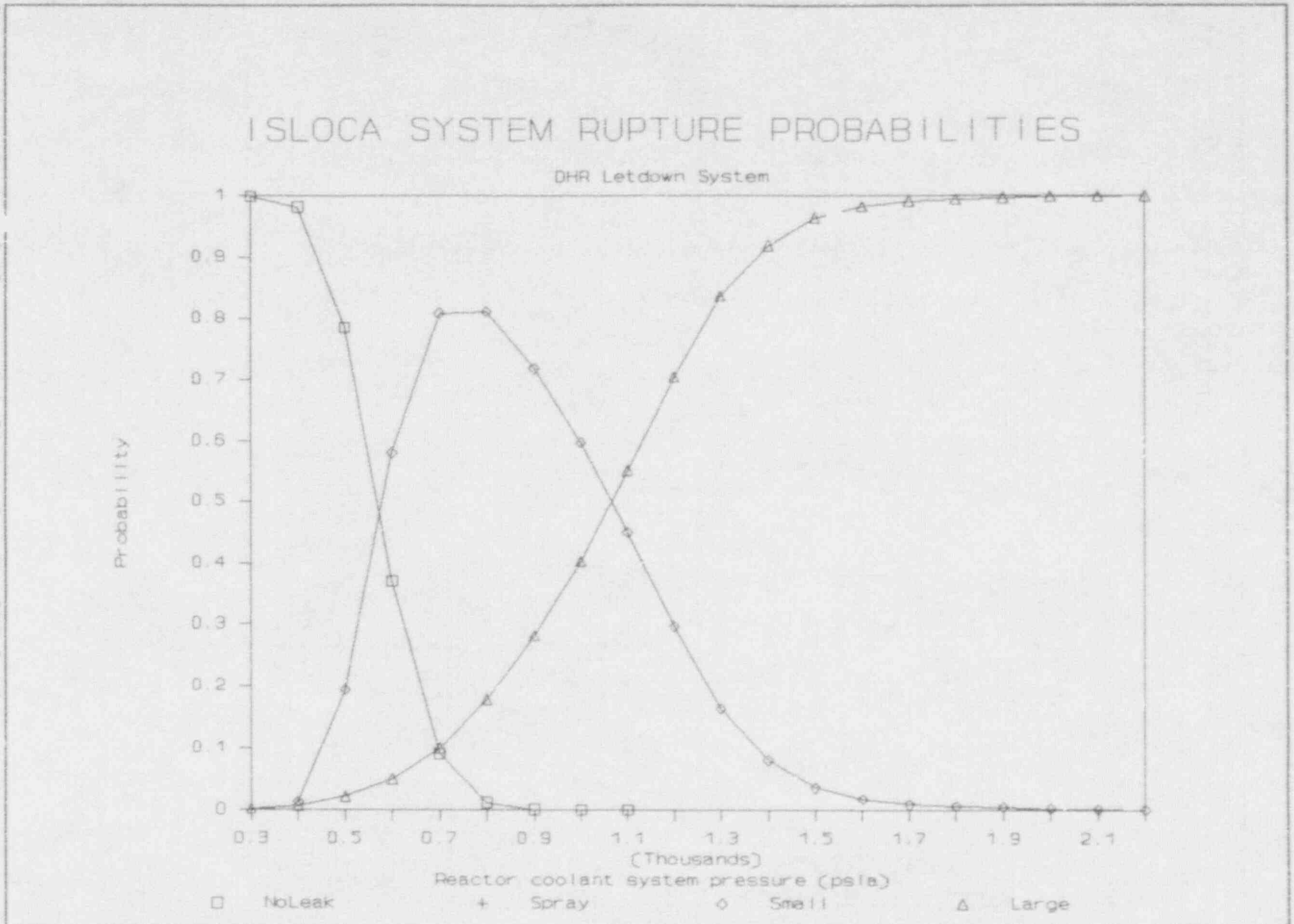

Figure H-1. System failure probabilities for the DHR letdown system

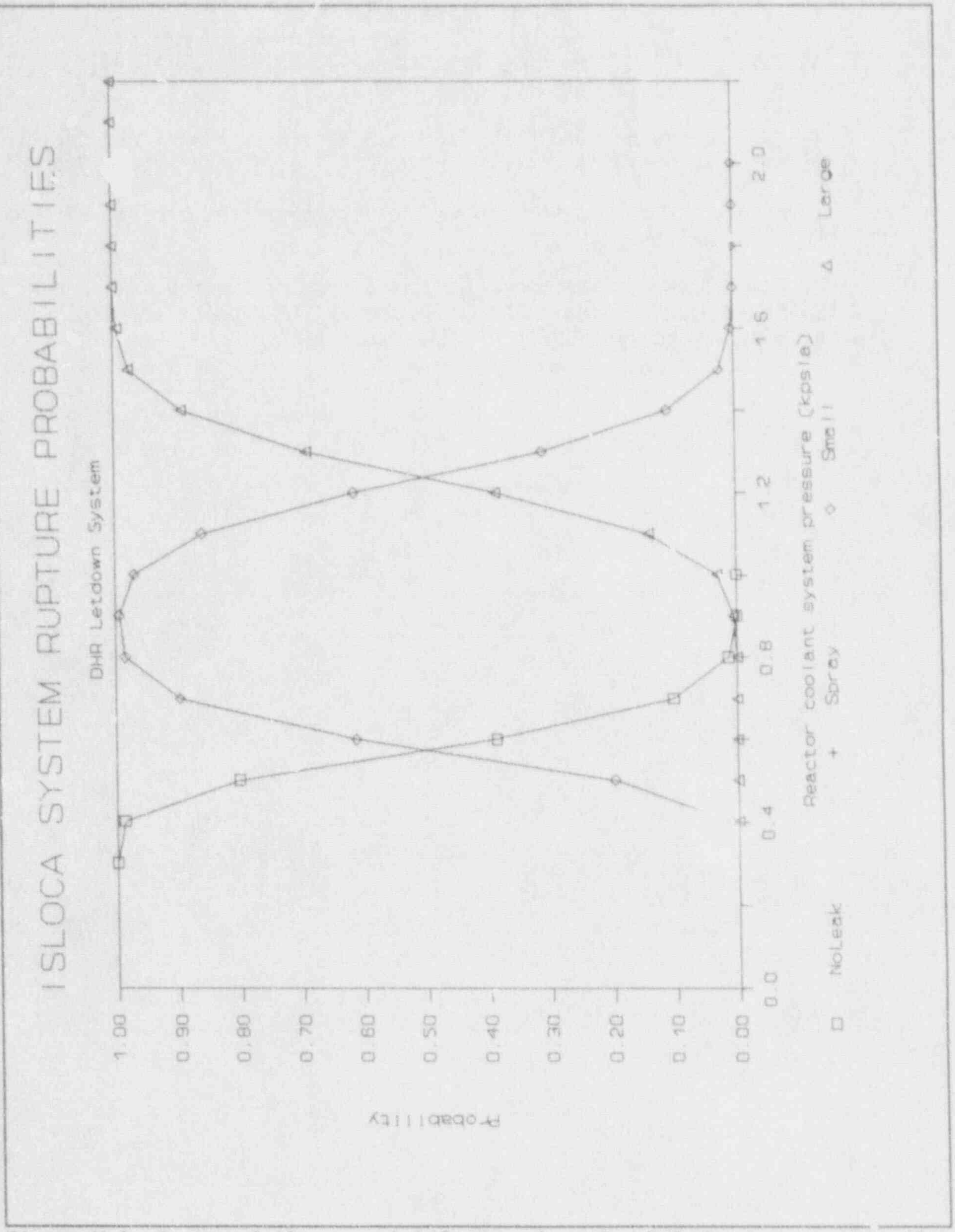

Figure H-2. System fai'ure probabilities for the DHR letdown system (log standard deviat'on of 0.10)

H.4. References

H-1. D. A. Wesley et al., *Pressure-Dependent Fragilities for Piping Components*, NUREG/CR-5603, October 1990.

H-2. J. Michael Griesmeyer and L. N. Smith, *A Reference Manual for the Event Progression Analysis Code (EVNTRE)*, NUREG/CR-5174, September 1989.

H-3. S. J. Higgins, *A User's Manual for the Postprocessing Program PSTEVNT*, NUREG/CR-5380, November 1989.

H-4. R. L. Iman and M. J. Shortencarier, *A Fortran 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use With Computer Models*, NUREG/CR-3624, March 1984.

LISTING 1

Program Used to Generate Distributions for HP2A-1

The following is a listing of the Fortran 77 program used to generate the Monte Carlo sample data required by the EVNTRE program for evaluation of the HP2A-1 model. Some of the subroutines used by the program are not shown in Listing 1. Listing 24 provides the source code for the missing subroutines.

```
*   PURPOSE:
*
*   This program calculates the normal distributions required for the
*   Monte Carlo evaluation of the ISLOCA HP2A-1 scenario.  Three
*   distributions are required.  The first distribution corresponds to
*   system pressure.  The required output for the first parameter is
*   actually the natural log of the system pressure.  The remaining
*   distributions are log normal, and are described by a log mean and
*   logarithmic std. dev.  The output is written in the format required by
*   the EVNTRE program.
*
*   INPUT:
*
*   The input is hardwired into the code, except for the value of the
*   required random seed, which is either read from the data file 'RANS.DAT'
*   or input by the user.  The option is provided by user dialog at run time.
*
*   OUTPUT:
*
*   The output is the required normal distributions, and is written to file
*   'MCARLO.DAT'.  The data are in the format required by the EVNTRE code for
*   use as sample data.  The last value of the random seed is also written to
*   the file 'RANS.DAT' for use in the next evaluation.
*
*   FILES:
*
*   -------------------------------------------
*   Unit   Description
*   ----   -----------------------------------
*     5    User input from console
*     6    Program output to console
*    10    Saved value of random seed
*    11    Output for use as an EVNTRE sample file
*   -------------------------------------------
*
*
*   WRITTEN BY:
*
*   John Schroeder 1/11/90
*
      PROGRAM HP2A1

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      CHARACTER*8 FNSAVE
      CHARACTER*3 FN5
      REAL  X1(10000),  X2(10000),  X3(10000)

      DATA IOU5/5/, IOU6/6/, IOSAVE/10/
```

```
      DATA FNSAVE /'RANS.DAT'/, FN5/'CON'/

      OPEN (UNIT=5,FILE='CON')
      OPEN (UNIT=6,FILE='CON')

      CALL SEEDIN(ISEED,IOU5,IOU6,IOSAVE,FNSAVE,FN5)
      WRITE(6,10) ISEED
10    FORMAT(' ISEED =',I15)

      N     = 10000                     ! Pick 10000 Numbers
      NX    = 3                         ! Number of distributions requested
      ISORT = 0                         ! Don't sort the numbers

*     Generate the distribution for the initial system pressure

      AMEAN = 650.                      ! Mean of Normal distribution
      STDEV = 50.                       ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X1,N,ISORT,ISEED,IER,IPOINT)
      DO 15  I=1, N
         X1(I) = ALOG(X1(I))            ! Use natural log of pressure
15    CONTINUE

*     Generate the distribution for failure of HCC-91

      AMEAN = 7.905                     ! Mean of Normal distribution
      STDEV = 0.360                     ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X2,N,ISORT,ISEED,IER,IPOINT)

*     Generate the distribution for failure of HP-4

      AMEAN = 6.862                     ! Mean of Normal distribution
      STDEV = 0.040                     ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X3,N,ISORT,ISEED,IER,IPOINT)

*     Write out distributions

      OPEN(11,FILE='MCARLO.DAT',STATUS='UNKNOWN')
      DO 20, I=1, N
         WRITE(11,25) I, NX,  X1(I),  X2(I),  X3(I)
20    CONTINUE
25    FORMAT(2I10, 5G12.5,/(20X,5G12.5))
      CLOSE(11)

      CALL SEEDOU(ISEED,IOU6,IOSAVE,FNSAVE)
      WRITE(6,10) ISEED

      CLOSE(5)
      CLOSE(6)
      CLOSE(10)

      END
```

LISTING 2

EVNTRE Key Word File for HP2A-1

The following is a listing of the EVNTRE key word file for the HP2A-1 model. The EVNTRE key word file controls the mode of execution, input and output options, and cutoff values used by the program during event tree evaluation.

```
$-- Calculation Control Keywords -----------------------------------------------
$
    MODE 3                      $ Specifies the calculational mode for EVNTRE.
$                                     1 = point estimate
$                                     3 = sampling mode (one vector each eval)
$                                     4 = sampling mode (two vectors each eval)
$
    NOBIN                       $ Turns the binning facility on/off.
$
$
    RUN                         $ Indicates that the tree is to be evaluated
$                                 after the input data has been processed.
$
$   KEEPCUT 1.0E-6              $ Specifies the path frequency below which a
$                                 path is terminated.
$
$-- Input File Specification Keywords ------------------------------------------
$
    TREEIN tree.dat             $ Specifies the input file name for the
$                                 tree definition input file.
$
$   BININ  bin.dat              $ Specifies the input file name for the
$                                 binning and sorting information input file.
$
    SAMDIN mc_pntr.dat          $ Specifies the input file name for the
$                                 sample definition information input file.
$
    SAM1IN mcarlo.dat           $ Specifies the input file name for the
$                                 first set of sample input vectors.
$
$   SAM2IN mcarlo.dat           $ Specifies the input file name for the
$                                 second set of sample input vectors.
$
$-- Report Request Keywords ----------------------------------------------------
$
$   PRTINP                      $ Turns on the annotated echo of input.
$
    STATS                       $ Indicates that a branch and case frequency
$                                 table report will be generated.
$
    PRUNE                       $ Causes unused cases to be dropped from the
$                                 branch and case frequency table.
$
    NWRTBIN                     $ Indicates that a binning result report will
$                                 be generated when the paths through the
$                                 tree are binned.
$
```

```
$    PRTCUT 1.0E-6              $ Specifies the minimum bin frequency required
$                                 to report a bin.
$
$    SAVEBIN                    $ Indicates that a binning results file will
$                                 be generated for post-processing.
$
$-- Output File Specification Keywords -----------------------------------------
$
$    INPOUT  echo.out           $ Specifies the output file name for the
$                                 annotated echo of input.
$
$    BINOUT  bin.out            $ Specifies the output file name for the
$                                 binning result report.
$
     STATOUT mc_freq.out        $ Specifies the output file name for the
$                                 branch and case frequency table.
$
     SAMROUT mc_post.out        $ Specifies the output file name for the
$                                 post-processing file.
$
ENDKEY                          $ Indicates the end of keyword input.
```

LISTING 3

EVNTRE Tree Definition File for HP2A-1

The following is a listing of the EVNTRE event tree definition file for the HP2A-1 model. This file provides the event tree structure and default probability and parameter values for the HP2A-1 model.

```
ISLOCA System Rupture Model -- HP2A-1
     5
    NQ
     1     1.000
       'MC Eval'
  1 What is the pressure in the Interfacing System?
     1 HCC91-HiP
     3         1
           1.000
     3                      $ Point estimates follow:
     1     6.477            $      1Ps'
     2     7.905            $      1Pf' for HCC-91
     3     6.862            $      1Pf' for HP-4
  2 Does HCC-91 fail?  (3", sch 10S, type 304 SS)
     2   HCC91-F HCC91-NoF
     5         1         2
     2         1         2
             1Ps'      1Pf'
         FUN-CMP
           EQUAL           0
       IF 1Ps' .GT. 1Pf' THEN HCC91-F
  3 If segment HCC-91 fails, what is the rupture size?
     2  HCC91-Lg HCC91-NoL
     2         1         2
     2
     1         2
               1
         HCC91-F
           1.000     0.000
       Otherwise
           0.000     1.000
  4 Does HP-4 fail?  (3" flow element, 150 psi rating)
     2     HP4-F   HP4-NoF
     5         1         2
     2         1         3
             1Ps'      1Pf'
         FUN-CMP
           EQUAL           0
       IF 1Ps' .GT. 1Pf' THEN HP4-F
  5 How large is the leak at HP-4?
     4    HP4-Lg    HP4-Sm    HP4-Sp   HP4-NoL
     5         1         2         3         4
     2         1         3
            1Ps'      1Pf'
        FUN-RPSZ
        GETHRESH           3     2.071     1.000     0.750
       Bin Ps'/Pf'
```

LISTING 4

EVNTRE Sample Definition File for HP2A-1

The following is a listing of the EVNTRE sample definition data file for the HP2A-1 model. This file supplies the specifications required to set up the sampling modes for the tree.

```
ISLOCA SAMPLE RUN
10000     1
3
M1,1,1,1,A  M1,1,2,1,A  M1,1,3,1,A
```

LISTING 5

EVNTRE Binning Definition File for HP2A-1

The following is a listing of the EVNTRE binning definition file for the HP2A-1 model. These data specify the logic used to select HP2A-1 event tree end states that are included in each system failure mode bin.

```
ISLOCA Component Failure Binning -- HP2A-1
  1         FSize
  4   4    NoLeak      Spray      Small      Large
  2   1         3          5              $ No leak
                2        * 4
        HCC91-NoL   HP4-NoL
  2   2         3          5              $ Spray
                2        * 3
           HP4-Sp
  2   3         3          5              $ Small leak
                2        * 2
           HP4-Sm
  2   4         3          5              $ Large leak
                1        + 1
        HCC91-NoF     HP4-F
```

LISTING 6

EVNTRE Frequency Output File for HP2A-1

The following is a listing of the EVNTRE frequency output file for the HP2A-1 model. This file contains the individual component failure mode probabilities resulting from the Monte Carlo evaluation of the HP2A-1 model.

```
          TREE ID: ISLOCA System Rupture Model -- HP2A-1
   # OF QUESTIONS:    5
     OBSERVATIONS: 10000
       FOR SERIES: ISLOCA SAMPLE RUN
      SEQUENCE ID:           MC Eval

******** QUESTION:    1  What is the pressure in the Interfacing System?

Q-TYPE/TIMES ASKED:        INDEP. INPUT PROB. INPUT PARM.              10000
          BRANCHES:             HCC91-HiP
                                   1
    REALIZED SPLIT:           1.000E+00

******** QUESTION:    2  Does HCC-91 fail?  (3", sch 10S, type 304 SS)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          10000
          BRANCHES:             HCC91-F   HCC91-NoF
                                   1          2
    REALIZED SPLIT:           1.000E-04 9.999E-01

******** QUESTION:    3  If segment HCC-91 fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                            10000
          BRANCHES:             HCC91-Lg  HCC91-NoL
                                   1          2
    REALIZED SPLIT:           1.000E-04 9.999E-01

                     SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     1.000E-04
      DEPENDENCIES:    2

     REQ. BRANCHES:    1

       DESCRIPTION:  HCC91-F

  CASE/BRANCH SPLIT:          1.000E-04 0.000E+00

 CASE NUMBER/SPLIT:    2     9.999E-01
       DESCRIPTION:             Otherwise

  CASE/BRANCH SPLIT:          0.000E+00 9.999E-01
```

```
******** QUESTION:    4  Does HP-4 fail?  (3" flow element, 150 psi rating)

Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                          10000
          BRANCHES:          HP4-F      HP4-NoF
                               1          2
    REALIZED SPLIT:        0.000E+00 1.000E+00

******** QUESTION:    5  How large is the leak at HP-4?

Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                          10000
          BRANCHES:          HP4-Lg     HP4-Sm     HP4-Sp     HP4-NoL
                               1          2          3          4
    REALIZED SPLIT:        0.000E+00 0.000E+00 1.266E-01 8.734E-01
```

LISTING 7

EVNTRE Frequency Output File for HP2A-1 (Sensitivity Case)

The following is a listing of the EVNTRE frequency output file for the HP2A-1 model. This file contains the individual component failure mode probabilities resulting from the Monte Carlo evaluation of the HP2A-1 model. The failure model probabilities provided in this listing result from using a pipe failure log standard deviation of 0.10 instead of 0.36 as was used to produce Listing 6.

```
           TREE ID:     ISLOCA System Rupture Model -- HP2A-1s
    # OF QUESTIONS:     5
      OBSERVATIONS: 10000
        FOR SERIES: ISLOCA SAMPLE RUN
       SEQUENCE ID:             MC Eval


 ********  QUESTION:    1  What is the pressure in the Interfacing System?
Q-TYPE/TIMES ASKED:        INDEP. INPUT PROB. INPUT PARM.               10000
          BRANCHES:            HCC91-HiP
                                  1
    REALIZED SPLIT:          1.000E+00


 ********  QUESTION:    2  Does HCC-91 fail?  (3", sch 10S, type 304 SS)
Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                           10000
          BRANCHES:            HCC91-F   HCC91-NoF
                                  1          2
    REALIZED SPLIT:          0.000E+00 1.000E+00


 ********  QUESTION:    3  If segment HCC-91 fails, what is the rupture size?
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                             10000
          BRANCHES:            HCC91-Lg  HCC91-NoL
                                  1          2
    REALIZED SPLIT:          0.000E+00 1.000E+00

                        SUMMARY BY CASE

 CASE NUMBER/SPLIT:     2    1.000E+00
       DESCRIPTION:              Otherwise
 CASE/BRANCH SPLIT:          0.000E+00 1.000E+00


 ********  QUESTION:    4  Does HP-4 fail?  (3" flow element, 150 psi rating)
Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                           10000
          BRANCHES:            HP4-F     HP4-NoF
                                  1          2
    REALIZED SPLIT:          0.000E+00 1.000E+00

 ********  QUESTION:    5  How large is the leak at HP-4?
Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                           10000
          BRANCHES:            HP4-Lg    HP4-Sm    HP4-Sp    HP4-NoL
                                  1          2         3         4
    REALIZED SPLIT:          0.000E+00 0.000E+00 1.187E-01 8.813E-01
```

LISTING 8

EVNTRE Frequency Output File for HP2A-1 (Using LHS Data)

The following is a listing of the EVNTRE frequency output file for the HP2A-1 model. This file contains the individual component failure mode probabilities resulting from the LHS evaluation of the HP2A-1 model. The sample data used to produce these results were obtained with the Sandia LHS program instead of the Fortran program in Listing 1.

```
          TREE ID:      ISLOCA System Rupture Model -- HP2A-1
   # OF QUESTIONS:    5
     OBSERVATIONS:  1000
       FOR SERIES: ISLOCA SAMPLE RUN
      SEQUENCE ID:           LHS Eval


 ********  QUESTION:    1  What is the pressure in the Interfacing System?

 Q-TYPE/TIMES ASKED:       INDEP. INPUT PROB. INPUT PARM.              1000
           BRANCHES:          HCC91-HiP
                                  1
     REALIZED SPLIT:        1.000E+00


 ********  QUESTION:    2  Does HCC-91 fail?  (3", sch 10S, type 304 SS)

 Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                          1000
           BRANCHES:          HCC91-F   HCC91-NoF
                                 1         2
     REALIZED SPLIT:        0.000E+00 1.000E+00


 ********  QUESTION:    3  If segment HCC-91 fails, what is the rupture size?

 Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                            1000
           BRANCHES:          HCC91-Lg  HCC91-NoL
                                 1         2
     REALIZED SPLIT:        0.000E+00 1.000E+00

                       SUMMARY BY CASE

  CASE NUMBER/SPLIT:    2     1.000E+00
        DESCRIPTION:             Otherwise

  CASE/BRANCH SPLIT:          0.000E+00 1.000E+00
```

```
********  QUESTION:     4  Does HP-4 fail?  (3" flow element, 150 psi rating)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          1000
          BRANCHES:            HP4-F      HP4-NoF
                                 1          2
    REALIZED SPLIT:          0.000E+00 1.000E+00

********  QUESTION:     5  How large is the leak at HP-4?

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          1000
          BRANCHES:            HP4-Lg     HP4-Sm     HP4-Sp     HP4-NoL
                                 1          2          3          4
    REALIZED SPLIT:          0.000E+00 0.000E+00 1.170E-01 8.830E-01
```

LISTING 9

PSTEVNT Key Word File for HP2A-1

The following is a listing of the PSTEVNT key word file for the HP2A-1 model. This file is used to control PSTEVNT execution during the rebinning process used to obtain aggregate system failure mode probabilities.

```
$-- Calculation Control Keywords (for logical constants) ----------------------
$
$   COLLAPS  xxxx              $ Reduce rebinned results with weighing
$                                factor
$
    REBIN                      $ Causes rebinning of accident progression
$                                bins
$
    RUN                        $ Causes PSTEVNT to proceed with data
$                                calculations
$
    NOSORT                     $ Do not produce sort tables
$
$-- Calculation Control Keywords (for assigned values) ------------------------
$
$
$-- Input File Specification Keywords -----------------------------------------
$
    ASCTRIN                    $ ASCII output from EVNTRE
$
    BININ   pst_bin.dat        $ Filename for rebinning input
$
    EVNTBIN mc_posts.asc       $ Filename for EVNTRE output file
$
$   SORTIN  sortin             $ Filename for sort specification data
$
$-- Report Request Keywords ---------------------------------------------------
$
    ASCSAV                     $ Rebinning result is ASCII
$
    RPTMLST                    $ Write EVNTRE master bin list to message file
$
    RPTRBIN                    $ Write rebinned bins to message file
$
$-- Output File Specification Keywords ----------------------------------------
$
    BINOUT  rebins.out         $ Rebinning result data
$
$   INPOUT  inpout             $ Annotated echo of input
$
    KEEPOUT keep.out           $ Master list of unique kept bins
$
$   SBINOUT sbinout            $ Rebinning result data (for additional
$                                post-processing)
$
$   SORTOUT sortout            $ Result of requested sorts
$
```

```
$   TABOUT  tabout                   $ Rebinning result descriptive table(s)
$
ENDKEY                               $ Indicates the end of keyword input.
```

LISTING 10

PSTEVNT Rebinning Data File for HP2A-1

The following is a listing of the PSTEVNT rebinning data file for the HP2A-1 model.

```
ISLOCA Component Failure Rebinning -- HP2A-1
  1         FSize
  4   4    NoLeak     Spray     Small     Large
  1   1         1
                1
           NoLeak
  1   2         1
                2
            Spray
  1   3         1
                3
            Small
  1   4         1
                4
            Large
```

LISTING 11

PSTEVNT Output Data File for HP2A-1

The following is a listing of the PSTEVNT output data file for the HP2A-1 model. This file contains the system failure mode probabilities for the HP2A-1 model.

```
HP2A-1 BASE CASE
AGGREGATED REBINNING RESULTS FOR: Component Failure Rebinning -- HP2A-1

 FREQUENCY:
  BIN          TOTAL        ID                FSize
  8.7340E-01   8.7340E-01   A                 NoLeak
  1.2650E-01   9.9990E-01   B                 Spray
  1.0000E-04   1.0000E+00   D                 Large


A TOTAL OF     3 OUT OF     3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
FREQUENCY
```

LISTING 12

PSTEVNT Output Data File for HP2A-1 (Sensitivity)

The following is a listing of the PSTEVNT output data file for the HP2A-1 model. This file contains the system failure mode probabilities for the HP2A-1 model. These results differ from those in Listing 11 in that a log standard deviation of 0.10 was used for the HCC-91 pressure capacity (instead of 0.36).

```
SENSITIVITY WITH HCC-91 LOG SIGMA = .1
AGGREGATED REBINNING RESULTS FOR: Component Failure Rebinning -- HP2A-1

 FREQUENCY:
  BIN         TOTAL       ID                FSize
  8.8130E-01  8.8130E-01  A                 NoLeak
  1.1870E-01  1.0000E+00  B                 Spray

A TOTAL OF     2 OUT OF     2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
FREQUENCY
```

LISTING 13

Program Used to Generate Distributions for HP2A-2

The following is a listing of the Fortran 77 program used to generate the Monte Carlo sample data required by the EVNTRE program for evaluation of the HP2A-2 scenario. Some of the subroutines used by the program are not shown in Listing 13. Listing 24 provides the source code for the missing subroutines.

```
*   PURPOSE:
*
*   This program calculates the normal distributions required for the
*   Monte Carlo evaluation of the ISLOCA HP2A-2 scenario.
*   The first distribution corresponds to
*   system pressure.  The required output for the first parameter is
*   actually the natural log of the system pressure.  The remaining
*   distributions are log normal, and are described by a log mean and
*   logarithmic std. dev.  The output is written in the format required by
*   the EVNTRE program.
*
*   INPUT:
*
*   The input is hardwired into the code, except for the value of the
*   required random seed, which is either read from the data file 'RANS.DAT'
*   or input by the user.  The option is provided by user dialog at run time.
*
*   OUTPUT:
*
*   The output is the required normal distributions, and is written to file
*   'MCARLO.DAT'.  The data are in the format required by the EVNTRE code for
*   use as sample data.  The last value of the random seed is also written to
*   the file 'RANS.DAT' for use in the next evaluation.
*
*   FILES:
*   ------------------------------------------------
*   Unit   Description
*   ----   -------------------------------------------
*     5    User input from console
*     6    Program output to console
*    10    Saved value of random seed
*    11    Output for use as an EVNTRE sample file
*   ------------------------------------------------
*
*   WRITTEN BY:
*
*   John Schroeder 1/11/90
*   The subroutines SEEDIN, GENNOR, and SEEDOU were written by Cory Attwood.
*
      PROGRAM HP2A2

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      CHARACTER*8 FNSAVE
      CHARACTER*3 FN5
      REAL  X1(10000),  X2(10000),  X3(10000),  X4(10000),  X5(10000)
      REAL  X6(10000),  X7(10000),  X8(10000)
```

```
      DATA IOU5/5/, IOU6/6/, IOSAVE/10/
      DATA FNSAVE /'RANS.DAT'/, FN5/'CON'/

      OPEN (UNIT=5,FILE='CON')
      OPEN (UNIT=6,FILE='CON')

      CALL SEEDIN(ISEED,IOU5,IOU6,IOSAVE,FNSAVE,FN5)
      WRITE(6,10) ISEED
10    FORMAT(' ISEED =',I15)

      N     = 10000                      ! Pick 10000 Numbers
      NX    = 8                          ! Number of distributions requested
      ISORT = 0                          ! Don't sort the numbers

*     Generate the distribution for the initial system pressure

      AMEAN = 2000.                      ! Mean of Normal distribution
      STDEV = 50.                        ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X1,N,ISORT,ISEED,IER,IPOINT)
      DO 15  I=1, N
         X1(I) = ALOG(X1(I))             ! Use natural log of pressure
15    CONTINUE

*     Generate the distribution for failure of P58 2

      AMEAN = 7.719                      ! Mean of Normal distribution
      STDEV = 0.250                      ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X2,N,ISORT,ISEED,IER,IPOINT)

*     Generate the distribution for failure of 6"-GCB-4

      AMEAN = 7.405                      ! Mean of Normal distribution
      STDEV = 0.360                      ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X3,N,ISORT,ISEED,IER,IPOINT)

*     Generate the distribution for failure of 6GCB4a

      AMEAN = 7.767                      ! Mean of Normal distribution
      STDEV = 0.120                      ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X4,N,ISORT,ISEED,IER,IPOINT)

*     Generate the distribution for failure of 6GCB4b

      AMEAN = 7.767                      ! Mean of Normal distribution
      STDEV = 0.120                      ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X5,N,ISOPT,ISEED,IER,IPOINT)

*     Generate the distribution for failure of HP-13

      AMEAN = 7.682                      ! Mean of Normal distribution
```

```
      STDEV = 0.250                   ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X6,N,ISORT,ISEED,IER,IPOINT)

*     Generate the distribution for failure of 4"-GCB-2

      AMEAN = 7.638                   ! Mean of Normal distribution
      STDEV = 0.360                   ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X7,N,ISORT,ISEED,IER,IPOINT)

*     Generate the distribution for failure of 4"-GCB-11

      AMEAN = 7.638                   ! Mean of Normal distribution
      STDEV = 0.360                   ! Standard Deviation of distribution

      CALL GENNOR(AMEAN,STDEV,X8,N,ISORT,ISEED,IER,IPOINT)

*     Write out distributions

      OPEN(11,FILE='MCARLO.DAT',STATUS='UNKNOWN')
      DO 20, I=1, N
         WRITE(11,25) I, NX, X1(I), X2(I), X3(I), X4(I), X5(I),
     *                   X6(I), X7(I), X8(I)
20    CONTINUE
25    FORMAT(2I10, 5G12.5,/(20X,5G12.5))
      CLOSE(11)

      CALL SEEDOU(ISEED,IOU6,IOSAVE,FNSAVE)
      WRITE(6,10) ISEED

      CLOSE(5)
      CLOSE(6)
      CLOSE(10)

      END
```

LISTING 14

EVNTRE Key Word File for HP2A-2

The following is a listing of the EVNTRE key word file for the HP2A-2 model. The EVNTRE key word file controls the mode of execution, input and output options, and cutoff values used by the program during event tree evaluation.

```
$-- Calculation Control Keywords ----------------------------------------------
$
    MODE 3                    $ Specifies the calculational mode for EVNTRE.
$                                   1 = point estimate
$                                   3 = sampling mode (one vector each eval)
$                                   4 = sampling mode (two vectors each eval)
$
    NOBIN                     $ Turns the binning facility on/off.
$
$
    RUN                       $ Indicates that the tree is to be evaluated
$                               after the input data has been processed
$
$   KEEPCUT 1.0E-6            $ Specifies the path frequency below which a
$                               path is terminated.
$
$-- Input File Specification Keywords -----------------------------------------
$
    TREEIN tree dat           $ Specifies the input file name for the
$                               tree definition input file.
$
$   BININ  bin.dat            $ Specifies the input file name for the
$                               binning and sorting information input file.
$
    SAMDIN mc_pntr.dat        $ Specifies the input file name for the
$                               sample definition information input file.
$
    SAM1IN mcarlo.dat         $ Specifies the input file name for the
$                               first set of sample input vectors
$
$   SAM2IN mcarlo.dat         $ Specifies the input file name for the
$                               second set of sample input vectors.
$
$-- Report Request Keywords ---------------------------------------------------
$
$   PRTINP                    $ Turns on the annotated echo of input.
$
    STATS                     $ Indicates that a branch and case frequency
$                               table report will be generated.
$
    PRUNE                     $ Causes unused cases to be dropped from the
$                               branch and case frequency table.
$
    NWRTBIN                   $ Indicates that a binning result report will
$                               be generated when the paths through the
$                               tree are binned.
$
```

```
$    PRTCUT 1.0E-6                  $ Specifies the minimum bin frequency required
$                                     to report a bin.
$
$    SAVEBIN                        $ Indicates that a binning results file will
$                                     be generated for post-processing.
$
$-- Output File Specification Keywords ----------------------------------------
$
$    INPOUT  echo.out               $ Specifies the output file name for the
$                                     annotated echo of input.
$
$    BINOUT  bin.out                $ Specifies the output file name for the
$                                     binning result report.
$
     STATOUT mc_freq.out            $ Specifies the output file name for the
$                                     branch and case frequency table.
$
     SAMROUT mc_post.out            $ Specifies the output file name for the
$                                     post-processing file.
$
ENDKEY                              $ Indicates the end of keyword input.
```

LISTING 15

EVNTRE Tree Definition File for HP2A-2

The following is a listing of the EVNTRE event tree definition file for the HP2A-2 model. This file provides the event tree structure and default probability and parameter values for the HP2A-2 model.

```
ISLOCA System Rupture Modcl -- HP2A-2
    15
    NQ
     1      1.000
       'MC Eval'
  1 What is the pressure in the Interfacing System?
     1 CCB-2-HiP
     3          1
            1.000
     8                      $ Point estimates follow:
     1      7.601           $     lPs' (log of system press.)
     2      7.719           $     lPf' for P58-2
     3      7.405           $     lPf' for GCB-4
     4      7.767           $     lPf' for GCB4a
     5      7.767           $     lPf' for GCB4b
     6      7.682           $     lPf' for HP-13
     7      7.638           $     lPf' for GCB-2
     8      7.638           $     lPf' for GCB-11
  2 Does HPI pump P58-2 fail (i.e. seal failure)?
     2    P582-F  P582-NoF
     5         1         2
     2         1         2
           lPs'      lPf'
         FUN-CMP
           EQUAL          0
       IF lPs' .GT. lPf' THEN P582-F
  3 How large is the failure at HPI pump P58-2?
     3   P582-Sm   P582-Sp  P582-NoL
     5         1         2         3
     2         1         2
           lPs'      lPf'
        FUN-RPSZ
        GETHRESH          2      1.00      0.75
       Bin Ps'/Pf'
  4 Does pipe GCB-4 fail? (6" pipe, sch 10S, type 304SS, 300# rated)
     2    GCB4-F  GCB4-NoF
     5         1         2
     2         1         3
           lPs'      lPf'
         FUN-CMP
           EQUAL          0
       IF lPs' .GT. lPf' THEN GCB4-F
```

```
 5 How large is the leak at GCB-4?
     2   GCB4-Lg  GCB4-NoL
     2         1         2
     2
     1         4
               1
         GCB4-F
           1.000     0.000
       Otherwise
           0.000     1.000
 6 Does GCB4a fail?  (6" flange, 300 psi rating)
     2   GCB4a-F GCB4a-NoF
     5         1         2
     2         1         4
            1Ps'      1Pf'
         FUN-CMP
           EQUAL         0
       IF 1Ps' .GT. 1Pf' THEN GCB4a-F
 7 How large is the leak at GCB4a?
     2  GCB4a-Lg GCB4a-NoL
     2         1         2
     2
     1         6
               1
         GCB4a-F
           1.000     0.000
       Otherwise
           0.000     1.000
 8 Does GCB4b fail?  (6" flange, 300 psi rating)
     2   GCB4b-F GCB4b-NoF
     5         1         2
     2         .         5
            1Ps'      1Pf'
         FUN-CMP
           EQUAL         0
       IF 1Ps' .GT. 1Pf' THEN GCB4b-F
 9 How large is the leak at GCB4b?
     2  GCB4b-Lg GCB4b-NoL
     2         1         2
     2
     1         8
               1
         GCB4b-F
           1.000     0.000
       Otherwise
           0.000     1.000
10 Does local-manual gate valve HP-13 fail?  (6", 300 psi rating)
     2    HP13-F  HP13-NoF
     5         1         2
     2         1         6
            1Ps'      1Pf'
         FUN-CMP
           EQUAL         0
       IF 1Ps' .GT. 1Pf' THEN HP13-F
```

```
11 How large is the leak at HP-13?
     2    HP13-Sm  HP13-NoL
     2          1         2
     2
     1         10
                1
           HP13-F
             1.000     0.000
        Otherwise
             0.000     1.000
12 Does pipe GCB-2 fail? (4" pipe, sch 10S, type 304 SS)
     2     GCB2-F  GCB2-NoF
     5          1         2
     2          1         7
              1Ps'      1Pf'
          FUN-CMP
            EQUAL         0
       IF 1Ps' .GT. 1Pf' THEN GCB2-F
13 How large is the leak at GCB-2?
     2    GCB2-Lg  GCB2-NoL
     2          1         2
     2
     1         12
                1
           GCB2-F
             1.000     0.000
        Otherwise
             0.000     1.000
14 Does pipe 4"-GCB-11 fail? (4", sch 10S, type 304 SS)
     2    GCB11-F GCB11-NoF
     5          1         2
     2          1         8
              1Ps'      1Pf'
          FUN-CMP
            EQUAL         0
       IF 1Ps' .GT. 1Pf' THEN GCB11-F
15 How large is the leak at GCB-11?
     2   GCB11-Lg GCB11-NoL
     2          1         2
     2
     1         14
                1
          GCB11-F
             1.000     0.000
        Otherwise
             0.000     1.000
```

LISTING 16

EVNTRE Sample Definition File for HP2A-2

The following is a listing of the EVNTRE sample definition data file for the HP2A-2 model. This file supplies the specification required to set up the sampling modes for the tree.

```
ISLOCA SAMPLE RUN
10000     1
8
M1,1,1,1,A  M1,1,2,1,A  M1,1,3,1,A  M1,1,4,1,A  M1,1,5,1,A
M1,1,6,1,A  M1,1,7,1,A  M1,1,8,1,A
```

LISTING 17

EVNTRE Binning Data File for HP2A-2

The following is a listing of the EVNTRE binning data file for the HP2A-2 model. These data specify the logic used to select HP2A-2 event tree end states that are included in each system failure mode bin.

```
ISLOCA Component Failure Binning -- HP2A-2
  1         FSize
  4    4   NoLeak     Spray      Small      Large
  7    1        3         5          7          9         11         13         15
                3       * 2        * 2        * 2        * 2        * 2        * 2
         P582-NoL  GCB4-NoL  GCB4a-NoL  GCB4b-NoL   HP13-NoL   GCB2-NoL  GCB11-NoL
  7    2        3         5          7          9         11         13         15
                2       * 2        * 2        * 2        * 2        * 2        * 2
          P582-Sp  GCB4-NoL  GCB4a-NoL  GCB4b-NoL   HP13-NoL   GCB2-NoL  GCB11-NoL
  7    3        3        11          5          7          9         13         15
              ( 1      + 1)        * 2        * 2        * 2        * 2        * 2
          P582-Sm   HP13-Sm   GCB4-NoL  GCB4a-NoL  GCB4b-NoL   GCB2-NoL  GCB11-NoL
  5    4        5         7          9         13         15
                1       + 1        + 1        + 1        + 1
          GCB4-Lg  GCB4a-Lg   GCB4b-Lg    GCB2-Lg   GCB11-Lg
```

LISTING 18

EVNTRE Frequency Output File for HP2A-2

The following is a listing of the EVNTRE frequency output file for the HP2A-2 model. This file contains the individual component failure mode probabilities resulting from the Monte Carlo evaluation of the HP2A-2 model.

```
          TREE ID: ISLOCA System Rupture Model -- HP2A-2
     # OF QUESTIONS    15
       OBSERVATIONS: 10000
         FOR SERIES: ISLOCA SAMPLE RUN
        SEQUENCE ID:           MC Eval

 ********  QUESTION:     1  What is the pressure in the Interfacing System?

 Q-TYPE/TIMES ASKED:        INDEP. INPUT PROB. INPUT PARM.             10000
           BRANCHES:            CCB-2-HiP
                                   1
     REALIZED SPLIT:          1.000E+00

 ********  QUESTION:     2  Does HPI pump P58-2 fail (i.e. seal failure)?

 Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                         10000
           BRANCHES:            P582-F     P582-NoF
                                   1          2
     REALIZED SPLIT:          3.202E-01 6.798E-01

 ********  QUESTION:     3  How large is the failure at HPI pump P58-2?

 Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                         10000
           BRANCHES:            P582-Sm    P582-Sp    P582-NoL
                                   1          2          3
     REALIZED SPLIT:          3.204E-01 4.346E-01 2.450E-01

 ********  QUESTION:     4  Does pipe GCB-4 fail? (6" pipe, sch 10S, type 304SS,
300# rated)
 Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                         10000
           BRANCHES:            GCB4-F     GCB4-NoF
                                   1          2
     REALIZED SPLIT:          7.096E-01 2.904E-01

 ********  QUESTION:     5  How large is the leak at GCB-4?

 Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                           10000
           BRANCHES:            GCB4-Lg    GCB4-NoL
                                   1          2
     REALIZED SPLIT:          7.096E-01 2.904E-01

                          SUMMARY BY CASE
```

```
 CASE NUMBER/SPLIT:     1     7.096E-01
      DEPENDENCIES:     4

     REQ. BRANCHES:     1

       DESCRIPTION:  GCB4-F

 CASE/BRANCH SPLIT:           7.096E-01 0.000E+00

 CASE NUMBER/SPLIT:     2     2.904E-01
       DESCRIPTION:         Otherwise

 CASE/BRANCH SPLIT:           0.000E+00 2.904E-01

********  QUESTION:     6  Does GCB4a fail?  (6" flange, 300 psi rating)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          10000
          BRANCHES:            GCB4a-F   GCB4a-NoF
                                  1         2
    REALIZED SPLIT:           8.670E-02 9.133E-01

********  QUESTION:     7  How large is the leak at GCB4a?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                            10000
          BRANCHES:            GCB4a-Lg  GCB4a-NoL
                                  1         2
    REALIZED SPLIT:           8.670E-02 9.133E-01

                         SUMMARY BY CASE

 CASE NUMBER/SPLIT:     1     8.670E-02
      DEPENDENCIES:     6

     REQ. BRANCHES:     1

       DESCRIPTION:  GCB4a-F

 CASE/BRANCH SPLIT:           8.670E-02 0.000E+00

 CASE NUMBER/SPLIT:     2     9.133E-01
       DESCRIPTION:         Otherwise

 CASE/BRANCH SPLIT:           0.000E+00 9.133E-01

********  QUESTION:     8  Does GCB4b fail?  (6" flange, 300 psi rating)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          10000
          BRANCHES:            GCB4b-F   GCB4b-NoF
                                  1         2
    REALIZED SPLIT:           8.530E-02 9.147E-01
```

```
********  QUESTION:    9  How large is the leak at GCB4b?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                         10000
          BRANCHES:            GCB4b-Lg  GCB4b-NoL
                                  1         2
    REALIZED SPLIT:           8.530E-02 9.147E-01

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1      8.530E-02
       DEPENDENCIES:    8

      REQ. BRANCHES:    1

        DESCRIPTION:  GCB4b-F

  CASE/BRANCH SPLIT:           8.530E-02 0.000E+00

  CASE NUMBER/SPLIT:    2      9.147E-01
        DESCRIPTION:          Otherwise

  CASE/BRANCH SPLIT:           0.000E+00 9.147E-01

********  QUESTION:   10  Does local-manual gate valve HP-13 fail?  (6", 300 psi
rating)
Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                       10000
          BRANCHES:            HP13-F    HP13-NoF
                                  1         2
    REALIZED SPLIT:           3.689E-01 6.311E-01

********  QUESTION:   11  How large is the leak at HP-13?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                         10000
          BRANCHES:            HP13-Sm   HP13-NoL
                                  1         2
    REALIZED SPLIT:           3.689E-01 6.311E-01

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1      3.689E-01
       DEPENDENCIES:   10

      REQ. BRANCHES:    1

        DESCRIPTION:  HP13-F

  CASE/BRANCH SPLIT:           3.689E-01 0.000E+00

  CASE NUMBER/SPLIT:    2      6.311E-01
        DESCRIPTION:          Otherwise

  CASE/BRANCH SPLIT:           0.000E+00 6.311E-01
```

```
********  QUESTION:   12  Does pipe GCB-2 fail? (4" pipe, sch 10S, type 304 SS)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                        10000
          BRANCHES:            GCB2-F    GCB2-NoF
                                 1          2
    REALIZED SPLIT:          4.562E-01 5.438E-01

********  QUESTION:   13  How large is the leak at GCB-2?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          10000
          BRANCHES:            GCB2-Lg   GCB2-NoL
                                 1          2
    REALIZED SPLIT:          4.562E-01 5.438E-01

                       SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      4.562E-01
      DEPENDENCIES:   12

     REQ. BRANCHES:    1

       DESCRIPTION:  GCB2-F

 CASE/BRANCH SPLIT:           4.562E-01 0.000E+00

 CASE NUMBER/SPLIT:    2      5.438E-01
       DESCRIPTION:          Otherwise

 CASE/BRANCH SPLIT:           0.000E+00 5.438E-01

********  QUESTION:   14  Does pipe 4"-GCB-11 fail? (4", sch 10S, type 304 SS)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                        10000
          BRANCHES:            GCB11-F   GCB11-NoF
                                 1          2
    REALIZED SPLIT:          4.508E-01 5.492E-01

********  QUESTION:   15  How large is the leak at GCB-11?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          10000
          BRANCHES:            GCB11-Lg  GCB11-NoL
                                 1          2
    REALIZED SPLIT:          4.508E-01 5.492E-01

                       SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      4.508E-01
      DEPENDENCIES:   14

     REQ. BRANCHES:    1
```

```
     DESCRIPTION.  GCB11-F

CASE/BRANCH SPLIT:           4.508E-01 0.000E+00

CASE NUMBER/SPLIT:     2     5.492E-01
     DESCRIPTION:           Otherwise

CASE/BRANCH SPLIT:           0.000E+00 5.492E-01
```

LISTING 19

EVNTRE Frequency Output File for HP2A-2 (Sensitivity)

The following is a listing of the EVNTRE frequency output file for the HP2A-2 model. This file contains the individual component failure mode probabilities resulting from the Monte Carlo evaluation of the HP2A-2 model. The failure model probabilities provided in this listing result from using a pipe failure log standard deviation of 0.10 instead of 0.36 as was used to produce Listing 18.

```
          TREE ID: ISLOCA System Rupture Model -- HP2A-2s
   # OF QUESTIONS:   15
     OBSERVATIONS: 10000
       FOR SERIES: ISLOCA SAMPLE RUN
      SEQUENCE ID:            MC Eval


 ********  QUESTION:    1  What is the pressure in the Interfacing System?

 Q-TYPE/TIMES ASKED:       INDEP. INPUT PROB. INPUT PARM.            10000
           BRANCHES:          CCB-2-HiP
                                 1
    REALIZED SPLIT:         1.000E+00


 ********  QUESTION:    2  Does HPI pump P58-2 fail (i.e. seal failure)?

 Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                        10000
           BRANCHES:          P582-F    P582-NoF
                                 1         2
    REALIZED SPLIT:         3.203E-01 6.797E-01


 ********  QUESTION:    3  How large is the failure at HPI pump P58-2?

 Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                        10000
           BRANCHES:          P582-Sm   P582-Sp   P582-NoL
                                 1         2         3
    REALIZED SPLIT:         3.205E-01 4.290E-01 2.505E-01


 ********  QUESTION:    4  Does pipe GCB-4 fail? (6" pipe, sch 10S, type 304SS,
300# rated)
 Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                        10000
           BRANCHES:          GCB4-F    GCB4-NoF
                                 1         2
    REALIZED SPLIT:         9.697E-01 3.030E-02


 ********  QUESTION:    5  How large is the leak at GCB-4?

 Q-TYPE/TIMES ASKED:       DEP. [illegible]PUT PROB.                 10000
           BRANCHES:          [illegible]-Lg    GCB4-NoL
                                           2
    REALIZED SPLIT:         [illegible]E-01 3.030E-02
```

SUMMARY BY CASE

```
CASE NUMBER/SPLIT:   1    9.697E-01
     DEPENDENCIES:   4

    REQ. BRANCHES:   1

      DESCRIPTION:  GCB4-F

CASE/BRANCH SPLIT:        9.697E-01 0.000E+00

CASE NUMBER/SPLIT:   2    3.030E-02
      DESCRIPTION:       Otherwise

CASE/BRANCH SPLIT:        0.000E+00 3.030E-02

********  QUESTION:    6  Does GCB4a fail?  (6" flange, 300 psi rating)

Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                    10000
          BRANCHES:          GCB4a-F   GCB4a-NoF
                               1          2
    REALIZED SPLIT:        8.530E-02 9.147E-01

********  QUESTION:    7  How large is the leak at GCB4a?

Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                      10000
          BRANCHES:          GCB4a-Lg  GCB4a-NoL
                               1          2
    REALIZED SPLIT:        8.530E-02 9.147E-01
```

SUMMARY BY CASE

```
CASE NUMBER/SPLIT:   1    8.530E-02
     DEPENDENCIES:   6

    REQ. BRANCHES:   1

      DESCRIPTION:  GCB4a-F

CASE/BRANCH SPLIT:        8.530E-02 0.000E+00

CASE NUMBER/SPLIT:   2    9.147E-01
      DESCRIPTION:       Otherwise

CASE/BRANCH SPLIT:        0.000E+00 9.147E-01

********  QUESTION:    8  Does GCB4b fail?  (6" flange, 300 psi rating)

Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                    10000
          BRANCHES:          GCB4b-F   GCB4b-NoF
                               1          2
    REALIZED SPLIT:        8.580E-02 9.142E-01
```

```
******** QUESTION:    9  How large is the leak at GCB4b?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          10000
          BRANCHES:            GCB4b-Lg  GCB4b-NoL
                                 1         2
    REALIZED SPLIT:          8.580E-02 9.142E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1     8.580E-02
      DEPENDENCIES:   8

      REQ. BRANCHES:  1

       DESCRIPTION:  GCB4b-F

 CASE/BRANCH SPLIT:          8.580E-02 0.000E+00

 CASE NUMBER/SPLIT:   2     9.142E-01
       DESCRIPTION:        Otherwise

 CASE/BRANCH SPLIT:          0.000E+00 9.142E-01

******** QUESTION:   10  Does local-manual gate valve HP-13 fail?  (6", 300 psi
rating)
Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                        10000
          BRANCHES:            HP13-F    HP13-NoF
                                 1         2
    REALIZED SPLIT:          3.712E-01 6.288E-01

******** QUESTION:   11  How large is the leak at HP-13?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          10000
          BRANCHES:            HP13-Sm   HP13-NoL
                                 1         2
    REALIZED SPLIT:          3.712E-01 6.288E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1     3.712E-01
      DEPENDENCIES:  10

      REQ. BRANCHES:  1

       DESCRIPTION:  HP13-F

 CASE/BRANCH SPLIT:          3.712E-01 0.000E+00

 CASE NUMBER/SPLIT:   2     6.288E-01
       DESCRIPTION:        Otherwise

 CASE/BRANCH SPLIT:          0.000E+00 6.288E-01
```

```
******** QUESTION:   12  Does pipe GCB-2 fail? (4" pipe, sch 10S, type 304 SS)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                         10000
          BRANCHES:           GCB2-F    GCB2-NoF
                                1          2
    REALIZED SPLIT:          3.555E-01 6.445E-01

******** QUESTION:   13  How large is the leak at GCB-2?

Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                           10000
          BRANCHES:           GCB2-Lg   GCB2-NoL
                                1          2
    REALIZED SPLIT:          3.555E-01 6.445E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1      3.555E-01
      DEPENDENCIES:  12

     REQ. BRANCHES:   1

       DESCRIPTION:  GCB2-F

 CASE/BRANCH SPLIT:          3.555E-01 0.000E+00

 CASE NUMBER/SPLIT:   2      6.445E-01
       DESCRIPTION:        Otherwise

 CASE/BRANCH SPLIT:          0.000E+00 6.445E-01

******** QUESTION:   14  Does pipe 4"-GCB-11 fail? (4", sch 10S, type 304 SS)

Q-TYPE/TIMES ASKED:       INDEP  CALC. PROB.                         10000
          BRANCHES:           G   1-F   GCB11-NoF
                                1          2
    REALIZED SPLIT:          3.633E-01 6.367E-01

******** QUESTION:   15  How large is the leak at GCB-11?

Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                           10000
          BRANCHES:           GCB11-Lg  GCB11-NoL
                                1          2
    REALIZED SPLIT:          3.633E-01 6.367E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1      3.633E-01
      DEPENDENCIES:  14

     REQ. BRANCHES:   1
```

DESCRIPTION: GCB11-F

CASE/BRANCH SPLIT: 3.633E-01 0.000E+00

CASE NUMBER/SPLIT: 2 6.367E-01
DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 6.367E-01

LISTING 20

PSTEVNT Key Word File for HP2A-2

The following is a listing of the PSTEVNT key word for the HP2A-2 model. This file is used to control PSTEVNT execution during the rebinning process used to obtain aggregate system failure mode probabilities.

```
$-- Calculation Control Keywords (for logical constants) ---------------------
$
$   COLLAPS xxxxx              $ Reduce rebinned results with weighing
$                                factor
$
    REBIN                      $ Causes rebinning of accident progression
$                                bins
$
    RUN                        $ Causes PSTEVNT to proceed with data
$                                calculations
$
    NOSORT                     $ Do not produce sort tables
$
$-- Calculation Control Keywords (for assigned values) -----------------------
$
$
$-- Input File Specification Keywords ----------------------------------------
$
    ASCTRIN                    $ ASCII output from EVNTRE
$
    BININ   pst_bin.dat        $ Filename for rebinning input
$
    EVNTBIN mc_post.asc        $ Filename for EVNTRE output file
$
$   SORTIN  sortin             $ Filename for sort specification data
$
$-- Report Request Keywords --------------------------------------------------
$
    ASCSAV                     $ Rebinning result is ASCII
$
    RPTMLST                    $ Write EVNTRE master bin list to message file
$
    RPTRBIN                    $ Write rebinned bins to message file
$
$-- Output File Specification Keywords ---------------------------------------
$
    BINOUT  rebin.out          $ Rebinning result data
$
$   INPOUT  inpout             $ Annotated echo of input
$
    KEEPOUT keep.out           $ Master list of unique kept bins
$
$   SBINOUT sbinout            $ Rebinning result data (for additional
$                                post-processing)
$
$   SORTOUT sortout            $ Result of requested sorts
$
```

```
$   TABOUT  tabout                $ Rebinning result descriptive table(s)
$
ENDKEY                            $ Ind'cates the end of keyword input.
```

LISTING 21

PSTEVNT Rebinning Data File for HP2A-2

The following is a listing of the PSTEVNT rebinning data file for the HP2A-2 model.

```
ISLOCA Component Failure Rebinning -- HP2A-2
  1         FSize
  4   4    NoLeak      Spray      Small      Large
  1   1         1
                1
           NoLeak
  1   2         1
                2
            Spray
  1   3         1
                3
            Small
  1   4         1
                4
            Large
```

LISTING 2z

PSTEVNT Output Data File for HP2A-2

The following is a listing of the PSTEVNT output data file for the HP2A-2 model. This file contains the system failure mode probabilities for the HP2A-2 model.

```
HP2A-2 BASE CASE
AGGREGATED REBINNING RESULTS FOR: Component Failure Rebinning -- HP2A-2


 FREQUENCY:
  BIN          TOTAL       ID                 FSize
  9.2290E-01  9.2290E-01 D                    Large
  4.1000E-02  9.6390E-01 C                    Small
  2.4300E-02  9.8820E-01 B                    Spray
  1.1800E-02  1.0000E+00 A                    NoLeak


A TOTAL OF     4 OUT OF     4 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
FREQUENCY
```

LISTING 23

PSTEVNT Output Data File for HP2A-2 (Sensitivity)

The following is a listing of the PSTEVNT output data file for the HP2A-2 model. This file contains the system failure mode probabilities for the HP2A-2 model. These results differ from those in Listing 22 in that a log standard deviation of 0.10 was used for the piping pressure capacity (instead of 0.36).

```
SENSITIVITY WITH PIPE FAILURE LOG SIGMA = .1
AGGREGATED REBINNING RESULTS FOR: Component Failure Rebinning -- HP2A-2

 FREQUENCY:
  BIN          TOTAL       ID                FSize
  9.8640E-01   9.8640E-01  D                 Large
  8.5000E-03   9.9490E-01  C                 Small
  2.9000E-03   9.9780E-01  B                 Spray
  2.2000E-03   1.0000E+00  A                 NoLeak


A TOTAL OF     4 OUT OF     4 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
FREQUENCY
```

LISTING 24

Program Used to Generate Distributions for DHR Letdown

The following is a listing of the Fortran 77 program used to generate the Monte Carlo sample data required by the EVNTRE program for evaluation of the DHR Letdown model.

```
*** PROGRAM DHRL ****************************************************************
*
*   PURPOSE:
*
*   This program calculates the normal distributions required for the
*   Monte Carlo evaluation of the ISLOCA DHR letdown scenario.  Thirty
*   distributions are required.  The first two distributions correspond to
*   pressure at different locations in the system.  The required output
*   for the first two parameters is actually the natural log of the
*   system pressure.  The remaining distributions are log normal, and
*   are described by a log mean and logarithmic std. dev.  The output is
*   written in the format required by the EVNTRE program.
*
*   INPUT:
*
*   The input is read from two different data files.  One provided by
*   input redirection (containing problem control info.), and the other
*   with the required filename 'UNIFORM'.  The last file must contain
*   the uniformly distributed reactor system pressures.  The first contains
*   the component failure data.
*
*   OUTPUT:
*
*   The output is the required normal distributions, and is written to file
*   'MCARLO.DAT'.  The data is in the format required by the EVNTRE code for
*   use as sample data.
*
*   FILES:
*   ----------------------------------------------
*   Unit   Description
*   ----   -------------------------------------
*     5    User input from console
*     6    Program output to console
*    10    Uniform distribution data
*    11    Output for use as an EVNTRE sample file
*   ----------------------------------------------
*
*   WRITTEN BY:
*
*   John Schroeder 1/11/90
*
      PROGRAM DHRL

      IMPLICIT NONE
      INTEGER I, IER, IPOINT, ISEED, ISORT, J, N, NDIS
      REAL  MEAN(30), STDEV(30), UX, XX(30,10000),  X(10000)
```

```
*     Read data file

      CALL READIN(ISEED, ISORT, N, NDIS, MEAN, STDEV)

*     The first two distributions require special treatment.  The are not
*     independent, and not based on a normal distribution.  Instead, a
*     a uniform distribution is used to describe the behavior of reactor
*     pressure over the range 300 to 2200 psi.  Then the first two
*     distributions are derived from the reactor pressure using a curve
*     fit to RELAP generated pressures as a function of reactor pressure.

*     Read in the uniform distribution, calculate the new pressures

      OPEN(10, FILE='UNIFORM', STATUS='OLD')
      DO 1  I = 1, N
          READ(10, *) UX
          XX(1,I) = ALOG(.9584 * UX - 10.22)
          XX(2,I) = ALOG(.5715 * UX + 103.6)
1     CONTINUE
      CLOSE(10)

*     Generate the remaining normal distributions

      DO 10,  I = 3, NDIS
          CALL GENNOR(MEAN(I),STDEV(I),X,N,ISORT,ISEED,IER,IPOINT)
          DO 5,  J = 1, N
              XX(I,J) = X(J)
5         CONTINUE
10    CONTINUE

*     Write out distributions

      OPEN(11, FILE='MCARLO.DAT', STATUS='UNKNOWN')
      DO 20  I = 1, N
          WRITE(11, 25) I, NDIS, (XX(J,I), J = 1, NDIS)
20    CONTINUE
25    FORMAT(2I10, 5G12.5,/(20X,5G12.5))
      CLOSE(11)

      STOP
      END

*** READIN ********************************************************************
*
*   PURPOSE:
*
*   This subroutine reads in the program control data, and the values
*   used to calculate the requested normal distributions.
*
*   ARGUMENTS:
*
*   ------------------------------------------------------------------
*   Variable    Description
*   --------    ------------------------------------------------------
*   ISEED       Random seed                                       [1]
```

```
*   ISORT       Sort flag -- 0 => no sort, 1 => sorted                     [1]
*   N           Number of values requested in each distribution            [1]
*   NDIS        Number of distributions requested                          [1]
*   MEAN        Array of mean values for each requested normal distr.      [1]
*   STDEV       Array of standard deviations for each distr.               [1]
*   ---------------------------------------------------------------------------
*   Notes:
*   1.  Value(s) returned to calling program unit
*
*   FILES:
*
*   input on unit 5 (console -- use redirection to feed in data file)
*
*   WRITTEN BY:
*
*   John Schroeder  1/19/90
*
      SUBROUTINE READIN(ISEED, ISORT, N, NDIS, MEAN, STDEV)

      IMPLICIT NONE
      CHARACTER*80 LINE
      INTEGER I, IDIS, ISEED, ISORT, N, NDIS
      REAL MEAN(*), STDEV(*)

*     Read a comment line (discarded), then program control info

      READ(5, '(A)') LINE
      READ(5, *) ISORT, N, NDIS, ISEED

*     Read a comment line (also discarded), then means and standard
*     deviations

      READ(5, '(/A)') LINE
      DO 10  I = 1, NDIS
          READ(5, *)  IDIS, MEAN(IDIS), STDEV(IDIS)
10    CONTINUE

      RETURN
      END

*** GENNOR ********************************************************************

      SUBROUTINE GENNOR(AMEAN,STDEV,X,N,ISORT,ISEED,IER,IPOINT)
C
C     GENERATES RANDOM SAMPLE OF N NUMBERS FROM NORMAL POPULATION.
C     INPUTS
C        AMEAN = MEAN OF POPULATION
C        STDEV = STANDARD DEVIATION (= SQRT OF VARIANCE) OF POPULATION
C        N = NUMBER OF VALUES WANTED
C        ISORT = 0 IF VALUES ARE TO BE IN ORDER GENERATED
C              = 1 IF VALUES ARE TO BE SORTED INTO INCREASING ORDER
C        ISEED = INITIAL SEED, AN INTEGER IN RANGE OF INTEGERS ON
C                THE MACHINE USED.  SEE COMMENTS IN FUNCTION URAND,
C                WHERE THIS RANGE IS DEFINED BY PROGRAMMER AND CHECKED
C                BY PROGRAM.
```

```
C     OUTPUT
C        X = REAL ARRAY, DIMENSIONED TO SIZE AT LEAST N IN CALLING
C             PROGRAM.  THE RANDOM SAMPLE IS RETURNED AS X.
C        IER = 0 IF NO ERRORS RECOGNIZED.
C            = 1 IF PROBLEM IN THE TAILS OF THE NORMAL DISTRIBUTICN.
C            = 2 IF URAND GOT ANSWER OUTSIDE OF [0., 1.]
C        IPOINT = THE ELEMENT OF X THAT CAUSED ERROR FLAG TO TURN ON.
C              SUBROUTINE RETURNS AS SOON AS IER > 0.

C     WRITTEN BY C. ATWOOD, DEC.1989, BASED ON EARLIER PROGRAMS
C
      IMPLICIT REAL(A-H,O-Z), INTEGER (I-N)
      DOUBLE PRECISION URAND
      DIMENSION X(N)
      logical debug
      data debug /.false./
C
C                                  GENERATE THE UNIFORM SAMPLE
C
      DO 40 I=1,N
         X(I) = URAND(ISEED,IDUMMY)
         if(debug) write(6,'('' uniform x(i)='',f9.6)' ) x(i)
   40    CONTINUE
C                                  SORT VALUES INTO ASCENDING ORDER
      IF(ISORT.NE.0) THEN
         CALL SORT(N,X)
         if(debug)
     +      write(6,'('' sorted uniform x ='',f9.6)' ) (x(i),i=1,n)
      ENDIF
C                                  CONVERT UNIFORM TO NORMAL(0,1)
C                                  THEN CORRECT FOR MEAN AND ST. DEV.
      DO 100 I=1,N
         P = X(I)
         if(debug) write(6,'('' p ='',g14.6)' ) p
         Z = ANORIN(P,IER)
         IF(IER.GT.0) THEN
            if(debug) write(6,60) ier, i, p, z
   60       format(' ier, i, x, z =',2i4,2g14.6)
            IPOINT = I
            RETURN
         ENDIF
         X(I) = AMEAN + STDEV*Z
         if(debug) write(6,'('' normal x(i)='',f9.6)' ) x(i)
  100    CONTINUE
      END

*** ANORDF *****************************************************************

      FUNCTION ANORDF(X)

c     Calculates standard normal cumulative distribution function

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DATA RT2INV/.7071067812/
```

```
      U = ABS(X) * RT2INV
      IF(X.GT.0) THEN
         ANS = 1 - ERFC(U)/2
      ELSE
         ANS = ERFC(U)/2
      ENDIF
c     debug print
c     write(6,'( '' normal cdf('',g14.6,'') ='',g14.6)' ) x, ans
      ANORDF = ANS
      RETURN
      END

*** ANORIN ********************************************************************

      FUNCTION ANORIN(P,IER)

c     Evaluates inverse normal cdf PHI-inverse(p)
c     For p in tail, starts with Wichura approximation, then refines
c        it N times using eq. (5.9.2) of Thisted (1988) Elements of
c        Statistical Computing, Chapman and Hall.
c     For p in center, uses Beasley-Springer algorithm
      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
c     Phi(XMAX) is about as close to 1 as we can get in single precision
c     1 - Phi(XMAX) = 2.9E-7
      DATA XMAX/5./
      DATA N/1/
c     On return, IER = 1 signals input error, IER = 2 is serious error
      IER = 0

      IF(P.GE.1. .OR. P.LE.0.) THEN
         IER = 1
         IF(P.GT.1. .OR. P.LT.0) THEN
            IER = 2
            WRITE(6,10) p
   10       FORMAT(' Input error to ANORIN(',E14.6,',IER)' )
         ENDIF
         IF(P.LE.0) ANORIN = -XMAX
         IF(P.GE.1) ANORIN = XMAX
         RETURN
      ELSE IF(P.LT..1 .OR. P.GT..9) THEN
         Z = WICHUR(P)
         DO 50 I=1,N
            CDF = ANORDF(Z)
            ARG = 2*P - CDF
            Z = WICHUR(ARG)
   50       CONTINUE
         ANORIN = Z
      ELSE
         ANORIN = PPND(P,IFAULT)
         IF(IFAULT.NE.0) THEN
            WRITE(6,100) P
  100       FORMAT(' Error fault in ANORIN(',E14.6,',IER)' )
            IER = 2
         ENDIF
      ENDIF
```

```
      RETURN
      END

*** BINARY ****************************************************************

      SUBROUTINE BINARY(IY,KOEF,INTSIZ)

C     FINDS COEFFICIENTS OF BINARY EXPANSION OF IY
C
C     IY = SUM OF ( KOEF(I) *  2**(I-1) )
C
C     WHERE SUMMATION IS FOR I FROM 1 TO INTSIZ
C     Y MUST BE .GE. 0 AND .LT. 2**INTSIZ

      IMPLICIT INTEGER(I-N)
      DIMENSION KOEF(INTSIZ)

      N = IY
      DO 50 I=1,INTSIZ
         N2 = N/2
         KOEF(I) = N - (2*N2)
         N = N2
   50    CONTINUE
      IF(N.EQ.0) RETURN
C                                   ERROR
      WRITE(6,100) IY, INTSIZ, (KOEF(I),I=1,INTSIZ)
  100 FORMAT(' ERROR IN SUBROUTINE BINARY'/
     +       ' IY =',I20/' INTSIZ =',I5/' KOEF ='/(20I3) )
      STOP
      END

*** ERFC ******************************************************************

      FUNCTION ERFC(X)

c     Evaluates the complementary error function at X.
c     Uses an algorithm of Press et al., 1986, Numerical Recipes:
c        The Art of Scientific Computing, Cambridge Univ. Press.
c        as presented in Section 5.10.1.1 of Thisted, 1988, Elements
c        of Statistical Computing, Chapman and Hall.
c     Calculation has relative error < 1.2E-7 for X > 0.

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DIMENSION A(10)
      DATA A/-1.26551223,  1.00002368,  0.37409196,  0.09678418,
     +       -0.18628806,  0.27886807, -1.13520398,  1.48851587,
     +       -0.82215223,  0.17087277/

      ARG = 1 / (1 + X/2)
      PSUM = A(1)
      TERM = 1
      DO 20 I=2,10
         TERM = TERM * ARG
         PSUM = PSUM + TERM*A(I)
   20    CONTINUE
```

```
      EARG = -(X**2) + PSUM
      ERFC = ARG * EXP(EARG)
      RETURN
      END

*** PPND *****************************************************************

      FUNCTION PPND(P,IFAULT)

c     Algorithm AS 111 Applied Statistics, 1977, Vol. 26, No. 1.
c        by J. D. Beasley and S. G. Springer
c     Used for inverse of normal cdf, in middle portion of distr.

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DATA ZERO/0./, HALF/0.5/, ONE/1./
      DATA SPLIT/0.42/
      DATA A0 /  2.50662 82388 4/
      DATA A1 /-18.61500 06252 9/
      DATA A2 / 41.39119 77353 4/
      DATA A3 /-25.44106 04963 7/
      DATA B1 / -8.47351 09309 0/
      DATA B2 / 23.08336 74374 3/
      DATA B3 /-21.06224 10182 6/
      DATA B4 /  3.13082 90983 3/
C        HASH SUM AB 143.70383 55807 6
      DATA C0 / -2.78718 93113 8/
      DATA C1 / -2.29796 47913 4/
      DATA C2 /  4.85014 12713 5/
      DATA C3 /  2.32121 27685 8/
      DATA D1 /  3.54388 92476 2/
      DATA D2 /  1.63706 78189 7/
C        HASH SUM CD 17.43746 52092 4

      IFAULT = 0
      Q = P - HALF
      IF(ABS(Q) .LE. SPLIT) THEN
         R = Q*Q
         PPND = Q * (((A3 * R + A2) * R + A1) * R + A0) /
     +          ((((B4 * R + B3) * R + B2) * R + B1) * R + ONE)
      ELSE
         R = P
         IF(Q .GT. ZERO) R = ONE - P
         IF(R .LE. ZERO) GO TO 800
         R = SQRT(-LOG(R))
         PPND = (((C3 * R + C2) * R + C1) * R + C0) /
     +          ((D2 * R + D1) * R + ONE)
         IF(Q .LT. ZERO) PPND = -PPND
      ENDIF
c     debug prints
c     write(6,100) p,q,r,ppnd
c 100 format(' in ppnd, p, q, r, ppnd =',4g14.6)
      RETURN

  800 CONTINUE
      IFAULT = 1
```

```
      PPND = ZERO
      RETURN
      END

*** SORT **********************************************************************

      SUBROUTINE SORT(N,RA)

c     Implementation of the heapsort algorithm given in
c     Press et al., Numerical Recipes, Cambridge Univ. Press, 1986
c     On input, RA is unsorted.  On output, RA is in ascending order.

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DIMENSION RA(N)

      IF (N.LE.0) RETURN
      L=N/2+1
      IR=N
10    CONTINUE
        IF(L.GT.1)THEN
          L=L-1
          RRA=RA(L)
        ELSE
          RRA=RA(IR)
          RA(IR)=RA(1)
          IR=IR-1
          IF(IR.EQ.1)THEN
            RA(1)=RRA
            RETURN
          ENDIF
        ENDIF
        I=L
        J=L+L
20      IF(J.LE.IR)THEN
          IF(J.LT.IR)THEN
            IF(RA(J).LT.RA(J+1))J=J+1
          ENDIF
          IF(RRA.LT.RA(J))THEN
            RA(I)=RA(J)
            I=J
            J=J+J
          ELSE
            J=IR+1
          ENDIF
        GO TO 20
        ENDIF
        RA(I)=RRA
      GO TO 10
      END

*** URAND *********************************************************************

      DOUBLE PRECISION FUNCTION URAND(IY,M2RET)

C     UNIFORM RANDOM NUMBER GENERATOR, TAKEN FROM FORSYTHE, MALCOLM AND
```

```
C     MOLER (1977) 'COMPUTER METHODS FOR MATHEMATICAL COMPUTATIONS',
C     PRENTICE HALL.  IT IS BASED ON SUGGESTIONS BY KNUTH (1969).
C     M2RET=M2 IS RETURNED ON FIRST CALL, FOR USE IN ADVISING
C     USER AS TO ALLOWABLE SEEDS.  IF THIS FIRST CALL IS ONLY
C     TO FIND M2, USE ANY IY, FOR EXAMPLE 0.
C     IY SHOULD BE INITIALIZED TO AN ARBITRARY INTEGER PRIOR TO THE
C     FIRST CALL THAT SERIOUSLY WANTS A RANDOM NUMBER, AND IY
C     SHOULD NOT BE ALTERED BETWEEN SUBSEQUENT CALLS.
C     VALUES OF URAND WILL BE RETURNED IN THE INTERVAL (0,1).

      IMPLICIT INTEGER(I-N)
      IMPLICIT DOUBLE PRECISION (A-H,O-Z)
C     ON IBM PC, MAX INTEGER IS 2**15 - 1, WITH 4-BYTE INTEGERS,
C     SMALLER WITH 2-BYTE INTEGERS.
      PARAMETER(INTMAX=31)
      DIMENSION KOEFA(INTMAX), KOEFY(INTMAX)
      DATA M2/0/, ITWO/2/
      IF(M2 .NE. 0) GO TO 20

C                                   FIRST ENTRY
C     COMPUTE MACHINE WORD LENGTH
C     INTSIZ = NUMBER OF BITS IN HOST MACHINE INTEGER WORD
C              E.G. 31 IF INTEGER*4, 15 IF INTEGER*2
C     LARGEST POSSIBLE INTEGER IS (M2 - 1) + M2
      M = 1
      INTSIZ = 0
   10 M2 = M
      INTSIZ = INTSIZ + 1
      M = ITWO * M2
      IF(M .GT. M2) GO TO 10
      HALFM = M2
      M2RET = M2

      IF(INTSIZ.GT.INTMAX) THEN
C                                   ERROR IN DIMENSION
         II = INTMAX
         WRITE(6,15) II, INTSIZ
   15    FORMAT(' DIMENSIONS TOO SMALL IN FUNCTION URAND'/
     +          ' CHANGE THE STATEMENT'/
     +          '        PARAMETER(INTMAX=',I2,')'/
     +          ' TO THE STATEMENT'/
     +          '        PARAMETER(INTMAX=',I2,')'/
     +          ' AND RECOMPILE THE PROGRAM')
         STOP
      ENDIF

C     COMPUTE MULTIPLIER AND INCREMENT
C     FOR LINEAR CONGRUENTIAL METHOD
      IA = 8 * INT( HALFM * DATAN(1.D0) / 8.D0 ) + 5
      IC = 2 * INT( HALFM * (0.5D0 - DSQRT(3.D0)/6.D0 ) ) + 1
      MIC = (M2 - IC) + M2

C     FOR BRUTE FORCE MODULAR ARITHMETIC, FIND BINARY COEFFS FOR IA
      CALL BINARY(IA,KOEFA,INTSIZ)
```

```
C     S IS THE SCALE FACTOR FOR CONVERTING TO FLOATING POINT
      S = 0.5 / HALFM
C                                   COMPUTE THE NEXT RANDOM NUMBER
   20 CONTINUE

C     FIND IY = IY*IA (MOD 2**INTSIZ)
C     IF MACHINE TREATS ORDINARY INTEGER MULTIPLICATION OVERFLOW
C     BY TAKING REMAINDER (MOD 2**INTSIZ), REPLACE NEXT GROUP OF
C     STATEMENTS BY IY = IY * IA

      CALL BINARY(IY,KOEFY,INTSIZ)
      IANS = 0
      IMULT = 1
      DO 60 I=1,INTSIZ
         IF(I.GT.1) IMULT = IMULT * 2
         IF(KOEFA(I).EQ.0) GO TO 60
         JMULT = IMULT
         DO 40 J=1,INTSIZ+1-I
            IF(J.GT.1) JMULT = JMULT * 2
            IF(KOEFY(J).EQ.0) GO TO 40
            INEW = IANS + JMULT
            IF(INEW.LT.IANS) INEW = ((IANS-M2)-M2)+JMULT
            IANS = INEW
   40       CONTINUE
   60    CONTINUE
      IY = IANS

C     THE NEXT STATEMENT IS FOR COMPUTERS THAT DO NOT ALLOW
C     INTEGER OVERFLOW ON ADDITION
         IF(IY .GT. MIC) IY = (IY - M2) - M2
      IY = IY + IC

C     THE NEXT STATEMENT IS FOR COMPUTERS WHERE THE WORD LENGTH
C     FOR ADDITION IS GREATER THAN FOR MULTIPLICATION
      IF(IY/2 .GT. M2) IY = (IY - M2) - M2

C     THE NEXT STATEMENT IS FOR COMPUTERS WHERE INTEGER OVERFLOW
C     AFFECTS THE SIGN BIT
      IF(IY .LT. 0) IY = (IY + M2) + M2

      URAND = S * IY
      RETURN
      END

*** WICHUR *******************************************************************

      FUNCTION WICHUR(P)

c     Approximates PHI-inverse(p), the normal value corresponding to a
c        tail probability 1 - p.  If p > .9, it has at least 2-digit
c        accuracy.  Presented as Algorithm 5.10.1 (due to Wichura)
c        by Thisted (1988), Elements of Statistical Computing,
c        Chapman and Hall.

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
```

```
      DATA PI2/6.283185308/

      ISIGN = 1
      TAILP = 1 - P
      IF(TAILP.GT..5) THEN
         TAILP = P
         ISIGN = -1
      ENDIF
      V = -2 * LOG(TAILP)
      X = LOG(PI2*V)
      T = ( ( (-14 + 6*X - X**2) / (2*V) + (2-X)) / V + X) / V
      ANS = SQRT(V*(1-T))
c     debug print
c     write(6,'('' tail prob ='',g14.6)' ) tailp
c     write(6,'('' v,x,t ='',3g15.6)' ) v, x, t
c     write(6,'('' normal quantile ='',g14.6)' ) ans
      WICHUR = ISIGN * ANS
      RETURN
      END
```

ISORT,	N,	NDIS,	ISEED
0,	10000,	30,	1234567

IDIS,	MEAN,	STDEV
1,	1.0,	1.0
2,	1.0,	1.0
3,	7.415,	0.360
4,	7.441,	0.200
5,	7.305,	0.360
[illegible]	7.731,	0.200
7,	6.737,	0.360
8,	6.994,	0.360
9,	7.276,	0.200
10,	7.415,	0.360
11,	7.719,	0.120
12,	7.719,	0.120
13,	7.719,	0.120
14,	7.719,	0.200
15,	7.593,	0.360
16,	7.818,	0.120
17,	7.609,	0.200
18,	7.682,	0.200
19,	6.068,	0.120
20,	6.937,	0.230
21,	7.396,	0.270
22,	7.616,	0.230
23,	7.818,	0.120
24,	7.818,	0.120
25,	7.368,	0.360
26,	7.593,	0.360
27,	7.825,	0.360
28,	7.124,	0.200
29,	7.638,	0.360
30,	7.818,	0.120

LISTING 25

EVNTRE Key Word File for DHR Letdown

The following is a listing of the EVNTRE key word file for the DHRL model. The EVNTRE key word file controls the mode of execution, input and output options, and cutoff values used by the program during event tree evaluation.

```
$-- Calculation Control Keywords -------------------------------------------
$
    MODE 3                      $ Specifies the calculational mode for EVNTRE.
$                                      1 = point estimate
$                                      3 = sampling mode (one vector each eval)
$
$
    NOBIN                       $ Turns the binning facility on/off.
$
$
    RUN                         $ Indicates that the tree is to be evaluated
$                                 after the input data has been processed.
$
$   KEEPCUT 1.0E-6              $ Specifies the path frequency below which a
$                                 path is terminated.
$
$-- Input File Specification Keywords --------------------------------------
$
    TREEIN tree.dat             $ Specifies the input file name for the
$                                 tree definition input file.
$
$   BININ  bin.dat              $ Specifies the input file name for the
$                                 binning and sorting information input file.
$
    SAMDIN mc_pntr.dat          $ Specifies the input file name for the
$                                 sample definition information input file.
$
    SAM1IN mcarlo.dat           $ Specifies the input file name for the
$                                 first set of sample input vectors.
$
$   SAM2IN hcube.dat            $ Specifies the input file name for the
$                                 second set of sample input vectors.
$
$-- Report Request Keywords ------------------------------------------------
$
$   PRTINP                      $ Turns on the annotated echo of input.
$
    STATS                       $ Indicates that a branch and case frequency
$                                 table report will be generated.
$
    PRUNE                       $ Causes unused cases to be dropped from the
$                                 branch and case frequency table.
$
    NWRTBIN                     $ Indicates that a binning result report will
$                                 be generated when the paths through the
$                                 tree are binned.
$
```

```
$   PRTCUT 1.0E-6                  $ Specifies the minimum bin frequency required
$                                    to report a bin.
$
$   SAVEBIN                        $ Indicates that a binning results file will
$                                    be generated for post-processing.
$
$-- Output File Specification Keywords ---------------------------------------
$
$   INPOUT  echo.out               $ Specifies the output file name for the
$                                    annotated echo of input.
$
$   BINOUT  bin.out                $ Specifies the output file name for the
$                                    binning result report.
$
    STATOUT mc_freq.out            $ Specifies the output file name for the
$                                    branch and case frequency table.
$
    SAMROUT mc_post.out            $ Specifies the output file name for the
$                                    post-processing file.
$
ENDKEY                             $ Indicates the end of keyword input.
```

LISTING 26

EVNTRE Tree Definition File for DHR Letdown

The following is a listing of the EVNTRE event tree definition file for the DHRL model. This file provides the event tree structure and default probability and parameter values for the DHRL model.

```
ISLOCA System Rupture Model -- DHR Letdown
    57
    NQ
     1     1.000
        'MC Eval'
  1 What is the pressure in the Interfacing System?
     1  DH4849-P
     3         1
           1.000
    30                    $ Pt estimate pressure data follows:
     1     1.000          $      1Ps' for DH-4849
     2     1.000          $      1Ps' for DH-2734
     3     7.415          $      1Pf' for 12"-GCB-7
     4     7.441          $      1Pf' for DH-1517
     5     7.305          $      1Pf' for 18"-GCB-8
     6     7.731          $      1Pf' for DH-2733
     7     6.737          $      1Pf' for 18"-HCB-1
     8     6.994          $      1Pf' for 14"-HCB-1
     9     7.276          $      1Pf' for DH-81
    10     7.415          $      1Pf' for 12"-GCB-8
    11     7.719          $      1Pf' for 12GCB8a
    12     7.719          $      1Pf' for 12GCB8b
    13     7.719          $      1Pf' for 12GCB8c
    14     7.719          $      1Pf' for P42-1
    15     7.593          $      1Pf' for 10"-GCB-1
    16     7.818          $      1Pf' for 10GCB1a
    17     7.609          $      1Pf' for DH-43
    18     7.682          $      1Pf' for DH-45
    19     6.068          $      1Pf' for E271T
    20     6.937          $      1Pf' for E271P
    21     7.396          $      1Pf' for E271C
    22     7.616          $      1Pf' for E271A
    23     7.818          $      1Pf' for E271a
    24     7.818          $      1Pf' for E271b
    25     7.368          $      1Pf' for 6"-GCB-10
    26     7.593          $      1Pf' for 10"-GCB-10
    27     7.825          $      1Pf' for 8"-GCB-10
    28     7.124          $      1Pf' for DH-128
    29     7.638          $      1Pf' for 4"-GCB-2
    30     7.818          $      1Pf' for FE-DH2B
  2 Does 12"-GCB-7 pipe fail?  (12", sch 20, type 304 SS)
     2  12GCB7-F 12GCB7-NF
     5         1         2
     2         1         3
            1Ps'      1Pf'
         FUN-CMP
           EQUAL         0
```

```
      IF 1Ps' .GT. 1Pf' THEN 12GCB7-F
3 If 12"-GCB-7 pipe fails, what is the rupture size?
    2 12GCB7-Lg 12GCB7-NL
    2         1         2
    2
    1         2
              1
       12GCB7-F
          1.000     0.000
      Otherwise
          0.000     1.000
4 Does DH-1517 fail?  (12" MOGV, 300 psi rating)
    2  DH1517-F DH1517-NF
    5         1         2
    2         2         4
          1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN DH1517-F
5 How large is the leak at DH-1517?
    2 DH1517-Sm DH1517-NL
    5         1         2
    2         2         4
          1Ps'      1Pf'
       FUN-RPSZ
       GETHRESH         1     1.000
      Bin Ps'/Pf'
6 Does 18"-GCB-8 pipe fail?  (18", sch 20, type 304 SS)
    2  18GCB8-F 18GCB8-NF
    5         1         2
    2         2         5
          1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN 18GCB8-F
7 If 18"-GCB-8 pipe fails, what is the rupture size?
    2 18GCB8-Lg 18GCB8-NL
    2         1         2
    2
    1         6
              1
       18GCB8-F
          1.000     0.000
      Otherwise
          0.000     1.000
8 Does DH-2733 fail?  (18" MOGV, 300 psi rating)
    2  DH2733-F DH2733-NF
    5         1         2
    2         2         6
          1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN DH2733-F
9 If DH-2733 fails, what is the rupture size?
    2 DH2733-Sm DH2733-NL
```

```
      5          1          2
      2          2          6
             1Ps'       1Pf'
          FUN-RPSZ
          GETHRESH          1      1.000
         Bin Ps'/Pf'
10 Does 18"-HCB-1 fail?  (18" pipe, sch 10S)
      2  18HCB1-F 18HCB1-NF
      5          1          2
      2          2          7
             1Ps'       1Pf'
           FUN-CMP
             EQUAL          0
         IF 1Ps' .GT. 1Pf' THEN 18HCB1-F
11 How large is the leak at 18"-HCB-1?
      2 18HCB1-Lg 18HCB1-NL
      2          1          2
      2
      1         10
                 1
          18HCB1-F
             1.000      0.000
         Otherwise
             0.000      1.000
12 Does 14"-HCB-1 fail?  (14" pipe, sch. 10S)
      2  14HCB1-F 14HCB1-NF
      5          1          2
      2          2          8
             1Ps'       1Pf'
           FUN-CMP
             EQUAL          0
         IF 1Ps' .GT. 1Pf' THEN 14HCB1-F
13 How large is the leak at 14"-HCB-1?
      2 14HCB1-Lg 14HCB1-NL
      2          1          2
      2
      1         12
                 1
          14HCB1-F
             1.000      0.000
         Otherwise
             0.000      1.000
14 Does DH-81 fail?  (14" SwCV, 150 psi rating)
      2     DH81-F  DH81-NoF
      5          1          2
      2          2          9
             1Ps'       1Pf'
           FUN-CMP
             EQUAL          0
         IF 1Ps' .GT. 1Pf' THEN DH81-F
15 How large is the leak at DH-81?
      2    DH81-Sm  DH81-NoL
      5          1          2
      2          2          9
             1Ps'       1Pf'
```

```
       FUN-RPSZ
       GETHRESH          1     1.000
      Bin Ps'/Pf'
16 Does 12"-GCB-8 fail?  (12" pipe, sch. 20)
    2  12GCB8-F 12GCB8-NF
    5         1         2
    2         2        10
          1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN 12GCB8-F
17 How large is the leak at 12"-GCB-8?
    2 12GCB8-Lg 12GCB8-NL
    2         1         2
    2
    1        16
              1
       12GCB8-F
          1.000     0.000
      Otherwise
          0.000     1.000
1? Does 12GCB8a fail?  (12" flange, 300 psi rating)
    2  12GCBa-F 12GCBa-NF
    5         1         2
    2         2        11
          1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN 12GCBa-F
19 How large is the leak at 12GCB8a?
    2 12GCBa-Lg 12GCBa-NL
    2         1         2
    2
    1        18
              1
       ?GCBa-F
          1.000     0.000
      Otherwise
          0.000     1.000
20 Does 12GCB8b fail?  (12" flange, 300 psi)
    2  12GCBb-F 12GCBb-NF
    5         1         2
    2         2        12
          1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN 12GCBb-F
21 If 12GCB8b fails, what is the rupture size?
    2 12GCBb-Lg 12GCBb-NL
    2         1         2
    2
    1        20
              1
       12GCBb-F
          1.000     0.000
```

```
        Otherwise
            0.000      1.000
22 Does 12GCB8c fail?  (12" flange, 300 psi)
     2  12GCBc-F 12GCBc-NF
     5         1         2
     2         2        13
             1Ps'      1Pf'
          FUN-CMP
            EQUAL        0
        IF 1Ps' .GT. 1Pf' THEN 12GCBc-F
23 If 12GCB8c fails, what is the rupture size?
     2 12GCBc-Lg 12GCBc-NL
     2         1         2
     2
     1        22
               1
         12GCBc-F
            1.000      0.000
        Otherwise
            0.000      1.000
24 Does P42-1 fail?  (DHR pump 1-1)
     2    P421-F  P421-NoF
     5         1         2
     2         2        14
             1Ps'      1Pf'
          FUN-CMP
            EQUAL        0
        IF 1Ps' .GT. 1Pf' THEN P421-F
25 If P42-1 fails, what is the rupture size?
     2   P421-Sm  P421-NoL
     5         1         2
     2         2        14
             1Ps'      1Pf'
          FUN-RPSZ
          GETHRESH          1     1.000
        Bin Ps'/Pf'
26 Does 10"-GCB-1 fail?  (10" pipe, sch. 20)
     2  10GCB1-F 10GCB1-NF
     5         1         2
     2         2        15
             1Ps'      1Pf'
          FUN-CMP
            EQUAL        0
        IF 1Ps' .GT. 1Pf' THEN 10GCB1-F
27 How large is the leak at 10"-GCB-1?
     2 10GCB1-Lg 10GCB1-NL
     2         1         2
     2
     1        26
               1
         10GCB1-F
            1.000      0.000
        Otherwise
            0.000      1.000
28 Does 10GCB1a fail?  (10" flange, 300 psi rating)
```

```
    2  1GCB1a-F 1GCB1a-NF
    5         1         2
    2         2        16
           1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN 1GCB1a-F
29 If 1OGCB1a fails, what is the rupture size?
    2 1GCB1a-Lg 1GCB1a-NL
    2         1         2
    2
    1        28
              1
       1GCB1a-F
          1.000     0.000
      Otherwise
          0.000     1.000
30 Does DH-43 fail?  (10" SwCV, 300 psi rating)
    2    DH43-F  DH43-NoF
    5         1         2
    2         2        17
           1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN DH43-F
31 If DH-43 fails, what is the rupture size?
    2   DH43-Sm  DH43-NoL
    5         1         2
    2         2        17
           1Ps'      1Pf'
        FUN-RPSZ
        GETHRESH           1     1.000
      Bin Ps'/Pf'
32 Does DH-45 fail?  (10" HWGV, 300 psi rating)
    2    DH45-F  DH45-NoF
    5         1         2
    2         2        18
           1Ps'      1Pf'
        FUN-CMP
          EQUAL         0
      IF 1Ps' .GT. 1Pf' THEN DH45-F
33 If DH-45 fails, what is the rupture size?
    2   DH45-Sm  DH45-NoL
    5         1         2
    2         2        18
           1Ps'      1Pf'
        FUN-RPSZ
        GETHRESH           1     1.000
      Bin Ps'/Pf'
34 Does E271T fail?  (DHR hx tube sheet flg)
    2   E271T-F E271T-NoF
    5         1         2
    2         2        19
           1Ps'      1Pf'
        FUN-CMP
```

```
        EQUAL          0
      IF 1Ps' .GT. 1Pf' THEN E271T-F
35 If E271T fails, what is the rupture size?
    3  E271T-Lg  E271T-Sm E271T-NoL
    5          1         2         3
    2          2        19
         1Ps'      1Pf'
       FUN-RPSZ
       GETHRESH          2     2.067      1.000
      Bin Ps'/Pf'
36 Does E271P fail?  (DHR hx plastic col)
    2   E271P-F E271P-NoF
    5          1         2
    2          2        20
         1Ps'      1Pf'
       FUN-CMP
         EQUAL           0
      IF 1Ps' .GT. 1Pf' THEN E271P-F
37 If E271P fails, what is the rupture size?
    2  E271P-Lg E271P-NoL
    2          1         2
    2
    1         36
               1
       E271P-F
         0.200     0.800
      Otherwise
         0.000     1.000
38 Does E271C fail?  (DHR hx cylinder rupture)
    2   E271C-F E271C-NoF
    5          1         2
    2          2        21
         1Ps'      1Pf'
       FUN-CMP
         EQUAL           0
      IF 1Ps' .GT. 1Pf' THEN E271C-F
39 If E271C fails, what is the rupture size?
    2  E271C-Lg E271C-NoL
    2          1         2
    2
    1         38
               1
       E271C-F
         1.000     0.000
      Otherwise
         0.000     1.000
40 Does E271A fail?  (DHR hx asym. head buckling)
    2   E271A-F E271A-NoF
    5          1         2
    2          2        22
         1Ps'      1Pf'
       FUN-CMP
         EQUAL           0
      IF 1Ps' .GT. 1Pf' THEN E271A-F
41 If E271A fails, what is the rupture size?
```

```
      2  E271A-Sm E271A-NcL
      2         1         2
      2
      1        40
                1
          E271A-F
            0.200     0.800
        Otherwise
            0.000     1.000
42 Does E271a fail?  (10" outlet flange, 300 psi rating)
      2   E271a-F E271a-NoF
      5         1         2
      2         2        23
              1Ps'      1Pf'
          FUN-CMP
            EQUAL         0
        IF 1Ps' .GT. 1Pf' THEN E271a-F
43 If E271a fails, what is the rupture size?
      2  E271a-Lg E271a-NoL
      2         1         2
      2
      1        42
                1
          E271a-F
            1.000     0.000
        Otherwise
            0.000     1.000
44 Does E271b fail?  (10" inlet flange, 300 psi rating)
      2   E271b-F E271b-NoF
      5         1         2
      2         2        24
              1Ps'      1Pf'
          FUN-CMP
            EQUAL         0
        IF 1Ps' .GT. 1Pf' THEN E271b-F
45 If E271b fails, what is the rupture size?
      2  E271b-Lg E271b-NoL
      2         1         2
      2
      1        44
                1
          E271b-F
            1.000     0.000
        Otherwise
            0.000     1.000
46 Does 6"-GCB-10 fail?  (6" pipe, sch. 10S)
      2  6GCB10-F 6GCB10-NF
      5         1         2
      2         2        25
              1Ps'      1Pf'
          FUN-CMP
            EQUAL         0
        IF 1Ps' .GT. 1Pf' THEN 6GCB10-F
47 If 6"-GCB-10 fails, what is the rupture size?
     2 6GCB10-Lg 6GCB10-NL
```

```
     2          1         2
     2
     1         46
                1
         6GCB10-F
            1.000      0.000
        Otherwise
            0.000      1.000
48 Does 10"-GCB-10 fail?  (10" pipe, sch. 20)
     2   1GCB10-F 1GCB10-NF
     5          1         2
     2          2        26
              1Ps'      1Pf'
           FUN-CMP
             EQUAL        0
        IF 1Ps' .GT. 1Pf' THEN 1GCB10-F
49 If 10"-GCB-10 fails, what is the rupture size?
     2 1GCB10-Lg 1GCB10-NL
     2          1         2
     2
     1         48
                1
         1GCB10-F
            1.000      0.000
        Otherwise
            0.000      1.000
50 Does 8"-GCB-10 fail?  (8" pipe, sch. 20)
     2  8GCB10-F 8GCB10-NF
     5          1         2
     2          2        27
              1Ps'      1Pf'
           FUN-CMP
             EQUAL        0
        IF 1Ps' .GT. 1Pf' THEN 8GCB10-F
51 If 8"-GCB-10 fails, what is the rupture size?
     2 8GCB10-Lg 8GCB10-NL
     2          1         2
     2
     1         50
                1
         8GCB10-F
            1.000      0.000
        Otherwise
            0.000      1.000
52 Does DH-128 fail?  (8" SwCV  300 psi rating)
     2   DH128-F DH128-NoF
     5          1         2
     2          2        28
              1Ps'      1Pf'
           FUN-CMP
             EQUAL        0
        IF 1Ps' .GT. 1Pf' THEN DH128-F
53 If DH-128 fails, what is the rupture size?
     2  DH128-Sm DH128-NoL
     5          1         2
```

```
     2          2         28
           1Ps'       1Pf'
        FUN-RPSZ
        GETHRESH          1      1.000
       Bin Ps'/Pf'
54 Does 4"-GCB-2 fail?  (4" pipe, sch. 10S)
     2   4GCB2-F 4GCB2-NoF
     5         1          2
     2         2         29
           1Ps'       1Pf'
         FUN-CMP
           EQUAL          0
       IF 1Ps' .GT. 1Pf' THEN 4GCB2-F
55 If 4"-GCB-2 fails, what is the rupture size?
     2  4GCB2-Lg 4GCB2-NoL
     2         1          2
     2
     1        54
               1
         4GCB2-F
           1.000      0.000
       Otherwise
           0.000      1.000
56 Does FE-DH2B fail?  (10" FE, 300 psi rating)
     2    DH2B-F  DH2B-NoF
     5         1          2
     2         2         30
           1Ps'       1Pf'
         FUN-CMP
           EQUAL          0
       IF 1Ps' .GT. 1Pf' THEN DH2B-F
57 If FE-DH2B fails, what is the rupture size?
     2   DH2B-Lg  DH2B-NoL
     2         1          2
     2
     1        56
               1
          DH2B-F
           1.000      0.000
       Otherwise
           0.000      1.000
```

LISTING 27

EVNTRE Sample Definition File for DHR Letdown

The following is a listing of the EVNTRE sample definition data file for the DHRL model. This file supplies the specifications required to set up the sampling modes for the tree.

```
ISLOCA SAMPLE RUN
10000     1
30
M1,1, 1,1,A  M1,1, 2,1,A  M1,1, 3,1,A  M1,1, 4,1,A  M1,1, 5,1,A
M1,1, 6,1,A  M1,1, 7,1,A  M1,1, 8,1,A  M1,1, 9,1,A  M1,1,10,1,A
M1,1,11,1,A  M1,1,12,1,A  M1,1,13,1,A  M1,1,14,1,A  M1,1,15,1,A
M1,1,16,1,A  M1,1,17,1,A  M1,1,18,1,A  M1,1,19,1,A  M1,1,20,1,A
M1,1,21,1,A  M1,1,22,1,A  M1,1,23,1,A  M1,1,24,1,A  M1,1,25,1,A
M1,1,26,1,A  M1,1,27,1,A  M1,1,28,1,A  M1,1,29,1,A  M1,1,30,1,A
```

LISTING 28

EVNTRE Binning Data File for DHR Letdown

The following is a listing of the EVNTRE binning data file for the DHRL model. These data specify the logic used to select DHRL event tree endstates that are included in each system failure mode bin.

```
ISLOCA Component Failure Binning -- DHR Letdown
  1         FSize
  3   3    NoLeak     Small     Large
 28   1   3   5   7   9  11  13  15  17  19  21  23  25  27  29  31  33  35  37
         39  41  43  45  47  49  51  53  55  57
          2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 3 * 2
        * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2
       NoLeak     ==>
 28   2   5   9  15  25  31  33  35  41  53   3   7  11  13  17  19  21  23  27
         29  37  39  43  45  47  49  51  55  57
         (1 + 1 + 1 + 1 + 1 + 1 + 2 + 1 + 1)* 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2
        * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2 * 2
       Small Only ==>
 20   3   3   7  11  13  17  19  21  23  27  29  35  37  39  43  45  47  49  51
         55  57
          1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1 + 1
        + 1 + 1
       Large      ==>
```

LISTING 20

EVNTRE Frequency Output File for DHR Letdown

The following is a listing of the EVNTRE frequency output file for the DHRL model. This file contains the individual component failure mode probabilities resulting from the Monte Carlo evaluation of the DHRL model.

```
          TREE ID: ISLOCA System Rupture Model -- DHR Letdown
   # OF QUESTIONS:   57
     OBSERVATIONS: 10000
       FOR SERIES: ISLOCA SAMPLE RUN
      SEQUENCE ID:           MC Eval

********  QUESTION:    1  What is the pressure in the Interfacing System?

Q-TYPE/TIMES ASKED:        INDEP. INPUT PROB. INPUT PARM.           10000
          BRANCHES:            DH4849-P
                                  1
    REALIZED SPLIT:           1.000E+00

********  QUESTION:    2  Does 12"-GCB-7 pipe fail?  (12", sch 20, type 304 SS)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                       10000
          BRANCHES:            12GCB7-F  12GCB7-NF
                                  1         2
    REALIZED SPLIT:           2.553E-01 7.447E-01

********  QUESTION:    3  If 12"-GCB-7 pipe fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                         10000
          BRANCHES:            12GCB7-Lg 12GCB7-NL
                                  1         2
    REALIZED SPLIT:           2.553E-01 7.447E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     2.553E-01
      DEPENDENCIES:    2

     REQ. BRANCHES:    1

       DESCRIPTION:  12GCB7-F

 CASE/BRANCH SPLIT:           2.553E-01 0.000E+00

 CASE NUMBER/SPLIT:    2     7.447E-01
       DESCRIPTION:          Otherwise

 CASE/BRANCH SPLIT:           0.000E+00 7.447E-01
```

```
******** QUESTION:    4  Does DH-1517 fail?  (12" MOGV, 300 psi rating)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                        10000
        BRANCHES:            DH1517-F  DH1517-NF
                                1         2
   REALIZED SPLIT:           1.300E-02 9.870E-01

******** QUESTION:    5  How large is the leak at DH-1517?

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                        10000
        BRANCHES:            DH1517-Sm DH1517-NL
                                1         2
   REALIZED SPLIT:           1.300E-02 9.870E-01

******** QUESTION:    6  Does 18"-GCB-8 pipe fail?  (18", sch 20, type 304 SS)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                        10000
        BRANCHES:            18GCB8-F  18GCB8-NF
                                1         2
   REALIZED SPLIT:           1.072E-01 8.928E-01

******** QUESTION:    7  If 18"-GCB-8 pipe fails, what is the rupture size?

Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                          10000
        BRANCHES:            18GCB8-Lg 18GCB8-NL
                                1         2
   REALIZED SPLIT:           1.072E-01 8.928E-01

                  SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1     1.072E-01
     DEPENDENCIES:    6

    REQ. BRANCHES:    1

      DESCRIPTION:  18GCB8-F

 CASE/BRANCH SPLIT:          1.072E-01 0.000E+00

 CASE NUMBER/SPLIT:   2     8.928E-01
      DESCRIPTION:         Otherwise

 CASE/BRANCH SPLIT:          0.000E+00 8.928E-01

******** QUESTION:    8  Does DH-2733 fail?  (18" MOGV, 300 psi rating)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                        10000
        BRANCHES:            DH2733-F  DH2733-NF
                                1         2
   REALIZED SPLIT:           5.000E-04 9.995E-01
```

```
********  QUESTION:    9  If DH-2733 fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          10000
         BRANCHES:             DH2733-Sm DH2733-NL
                                   1         2
    REALIZED SPLIT:            5.000E-04 9.995E-01

********  QUESTION:   10  Does 18"-HCB-1 fail?  (18" pipe, sch 10S)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          10000
         BRANCHES:             18HCB1-F  18HCB1-NF
                                   1         2
    REALIZED SPLIT:            4.470E-01 5.530E-01

********  QUESTION:   11  How large is the leak at 18"-HCB-1?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                            10000
         BRANCHES:             18HCB1-Lg 18HCB1-NL
                                   1         2
    REALIZED SPLIT:            4.470E-01 5.530E-01

                        SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1      4.470E-01
      DEPENDENCIES:    10

     REQ. BRANCHES:     1

       DESCRIPTION:  18HCB1-F

  CASE/BRANCH SPLIT:           4.470E-01 0.000E+00

  CASE NUMBER/SPLIT:    2      5.530E-01
       DESCRIPTION:           Otherwise

  CASE/BRANCH SPLIT:           0.000E+00 5.530E-01

********  QUESTION:   12  Does 14"-HCB-1 fail?  (14" pipe, sch. 10S)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                          10000
         BRANCHES:             14HCB1-F  14HCB1-NF
                                   1         2
    REALIZED SPLIT:            2.695E-01 7.305E-01

********  QUESTION:   13  How large is the leak at 14"-HCB-1?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                            10000
         BRANCHES:             14HCB1-Lg 14HCB1-NL
                                   1         2
    REALIZED SPLIT:            2.695E-01 7.305E-01
```

SUMMARY BY CASE

```
CASE NUMBER/SPLIT:   1      2.695E-01
     DEPENDENCIES:   12

    REQ. BRANCHES:   1

      DESCRIPTION:  14HCB1-F

CASE/BRANCH SPLIT:          2.695E-01 0.000E+00

CASE NUMBER/SPLIT:   2      7.305E-01
      DESCRIPTION:         Otherwise

CASE/BRANCH SPLIT:          0.000E+00 7.305E-01

********  QUESTION:   14  Does DH-81 fail?  (14" SwCV, 150 psi rating)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                       10000
          BRANCHES:           DH81-F    DH81-NoF
                                1          2
   REALIZED SPLIT:          6.750E-02 9.325E-01

********  QUESTION:   15  How large is the leak at DH-81?

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                       10000
          BRANCHES:           DH81-Sm   DH81-NoL
                                1          2
   REALIZED SPLIT:          6.750E-02 9.325E-01

********  QUESTION:   16  Does 12"-GCB-8 fail?  (12" pipe, sch. 20)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                       10000
          BRANCHES:          12GCB8-F  12GCB8-NF
                                1          2
   REALIZED SPLIT:          7.120E-02 9.288E-01

********  QUESTION:   17  How large is the leak at 12"-GCB-8?

Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                         10000
          BRANCHES:          12GCB8-Lg 12GCB8-NL
                                1          2
   REALIZED SPLIT:          7.120E-02 9.288E-01
```

SUMMARY BY CASE

```
CASE NUMBER/SPLIT:   1      7.120E-02
     DEPENDENCIES:   16

    REQ. BRANCHES:   1
```

DESCRIPTION: 12GCB8-F

CASE/BRANCH SPLIT: 7.120E-02 0.000E+00

CASE NUMBER/SPLIT: 2 9.288E-01
DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 9.288E-01

******** QUESTION: 18 Does 12GCB8a fail? (12" flange, 300 psi rating)

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 10000
BRANCHES: 12GCBa-F 12GCBa-NF
1 2
REALIZED SPLIT: 0.000E+00 1.000E+00

******** QUESTION: 19 How large is the leak at 12GCB8a?

Q-TYPE/TIMES ASKED: DEP. INPUT PROB. 10000
BRANCHES: 12GCBa-Lg 12GCBa-NL
1 2
REALIZED SPLIT: 0.000E+00 1.000E+00

SUMMARY BY CASE

CASE NUMBER/SPLIT: 2 1.000E+00
DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 1.000E+00

******** QUESTION: 20 Does 12GCB8b fail? (12" flange, 300 psi)

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 10000
BRANCHES: 12GCBb-F 12GCBb-NF
1 2
REALIZED SPLIT: 0.000E+00 1.000E+00

******** QUESTION: 21 If 12GCB8b fails, what is the rupture size?

Q-TYPE/TIMES ASKED: DEP. INPUT PROB. 10000
BRANCHES: 12GCBb-Lg 12GCBb-NL
1 2
REALIZED SPLIT: 0.000E+00 1.000E+00

SUMMARY BY CASE

CASE NUMBER/SPLIT: 2 1.000E+00
DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 1.000E+00

******** QUESTION: 22 Does 12GCB8c fail? (12" flange, 300 psi)

```
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                        10000
          BRANCHES:          12GCBc-F  12GCBc-NF
                                1          2
    REALIZED SPLIT:         0.000E+00 1.000E+00
```

******** QUESTION: 23 If 12GCB8c fails, what is the rupture size?

```
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                          10000
          BRANCHES:          12GCBc-Lg 12GCBc-NL
                                1          2
    REALIZED SPLIT:         0.000E+00 1.000E+00

                    SUMMARY BY CASE

  CASE NUMBER/SPLIT:    2      1.000E+00
        DESCRIPTION:          Otherwise

  CASE/BRANCH SPLIT:           0.000E+00 1.000E+00
```

******** QUESTION: 24 Does P42-1 fail? (DHR pump 1-1)

```
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                        10000
          BRANCHES:          P421-F    P421-NoF
                                1          2
    REALIZED SPLIT:         3.000E-04 9.997E-01
```

******** QUESTION: 25 If P42-1 fails, what is the rupture size?

```
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                        10000
          BRANCHES:          P421-Sm   P421-NoL
                                1          2
    REALIZED SPLIT:         3.000E-04 9.997E-01
```

******** QUESTION: 26 Does 10"-GCB-1 fail? (10" pipe, sch. 20)

```
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                        10000
          BRANCHES:          10GCB1-F  10GCB1-NF
                                1          2
    REALIZED SPLIT:         3.150E-02 9.685E-01
```

******** QUESTION: 27 How large is the leak at 10"-GCB-1?

```
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                          10000
          BRANCHES:          10GCB1-Lg 10GCB1-NL
                                1          2
    REALIZED SPLIT:         3.150E-02 9.685E-01

                    SUMMARY BY CASE
```

```
  CASE NUMBER/SPLIT:    1     3.150E-02
       DEPENDENCIES:   26

      REQ. BRANCHES:    1

        DESCRIPTION:  10GCB1-F

  CASE/BRANCH SPLIT:            3.150E-02 0.000E+00

  CASE NUMBER/SPLIT:    2     9.685E-01
        DESCRIPTION:          Otherwise

  CASE/BRANCH SPLIT:            0.000E+00 9.685E-01

********  QUESTION:   28  Does 10GCB1a fail?  (10" flange, 300 psi rating)

 Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                      10000
           BRANCHES:            1GCB1a-F  1GCB1a-NF
                                   1         2
     REALIZED SPLIT:            0.000E+00 1.000E+00

********  QUESTION:   29  If 10GCB1a fails, what is the rupture size?

 Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                        10000
           BRANCHES:            1GCB1a-Lg 1GCB1a-NL
                                   1         2
     REALIZED SPLIT:            0.000E+00 1.000E+00

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:    2     1.000E+00
        DESCRIPTION:          Otherwise

  CASE/BRANCH SPLIT:            0.000E+00 1.000E+00

********  QUESTION:   30  Does DH-43 fail?  (10" SwCV, 300 psi rating)

 Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                      10000
           BRANCHES:            DH43-F    DH43-NoF
                                   1         2
     REALIZED SPLIT:            2.500E-03 9.975E-01

********  QUESTION:   31  If DH-43 fails, what is the rupture size?

 Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                      10000
           BRANCHES:            DH43-Sm   DH43-NoL
                                   1         2
     REALIZED SPLIT:            2.500E-03 9.975E-01
```

```
********  QUESTION:   32  Does DH-45 fail?  (10" HWGV, 300 psi rating)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                         10000
         BRANCHES:           DH45-F    DH45-NoF
                               1          2
   REALIZED SPLIT:         9.000E-04 9.991E-01

********  QUESTION:   33  If DH-45 fails, what is the rupture size?

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                         10000
         BRANCHES:           DH45-Sm   DH45-NoL
                               1          2
   REALIZED SPLIT:         9.000E-04 9.991E-01

********  QUESTION:   34  Does E271T fail?  (DHR hx tube sheet flg)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                         10000
         BRANCHES:           E271T-F   E271T-NoF
                               1          2
   REALIZED SPLIT:         8.546E-01 1.454E-01

********  QUESTION:   35  If E271T fails, what is the rupture size?

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                         10000
         BRANCHES:           E271T-Lg  E271T-Sm  E271T-NoL
                               1          2          3
   REALIZED SPLIT:         4.272E-01 4.274E-01 1.454E-01

********  QUESTION:   36  Does E271P fail?  (DHR hx plastic col)

Q-TYPE/TIMES ASKED:       INDEP. CALC. PROB.                         10000
         BRANCHES:           E271P-F   E271P-NoF
                               1          2
   REALIZED SPLIT:         2.994E-01 7.006E-01

********  QUESTION:   37  If E271P fails, what is the rupture size?

Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                           12994
         BRANCHES:           E271P-Lg  E271P-NoL
                               1          2
   REALIZED SPLIT:         5.988E-02 9.401E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1     2.994E-01
     DEPENDENCIES:   36

    REQ. BRANCHES:    1
```

```
      DESCRIPTION:  E271P-F

 CASE/BRANCH SPLIT:          5.988E-02 2.395E-01

 CASE NUMBER/SPLIT:    2     7.006E-01
      DESCRIPTION:          Otherwise

 CASE/BRANCH SPLIT:          0.000E+00 7.006E-01


******** QUESTION:   38  Does E271C fail?  (DHR hx cylinder rupture)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                     12994
         BRANCHES:             E271C-F   E271C-NoF
                                 1          2
    REALIZED SPLIT:          4.480E-02 9.552E-01


******** QUESTION:   39  If E271C fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                       12994
         BRANCHES:             E271C-Lg  E271C-NoL
                                 1          2
    REALIZED SPLIT:          4.480E-02 9.552E-01

                        SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     4.480E-02
     DEPENDENCIES:    38

     REQ BRANCHES:     1

      DESCRIPTION:  E271C-F

 CASE/BRANCH SPLIT:          4.480E-02 0.000E+00

 CASE NUMBER/SPLIT:    2     9.552E-01
      DESCRIPTION:          Otherwise

 CASE/BRANCH SPLIT:          0.000E+00 9.552E-01


******** QUESTION:   40  Does E271A fail?  (DHR hx asym. head buckling)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                     12994
         BRANCHES:             E271A-F   E271A-NoF
                                 1          2
    REALIZED SPLIT:          4.600E-03 9.954E-01


******** QUESTION:   41  If E271A fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                       13078
         BRANCHES:             E271A-Sm  E271A-NoL
                                 1          2
```

```
REALIZED SPLIT:          9.200E-04 9.991E-01

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     4.600E-03
     DEPENDENCIES:   40

    REQ. BRANCHES:    1

      DESCRIPTION:  E271A-F

CASE/BRANCH SPLIT:          9.200E-04 3.680E-03

CASE NUMBER/SPLIT:    2     9.954E-01
      DESCRIPTION:         Otherwise

CASE/BRANCH SPLIT:          0.000E+00 9.954E-01

********  QUESTION:   42  Does E271a fail?  (10" outlet flange, 300 psi rating)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                        13078
          BRANCHES:              E271a-F   E271a-NoF
                                  1          2
   REALIZED SPLIT:           0.000E+00 1.000E+00

********  QUESTION:   43  If E271a fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          13078
          BRANCHES:              E271a-Lg  E271a-NoL
                                  1          2
   REALIZED SPLIT:           0.000E+00 1.000E+00

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    2     1.000E+00
      DESCRIPTION:         Otherwise

CASE/BRANCH SPLIT:          0.000E+00 1.000E+00

********  QUESTION:   44  Does E271b fail?  (10" inlet flange, 300 psi rating)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                        13078
          BRANCHES:              E271b-F   E271b-NoF
                                  1          2
   REALIZED SPLIT:           0.000E+00 1.000E+00

********  QUESTION:   45  If E271b fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          13078
          BRANCHES:              E271b-Lg  E271b-NoL
                                  1          2
```

```
     REALIZED SPLIT:          0.000E+00 1.000E+00

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:    2     1.000E+00
        DESCRIPTION:         Otherwise

  CASE/BRANCH SPLIT:          0.000E+00 1.000E+00


******** QUESTION:   46  Does 6"-GCB-10 fail?  (6" pipe, sch. 10S)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                     13078
          BRANCHES:            6GCB10-F  6GCB10-NF
                                  1         2
     REALIZED SPLIT:          8.220E-02 9.178E-01


******** QUESTION:   47  If 6"-GCB-10 fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                       13078
          BRANCHES:            6GCB10-Lg 6GCB10-NL
                                  1         2
     REALIZED SPLIT:          8.220E-02 9.178E-01

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1     8.220E-02
       DEPENDENCIES:   46

      REQ. BRANCHES:    1

        DESCRIPTION:  6GCB10-F

  CASE/BRANCH SPLIT:          8.220E-02 0.000E+00

  CASE NUMBER/SPLIT:    2     9.178E-01
        DESCRIPTION:         Otherwise

  CASE/BRANCH SPLIT:          0.000E+00 9.178E-01


******** QUESTION:   48  Does 10"-GCB-10 fail?  (10" pipe, sch. 20)

Q-TYPE/TIMES ASKED:        INDEP. CALC. PROB.                     13078
          BRANCHES:            1GCB10-F  1GCB10-NF
                                  1         2
     REALIZED SPLIT:          2.950E-02 9.705E-01


******** QUESTION:   49  If 10"-GCB-10 fails, what is the rupture size?

Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                       13078
          BRANCHES:            1GCB10-Lg 1GCB10-NL
                                  1         2
```

REALIZED SPLIT: 2.950E-02 9.705E-01

SUMMARY BY CASE

CASE NUMBER/SPLIT: 1 2.950E-02
DEPENDENCIES: 48

REQ. BRANCHES: 1

DESCRIPTION: 1GCB10-F

CASE/BRANCH SPLIT: 2.950E-02 0.000E+00

CASE NUMBER/SPLIT: 2 9.705E-01
DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 9.705E-01

******** QUESTION: 50 Does 8"-GCB-10 fail? (8" pipe, sch. 20)

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 13078
BRANCHES: 8GCB10-F 8GCB10-NF
1 2
REALIZED SPLIT: 7.300E-03 9.927E-01

******** QUESTION: 51 If 8"-GCB-10 fails, what is the rupture size?

Q-TYPE/TIMES ASKED: DEP. INPUT PROB. 13078
BRANCHES: 8GCB10-Lg 8GCB10-NL
1 2
REALIZED SPLIT: 7.300E-03 9.927E-01

SUMMARY BY CASE

CASE NUMBER/SPLIT: 1 7.300E-03
DEPENDENCIES: 50

REQ. BRANCHES: 1

DESCRIPTION: 8GCB10-F

CASE/BRANCH SPLIT: 7.300E-03 0.000E+00

CASE NUMBER/SPLIT: 2 9.927E-01
DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 9.927E-01

******** QUESTION: 52 Does DH-128 fail? (8" SwCV, 300 psi rating)

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 13078
BRANCHES: DH128-F DH128-NoF

```
                                1         2
   REALIZED SPLIT:          1.419E-01 8.581E-01

******** QUESTION:   53  If DH-128 fails, what is the rupture size?

Q-TYPE/TIMES ASKED:         INDEP. CALC. PROB.                        13078
          BRANCHES:            DH128-Sm  DH128-NoL
                                1         2
   REALIZED SPLIT:          1.420E-01 8.580E-01

******** QUESTION:   54  Does 4"-GCB-2 fail?  (4" pipe, sch. 10S)

Q-TYPE/TIMES ASKED:         INDEP. CALC. PROB.                        13078
          BRANCHES:            4GCB2-F   4GCB2-NoF
                                1         2
   REALIZED SPLIT:          2.200E-02 9.780E-01

******** QUESTION:   55  If 4"-GCB-2 fails, what is the rupture size?

Q-TYPE/TIMES ASKED:         DEP. INPUT PROB.                          13078
          BRANCHES:            4GCB2-Lg  4GCB2-NoL
                                1         2
   REALIZED SPLIT:          2.200E-02 9.780E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      2.200E-02
     DEPENDENCIES:    54

     REQ. BRANCHES:    1

       DESCRIPTION:   4GCB2-F

 CASE/BRANCH SPLIT:           2.200E-02 0.000E+00

 CASE NUMBER/SPLIT:    2      9.780E-01
       DESCRIPTION:          Otherwise

 CASE/BRANCH SPLIT:           0.000E+00 9.780E-01

******** QUESTION:   56  Does FE-DH2B fail?  (10" FE, 300 psi rating)

Q-TYPE/TIMES ASKED:         INDEP. CALC. PROB.                        13078
          BRANCHES:            DH2B-F    DH2B-NoF
                                1         2
   REALIZED SPLIT:          0.000E+00 1.000E+00

******** QUESTION:   57  If FE-DH2B fails, what is the  upture size?

Q-TYPE/TIMES ASKED:         DEP. INPUT PROB.                          13078
```

BRANCHES:		DH2B-Lg 1	DH2B-NoL 2
REALIZED SPLIT:		0.000E+00	1.000E+00

SUMMARY BY CASE

CASE NUMBER/SPLIT:	2	1.000E+00	
DESCRIPTION:		Otherwise	
CASE/BRANCH SPLIT:		0.000E+00	1.000E+00

LISTING 30

PSTEVNT Key Word File for DHR Letdown

The following is a listing of the PSTEVNT key word for the DHRL model. This file is used to control PSTEVNT execution during the rebinning process used to obtain aggregate system failure mode probabilities.

```
$-- Calculation Control Keywords (for logical constants) ---------------------
$
$   COLLAPS  XXXX              $ Reduce rebinned results with weighting
$                                factor
$
    REBIN                      $ Causes rebinning of accident progression
$                                bins
$
    RUN                        $ Causes PSTEVNT to procede with data
$                                calculations
$
    NOSORT                     $ Do not produce sort tables
$
$-- Calculation Control Keywords (for assigned values) ----------------------
$
$
$-- Input File Specification Keywords ---------------------------------------
$
    ASCTRIN                    $ ASCII output from EVNTRE
$
    BININ   pst_bin.dat        $ Filename for rebinning input
$
    EVNTBIN mc_psti.asc        $ Filename for EVNTRE output file
$
$   SORTIN  sortin             $ Filename for sort specification data
$
$-- Report Request Keywords -------------------------------------------------
$
    ASCSAV                     $ Rebinning result is ASCII
$
    RPTMLST                    $ Write EVNTRE master bin list to message file
$
    RPTRBIN                    $ Write rebinned bins to message file
$
$-- Output File Specification Keywords --------------------------------------
$
    BINOUT  rbin1.out          $ Rebinning result data
$
$   INPOUT  inpout             $ Annotated echo of input
$
    KEEPOUT keep.out           $ Master list of of unique kept bins
$
$   SBINOUT sbinout            $ Rebinning result data (for additional
$                                post-processing)
$
$   SORTOUT sortout            $ Result of requested sorts
$
```

```
$    TABOUT  tabout                $ Rebinning result descriptive table(s)
$
ENDKEY                             $ Indicates the end of keyword input.
```

LISTING 31

PSTEVNT Rebinning Data File for DHR Letdown

The following is a listing of the PSTEVNT rebinning data file for the DHRL model.

```
ISLOCA -- DHR Letdown
  1         FSize
  3   3    NoLeak     Small     Large
  1   1         1
                1
           NoLeak
  1   2         1
                2
            Small
  1   3         1
                3
            Large
```

LISTING 32

PSTEVNT Output Data File for DHR Letdown

The following is a listing of the PSTEVNT output data file for the DHRL model. This file contains the system failure mode probabilities for the DHRL model.

```
DHR LETDOWN BASE CASE
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

 FREQUENCY:
  BIN          TOTAL      ID                  FSize
  6.0644E-01  6.0644E-01 C                    Large
  2.5116E-01  8.5760E-01 B                    Small
  1.4240E-01  1.0000E+00 A                    NoLeak


 A TOTAL OF    3 OUT OF    3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
 FREQUENCY
```

LISTING 33

PSTEVNT Output Data File for DHR Letdown (Sensitivity)

The following is a listing of the PSTEVNT output data file for the DHRL model. This file contains the system failure mode probabilities for the DHRL model. These results differ from those in Listing 32 in that a log standard deviation of 0.10 was used for the piping pressure capacity (instead of 0.36).

```
SENSITIVITY WITH PIPE FAILURE LOG SIGMA = .10
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

 FREQUENCY:
  BIN          TOTAL       ID                 FSize
  5.0654E-01   5.0654E-01  C                  Large
  3.4806E-01   8.5460E-01  B                  Small
  1.4540E-01   1.0000E+00  A                  NoLeak


A TOTAL OF     3 OUT OF     3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
FREQUENCY
```

LISTING 34

PSTEVNT Output Data Files for DHR Letdown Cumulative Distributions

The following is a collection of PSTEVNT output data files for the DHRL model with a pipe failure log standard deviation of .36. The data provided here are used to construct the cumulative distributions shown if Figure 1. Each output summary represents a full event tree evaluation of the DHRL model at the indicated constant pressure (300 psi to 2200 psi in steps of 100 psi).

BINNED OUTPUT FOR DHR LETDOWN MODEL -- CONSTANT RCS PRESSURE, PIPE FAILURE LOG SIGMA = .36

RCS PRESSURE = 300. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9830E-01	9.9830E-01	A	NoLeak
1.7000E-03	1.0000E+00	C	Large

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 400. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.8118E-01	9.8118E-01	A	NoLeak
1.2300E-02	9.9348E-01	B	Small
6.5200E-03	1.0000E+00	C	Large

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 500. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
7.8556E-01	7.8556E-01	A	

			NoLeak
1.9310E-01	9.7866E-01	B	
			Small
2.1340E-02	1.0000E+00	C	
			Large

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 600. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
5.8026E-01	5.8026E-01	B	
			Small
3.7008E-01	9.5034E-01	A	
			NoLeak
4.9660E-02	1.0000E+00	C	
			Large

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 700. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
8.0860E-01	8.0860E-01	B	
			Small
1.0040E-01	9.0900E-01	C	
			Large
9.1000E-02	1.0000E+00	A	
			NoLeak

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 800. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
8.1044E-01	8.1044E-01	B	
			Small

1.7796E-01 9.8840E-01 C

Large

1.1600E-02 1.0000E+00 A

NoLeak

A TOTAL OF   3 OUT OF   3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 900. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
7.1838E-01	7.1838E-01	B	Small
2.8082E-01	9.9920E-01	C	Large
8.0000E-04	1.0000E+00	A	NoLeak

A TOTAL OF   3 OUT OF   3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1000. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
5.9724E-01	5.9724E-01	B	Small
4.0266E-01	9.9990E-01	C	Large
1.0000E-04	1.0000E+00	A	NoLeak

A TOTAL OF   3 OUT OF   3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1100. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
5.5126E-01	5.5126E-01	C	Large
4.4874E-01	1.0000E+00	B	

Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1200. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
7.0464E-01	7.0464E-01	C	Large
2.9536E-01	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1300. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
8.3626E-01	8.3626E-01	C	Large
1.6374E-01	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1400. PSI
AGGREGATED REBINNING RESULTS DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.1968E-01	9.1968E-01	C	Large
8.0320E-02	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1500. PSI

AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.6400E-01	9.6400E-01	C	Large
3.6000E-02	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1600. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.8252E-01	9.8252E-01	C	Large
1.7480E-02	1.0000F+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1700. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9120E-01	9.9120E-01	C	Large
8.8000E-03	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1800. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9354E-01	9.9354E-01	C	Large

6.4600E-03 1.0000E+00 B

Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1900. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9540E-01	9.9540E-01	C	Large
4.6000E-03	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 2000. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9748E-01	9.9748E-01	C	Large
2.5200E-03	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 2100. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9862E-01	9.9862E-01	C	Large
1.3800E-03	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 2200. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9934E-01	9.9934E-01	C	Large
6.6000E-04	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

LISTING 35

PSTEVNT Output Data Files for DHR Letdown Cumulative Distributions (Sensitivity)

The following is a collection of PSTEVNT output data files for the DHRL model with a pipe failure log standard deviation of 0.10. The data provided here are used to construct the cumulative distributions shown if Figure 2. Each output summary represents a full event tree evaluation of the DHRL model at the indicated constant pressure (300 psi to 2200 psi in steps of 100 psi).

```
SENSITIVITY RESULTS -- CONSTANT RCS PRESSURE, PIPE FAILURE LOG SIGMA = .1


RCS PRESSURE = 300. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown


 FREQUENCY:
  BIN         TOTAL       ID                FSize
  1.0000E+00  1.0000E+00 A
                                            NoLeak


A TOTAL OF    1 OUT OF    1 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
FREQUENCY


RCS PRESSURE = 400. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown


 FREQUENCY:
  BIN         TOTAL       ID                FSize
  9.8748E-01  9.8748E-01 A
                                            NoLeak
  1.2500E-02  9.9998E-01 B
                                            Small
  2.0000E-05  1.0000E+00 C
                                            Large


 A TOTAL OF    3 OUT OF    3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL
FREQUENCY


RCS PRESSURE = 500. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown


 FREQUENCY:
  BIN         TOTAL       ID                FSize
  8.0196E-01  8.0196E-01 A
                                            NoLeak
  1.9800E-01  9.9996E-01 B
                                            Small
```

4.0000E-05 1.0000E+00 C
Large

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 600. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
6.1286E-01	6.1286E-01	B	Small
3.8708E-01	9.9994E-01	A	NoLeak
6.0000E-05	1.0000E+00	C	Large

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 700. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
8.9800E-01	8.9800E-01	B	Small
1.0188E-01	9.9988E-01	A	NoLeak
1.2000E-04	1.0000E+00	C	Large

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 800. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.8490E-01	9.8490E-01	B	Small
1.4300E-02	9.9920E-01	A	NoLeak
8.0000E-04	1.0000E+00	C	

Large

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 900. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9398E-01	9.9398E-01	B	Small
5.2200E-03	9.9920E-01	C	Large
8.0000E-04	1.0000E+00	A	NoLeak

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1000. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.6894E-01	9.6894E-01	B	Small
3.0860E-02	9.9980E-01	C	Large
2.0000E-04	1.0000E+00	A	NoLeak

A TOTAL OF 3 OUT OF 3 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1100. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
8.5963E-01	8.5963E-01	B	Small
1.4036E-01	9.9999E-01	C	Large

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1200. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
6.1330E-01	6.1330E-01	B	Small
3.8670E-01	9.9999E-01	C	Large

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1300. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
6.9044E-01	6.9044E-01	C	Large
3.0956E-01	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1400. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
8.9134E-01	8.9134E-01	C	Large
1.0866E-01	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1500. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.7576E-01	9.7576E-01	C	Large
2.4240E-02	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1600. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9466E-01	9.9466E-01	C	Large
5.3400E-03	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1700. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9930E-01	9.9930E-01	C	Large
7.0000E-04	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1800. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9980E-01	9.9980E-01	C	Large
2.0000E-04	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 1900. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9990E-01	9.9990E-01	C	Large
1.0000E-04	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 2000. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
9.9990E-01	9.9990E-01	C	Large
1.0000E-04	1.0000E+00	B	Small

A TOTAL OF 2 OUT OF 2 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 2100. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
1.0000E+00	1.0000E+00	C	Large

A TOTAL OF 1 OUT OF 1 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

RCS PRESSURE = 2200. PSI
AGGREGATED REBINNING RESULTS FOR: DHR Letdown

FREQUENCY:

BIN	TOTAL	ID	FSize
1.0000E+00	1.0000E+00	C	
			Larg

A TOTAL OF 1 OUT OF 1 BINS WERE REQUIRED TO CAPTURE 100.00% OF THE TOTAL FREQUENCY

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11. ABSTRACT (200 words or less)

This document presents information essential to understanding the risk associated with inter-system loss-of-coolant accidents (ISLOCAs). The methodology developed and presented in this document provides a state-of-the-art method for identifying and evaluating plant-specific hardware designs, human performance issues, and accident consequence factors relevant to the prediction of the ISLOCA risk. This ISLOCA methodology was developed and then applied to a Babcock and Wilcox (B&W) nuclear power plant. The results from this application are described in detail. For this particular B&W reference plant, the assessment indicated that the probability of a severe ISLOCA is approximately 2.2E – 06/reactor-year.

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