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Assessment Of The Potential For ISLOCA At The Davis-Besse Nuclear Power Station

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ABSTRACT

Interfacing System Loss-Of-Coolant Accidents (ISLOCAs) have been identified as important contributors to risk for some nuclear power plants. This document presents a methodology for identifying and evaluating plant specific hardware designs, human performance issues, and accident consequence factors relevant to the estimation of ISLOCA risk. Also presented is a description of the application of this methodology at the Davis-Besse Nuclear Power Station.

SUMMARY

Interfacing Systems Loss-Of-Coolant Accidents (ISLOCAs) have been identified in some Probabilistic Risk Assessments (PRAs) as major contributors to risk at Nuclear Power Plants (NPPs). They have the potential to result in core melt and containment bypass, which may lead to the early release of large quantities of fission products. Recent events at several operating reactors have been identified as ISLOCA precursors. These events have raised concerns over the frequency of occurrence, potential initiators, and means of identifying and mitigating this potential accident. In response to these concerns, a June 7, 1989 memorandum titled "Request for Office of Nuclear Regulatory Research (RES) Support for Resolution of the ISLOCA Issue", was transmitted from Dr. Thomas E. Murley to Dr. Eric S. Beckjord. The ISLOCA Research Program described in this report was initiated in response the this memorandum.

The objective of the ISLOCA Research Program is to provide the NRC with qualitative and quantitative information on the hardware, human factors, and accident consequence issues that dominate nuclear power plant risks for Interfacing System Loss Of Coolant Accidents (ISLOCA). To meet this objective, a methodology has been developed to estimate the core damage frequency and risk associated with an ISLOCA and this methodology is being applied for individual NPPs. The application will examine as many as six nuclear power plants. This report describes the ISLOCA methodology and documents the results from its application at the first of the plants, the Davis-Besse Nuclear Power Station.

An eight step methodology was developed to perform qualitative and quantitative evaluations for an ISLOCA. The steps and their relationship are show in Figure S1. Application of this methodology to Davis-Besse was performed by a team of PRA and human factors specialists. The important results that are specific to Davis-Besse are:

1. Human errors which could occur during startup and shutdown of the plant were found to be significant contributors to ISLOCA core melt frequency and risk. Human errors that strongly influenced the ISLOCA initiating events were latent human errors (errors whose consequences lie dormant for a long time) in conjunction with human errors of commission which occurred during execution of the normal procedural tasks.

2. The ISLOCA scenarios that were influenced primarily by hardware failures were relatively small contributors to core melt frequency and to the risk associated with an ISLOCA.

3. Isolation of the break would be an important mitigating action during an ISLOCA because makeup capability for the BWST is insufficient to maintain an adequate reactor coolant inventory for breaks outside the containment that are larger than two inches in diameter. Although the hardware failure analysis indicates that hardware would be available to isolate these ISLOCA breaks, adequate procedures or training are not available to ensure that this hardware is used.

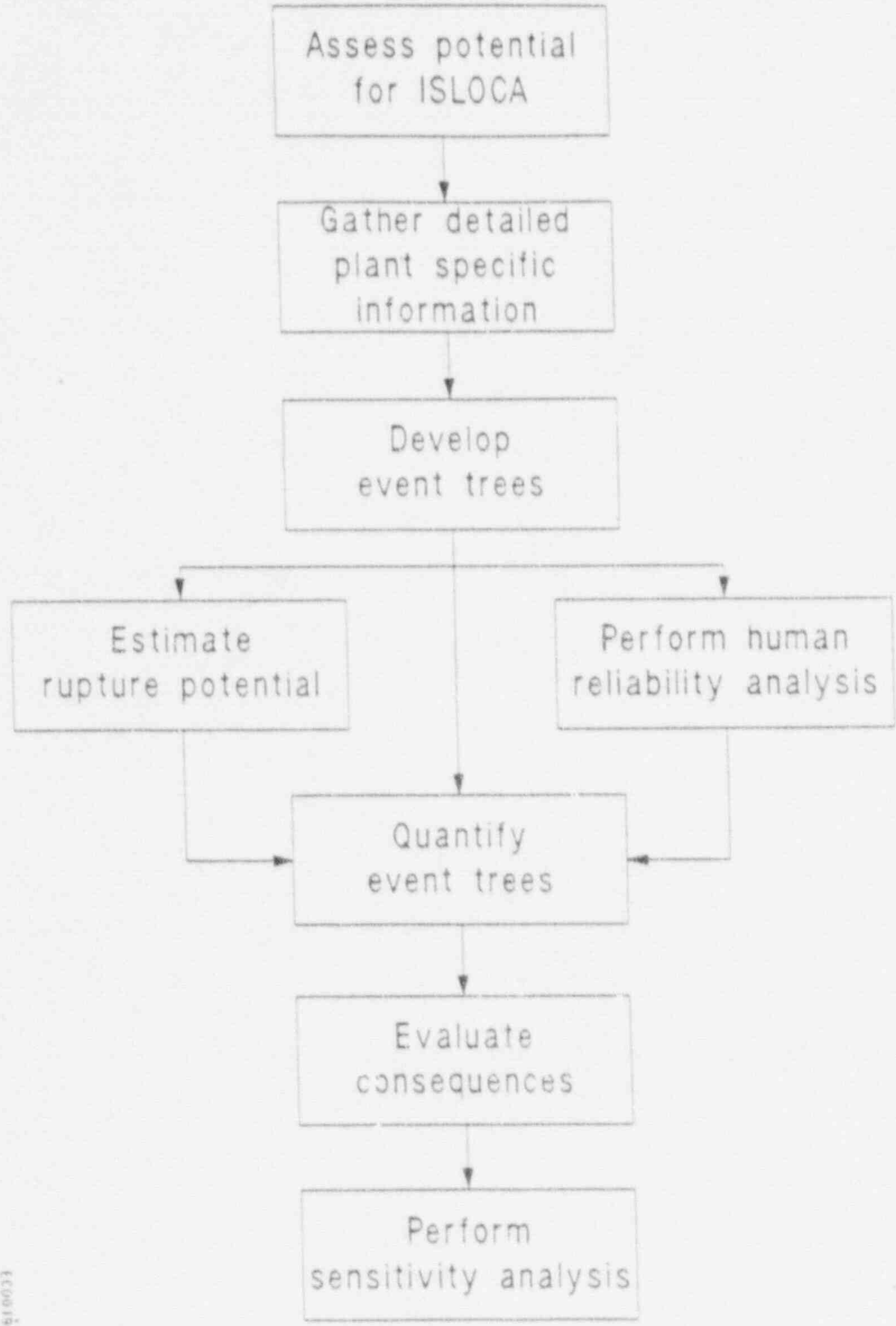

Figure S1 Approach for Plant Specific Evaluation of ISLOCA

4. A significant reduction in risk could be achieved through relatively simple changes to procedures, training, and instrumentation. It appears that improvements in both safety culture and situational awareness would be important in reducing the potential for an ISLOCA.

5. There is adequate equipment separation and redundancy so that damage by flooding or by spraying adjacent equipment are not risk significant.

6. The ISLOCA methodology has been successful in providing important insights on the relative contribution of both hardware faults and human actions to core melt frequency and risk.

Caution must be exercised when considering the extrapolation of the Davis-Besse results to draw general conclusions that would apply to other plants. The strong influence of human errors on ISLOCA risk during both startup and shutdown do indicate that ISLOCA evaluations for other plants should include a comprehensive assessment of the role of the plant personnel. This assessment should consider the potential for errors of commission and the effect of possible latent errors during the normal execution of procedures.

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CONTENTS

Assessment Of The Potential For ISLOCA At The Davis-Besse Nuclear Power Station

1. INTRODUCTION

The Reactor Safety Study, WASH-1400 [1], identified a class of accidents that can result in overpressurization and rupture of systems that interface with the reactor coolant system. These events were postulated to be caused by the failure of the check valves and motor operated valves normally used for system isolation. For a subset of these interfacing system loss-of-coolant accidents (ISLOCAs), called V-sequences or event V, the system rupture occurred outside the containment. If the rupture caused core damage, some ISLOCAs were shown to be significant contributors to risk since the fission products bypassed the containment and were discharged directly to the environment. Subsequent probabilistic risk assessments (PRAs), including the NUREG-1150 results for Surry [2] and Sequoyah [3], have identified ISLOCAs as important contributors to public health risk. Researchers at Brookhaven National Laboratory have evaluated the vulnerability of reactor designs to an ISLOCA and identified improvements that would reduce ISLOCA frequency [4,5]. Leak Testing of PIVs if they were never +

Recent events at several operating reactors have been identified as precursors to an ISLOCA. These events have raised concerns over the frequency of occurrence, potential initiators, and means of identifying and mitigating this potential accident. In response to these concerns, a June 7, 1989 memorandum titled "Request for Office of Nuclear Regulatory Research (RES) Support for Resolution of the ISLOCA Issue", was transmitted from Dr. Thomas E. Murley to Dr. Eric S. Beckjord. The ISLOCA Research Program described in this report was initiated in response to this memorandum.

The objective of the ISLOCA Research Program is to provide the NRC with qualitative and quantitative information on the hardware, human factors, and accident consequence issues that dominate nuclear power plant risks for Interfacing System Loss Of Coolant Accidents (ISLOCA). This information is to be used in:

- Developing a PRA framework for evaluating the ISLOCA and identifying insights with respect to the risk contribution from both hardware and human error issues along with recommendations for risk reduction.

- Highlighting the effects of specific types of human errors and their root causes, on ISLOCA risk along with recommendations for risk reduction.

- Evaluating the fragility of low pressure systems when exposed to high pressure, high temperature reactor coolant system. This evaluation will include identification of likely failure locations and their probabilities of failure.

- Identifying and describing potential ISLOCA sequences with respect to sequence timing, possible accident management strategies and effects of ISLOCAs on other equipment and systems.

- Estimating the consequences associated with postulated ISLOCA events, including estimates of source terms and offsite consequences. Again, important issues will be identified and recommendations will be made on possible consequence reduction actions.

Real and potential ISLOCA problems considered in this program are limited to those that could result in core damage and could bypass the containment.

To meet the program objectives, a methodology has been developed to estimate the core damage frequency and risk associated with an ISLOCA and this methodology is being applied for as many as six nuclear power plants. This report describes the ISLOCA methodology and documents the results from its application at the first of the plants, the Davis-Besse Nuclear Power Station. These results tend to emphasize the effect of hardware failures and human actions on the ISLOCA core damage frequencies. The risk values are considered to be most useful in comparing results from the sensitivity studies. The identification of the uncertainties in this estimate is provided.

Section 2 of this report describes the methodology developed to evaluate the effects of an ISLOCA, the approach taken for its application to a specific plant, and a description of the Davis-Besse systems that were identified as interfacing systems. Section 3 contains a description of the Davis-Besse interfacing systems and the possible ISLOCA sequences. Section 4 describes the plant specific results from the assessment of ISLOCA at Davis-Besse and Section 5 contains the conclusions and recommendations based on this assessment. Appendices are used to document the details of many of the evaluations.

2. APPROACH

The general approach that is being used to evaluate ISLOCA risk and plant vulnerabilities is to perform a detailed analysis for a diverse sample of six plants and, to the extent possible, extrapolate and generalize these results for additional plants. To accomplish the detailed plant analysis, a methodology was developed that was designed to meet the program objectives discussed in the previous section. The steps in this individual plant methodology are illustrated in Figure 1. Subsections 2.1 through 2.8 briefly discuss each of the steps.

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Prior to initiation of individual plant evaluations, a review of historical plant operating information was performed to provide insights on potential ISLOCA issues. The major emphasis of this evaluation was an identification and evaluation of Licensee Event Reports (LERs) that (a) involved valve failures resulting from either hardware or human causes or (b) indicated an ISLOCA had occurred. The results from this search provided information on the causes and frequencies of valve failures and provided important insights on the systems involved and the potential causes of ISLOCAs that have occurred. This information was used during the plant visits to aid in identifying systems to be reviewed, during the development of the events in the event trees, and for quantification of the failure rates of some interfacing system valves. A brief summary of the results of this evaluation is documented in Appendix A.

2.1 Assess Potential For ISLOCA

The initial step in the individual plant evaluation approach is to make a preliminary assessment of the potential for an ISLOCA. Plant specific information on the potential systems that could be involved in an ISLOCA are obtained during a short data gathering visit to the plant. Detailed information is obtained on the hardware and operations of a wide range of low and high pressure interfacing systems. Examples of information collected include: plant procedures, P&IDs, isometric

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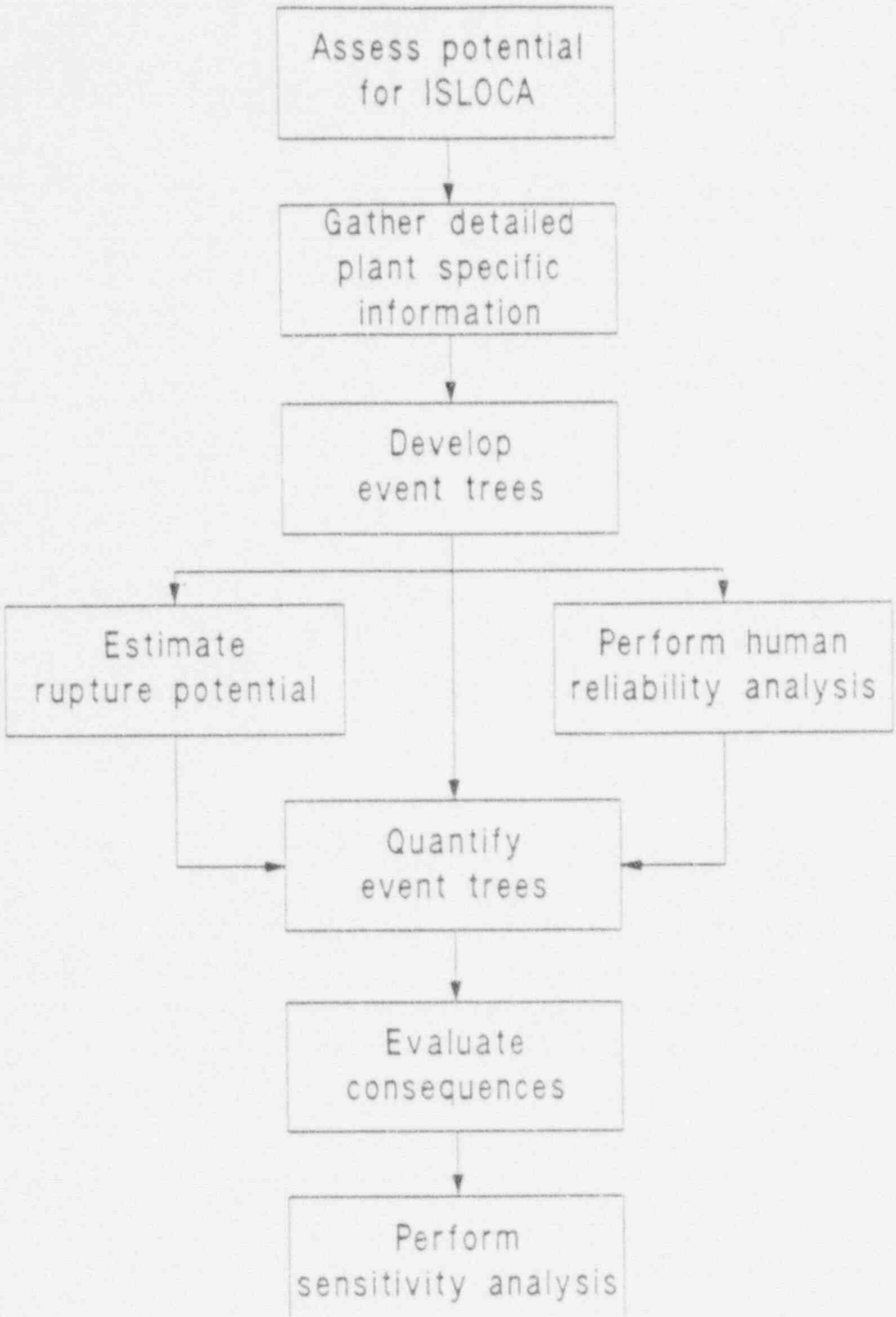


Figure 1 Approach for Plant Specific Evaluation of ISLOCA

drawings, training manuals, etc. This information is then reviewed by a team of PRA and human factors specialists to become familiar with the systems and operations that have the potential to initiate, prevent, or mitigate an ISLOCA. All systems that interface with the RCS are identified. A determination is then made of the maximum interfacing system break size that would not be expected to result in core damage. The interfacing systems are screened to identify those that had pipe sizes larger than this maximum and that could cause the containment to be bypassed. The systems that meet the screening criteria are analyzed further to identify potential ISLOCA initiators and sequences. The identified sequences are developed in sufficient detail to guide a team of PRA and human factors specialists in obtaining detailed information during an extended plant visit.

2.2. Gather Detailed Plant Specific Information

An extended visit to the plant is necessary to gather the information needed to complete the review, development, and assessment of the candidate ISLOCA sequences. Members of the team that developed the candidate sequences obtain the needed information by interviewing operations personnel and walking down the systems of interest. The types of information that are obtained during this visit include:

a. Detailed information on the hardware that would be involved in an ISLOCA. For example data on: control valves, relief valves, piping, flanges, pumps, heat exchangers, etc.

b. Detailed information on the procedures and guidelines followed by plant personnel during startup, normal power operation, and shutdown of the plant.

c. Detailed information on the factors that could influence the performance of the plant personnel as it relates to initiation, detection, prevention, or mitigation of an ISLOCA.

2.3 Develop Event Trees

After the specific plant information is collected, the final list of interfaces and sequences is generated and the detailed analysis begins. This analysis is performed through a joint effort of the PRA and human factors specialists. The sequences are modeled using component level event trees that combine the hardware faults and the human errors that compose each sequence. Generally the event trees comprise three phases:

1. The initiating events, which are those combinations of failures, both hardware and human related, that result in a breach of the pressure isolation boundary and allow high pressure RCS water to enter the lower pressure interfacing system

2. The rupture events which identify the probability of a rupture in the interfacing system, its size, and its location

3. The post-rupture events that estimate the performance of the control room operators in recovery from or mitigating the consequences of an ISLOCA.

2.4 Estimate Rupture Potential

During an ISLOCA it is important to assess the performance of those components that are designed for low pressure conditions when they are exposed to high pressures associated with and ISLOCA. The basic approach for performing this assessment is:

a. An event tree model is built that asks questions about the failure mode of each of the important low pressure components. This model is structured and input to the EVNTRE computer code which was developed for the assessment of complex event trees by the NUREG-1150 program.

b. The failure probability of each piece of equipment in the low pressure rated system is described as a lognormal distribution with a specified median failure pressure and standard deviation.

c. Thermal-hydraulic simulations of the systems are performed to estimate the pressure distribution in the system based on the expected initiating event, initial primary system conditions, and on the expected performance of relief valves designed to protect the systems.

d. Each question in the event tree is answered by (1) randomly selecting a failure pressure from the failure pressure distribution of the appropriate component and (2) comparing the selected component failure pressure with a selected system pressure. The system pressure was randomly selected based on the expected operating conditions and assuming a normal distribution with an estimated mean and standard deviation. If the sampled component failure pressure is below the sampled system pressure, the component has failed. Otherwise no failure is assumed. Each component in the low pressure rated system is evaluated in this manner until all questions in the event tree have been examined. This process is repeated approximately 10,000 times in a true Monte Carlo simulation, which is feasible because of the relatively small size of the EVNTRE model.

e. Once the simulation is completed, the output is binned and estimates can be made about the relative frequency of equipment failures given system overpressurization.

The component and piping failure pressures used for the rupture calculations were developed in an independent structural analysis performed by Impell Corporation. Not only were failure pressures calculated but likely leak rates and leak areas as well. In this respect, flanges exhibit somewhat unique behavior in that there are actually two failure pressures of interest. First, is the estimated Gross

Leak Pressure (GLP) at which pressure a measurable leak area appears. At lower pressures, leakage is possible but at very small rates (measured in mg/sec) from seepage around the gasket. 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_0, 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_0, and greater than P_0) are associated with three sizes of leaks, namely spray leaks, small leaks and large leaks, respectively.

2.5 Perform Human Reliability Analysis

The methodology for human reliability analysis (HRA) was developed using guidelines from the NRC sponsored TALENT Program, the Systematic Human Action Reliability Procedure (SHARP) [6], the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (THERP) [7], and the draft IEEE standard P1082/D7 [8].

The HRA methodology uses the seven steps of the SHARP method as a general framework. These seven steps are as follows:

1. Ensuring that all of the many types of human actions and interactions are considered in this analysis.

2. Identifying and screening the specific human interactions which are significant contributors to the safety and operation of the plant.

3. Developing a detailed description of important human interactions through the definition of key factors needed to complete the model, e.g., representation, impact assessment, and quantification.

4. Selecting and applying appropriate techniques for modeling the important human actions in logic structures.

5. Evaluating the impact of significant human actions which were identified in step #4.

6. Quantifying the probabilities for the various human actions and interactions, determining sensitivities, and establishing uncertainty ranges.

7. Documenting all of the necessary information for the analysis to be understandable, traceable, and reproducible.

Based upon preliminary operational information from the plant, Steps 1 and 2 will identify human error actions which were involved in potential ISLOCA accidents. Initial screening human error probabilities (HEP's) can then be assigned using the fine screening techniques from SHARP. As detailed plant information becomes available, a second set of screening HEP's can then be generated. These screening HEP's are then used in the ISLOCA event trees that were developed through a joint effort between the PRA and human factors personnel. The screening values allow the PRA analysts to determine where detailed HRA information should be developed using steps 3 through 6.

THERP type HRA event trees were chosen for modeling most of the human actions for the detailed analysis (Step 4). However, several ISLOCA scenarios may not lend themselves to THERP event trees, since there may be cases which involve errors of commission as well as omission. In these cases, HRA fault trees and commission event trees (COMETs) can be used alone, or in conjunction with the THERP event trees. Detailed analyses are conducted using the fault trees and/or THERP event trees to estimate the probabilities of the dominant human actions.

2.6 Quantify Event Trees

The events listed on the ISLOCA event tree are supported by separate calculations that generate the probabilities. The means of obtaining the rupture event probabilities and the probabilities relating to failure of plant personnel were discussed previously. Hardware failure probabilities were generally developed using fault trees and the hardware data base documented in Appendix B. The ISLOCA event trees were quantified using the ETA-II pc-computer code.

2.7 Consequences

Once the ISLOCA sequence event trees were quantified and the sequence frequencies generated, they were combined with the corresponding consequences calculated using the MACCS code to produce the overall ISLOCA risk estimates. These MACCS consequences were generated using a hybrid input deck. The source term used was taken from the Oconee PRA and scaled for the Davis-Besse power level. Like Davis-Besse, Oconee is a B&W supplied NSSS and the source term used is the one identified with the containment bypass V-sequence. The site information was taken from the Surry deck used in the NUREG-1150 program. The Surry site was chosen by reviewing the Sandia Siting Study and calculating an average site based on weather weighted population density. This average population density was then compared to the five NUREG-1150 sites and Surry was chosen because it most closely matched the average population density.

2.8 Sensitivity Studies

A number of issues can be examined through sensitivity studies to assess their relative influence on core melt frequency and risk. These issues are related to the methods used to perform the plant evaluations as well as uncertainties that may be specific to each plant. For the initial plant evaluations, issues were chosen for examination through sensitivity studies because: (1) there was a relatively large uncertainty in the values used for a particular parameter, (2) a potential fix was postulated that was expected to result in a significant reduction in core damage frequency and risk, or (3) a different means of establishing probabilities was being considered which could be used for evaluation of future plants.

3. DESCRIPTION OF THE DAVIS-BESSE INTERFACING SYSTEMS

The Davis-Besse Nuclear Power Station is sited on Lake Erie in Ottawa County, Ohio, approximately six miles northeast of Oak Harbor. The Plant is owned and operated by Centerior Energy Corporation, which was formed by the union of Toledo Edison Company and Cleveland Electric Illuminating. Commercial operations began in September 1976. Davis-Besse reactor is designed for a core power level of 2,772 MWt and a net electrical output of 906 MWe. The NSSS is supplied by Babcock and Wilcox (B&W) with Bechtel providing the Architect Engineering services and Chicago Bridge & Iron Company being responsible for the detail-design and construction of the containment vessel.

3.1. Interfacing Systems

A screening of all interfacing systems was made to identify those systems that needed further evaluation. The criteria used in this screening was that any system with an interfacing pipe size larger than one inch should be evaluated. The one inch pipe size was selected based on an estimation of the discharge from a high pressure one inch pipe break, which was about 200 gpm. A 200 gpm leak rate outside of the containment is considered to be critical based on: the capacity of the BWST (approximately 480,000 gal), the capacity of a single RCS makeup pump (150 gpm), and the normal makeup rate to the BWST (150 gpm). Based on these considerations and the number of hours it would take for the plant to achieve cold shutdown (conservatively assumed to be about 10 hours), leak rates of 200 gpm or less were judged not to be risk significant.

The screening resulted in the selection of the High Pressure Injection (HPI) discharge lines, the Low Pressure Injection (LPI) discharge lines and the Decay Heat Removal (DHR) letdown lines. Figure 2 is a schematic diagram showing the hardware configuration of the HPI system and Figure 3 provides similar information for the RHR/LPI system. Additional details on these systems are provided in Appendix C. The HPI interface comprises four separate reactor pressure vessel (RPV) injection lines. Starting

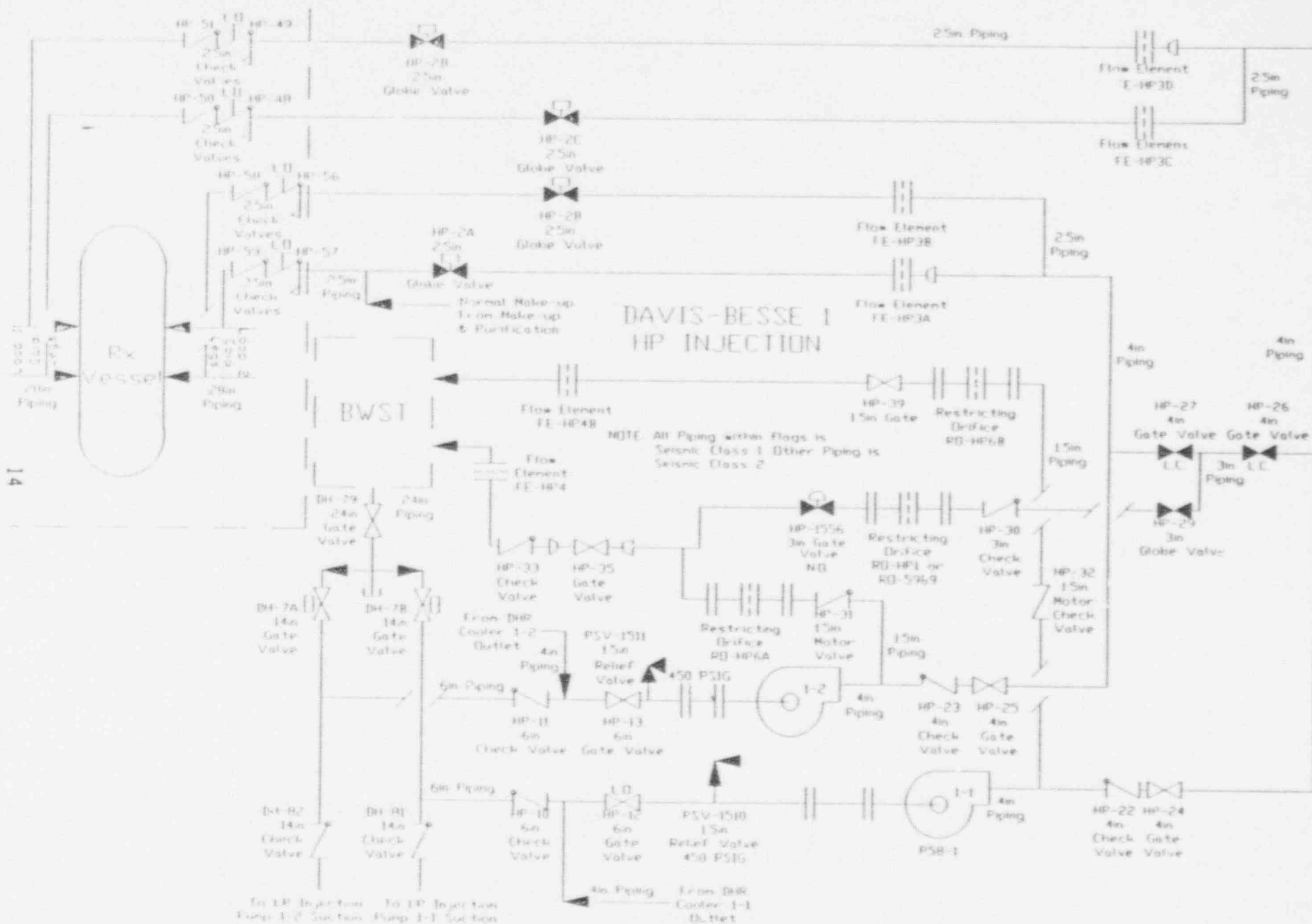


Figure 2 Schematic Diagram of the Davis-Besse HPI System

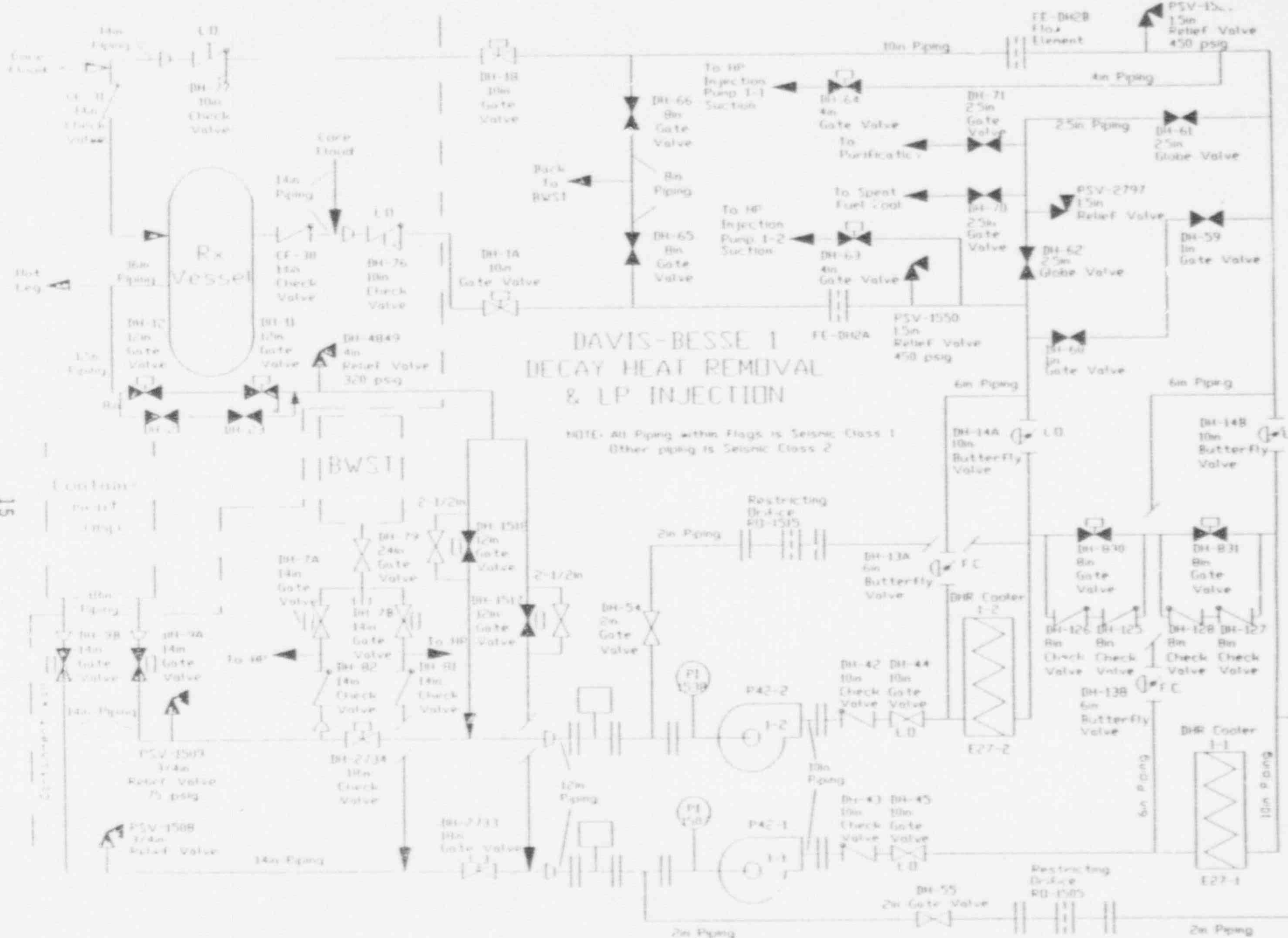


Figure 3 Schematic Diagram of the Davis-Besse DHR/LPI System

from the RPV, each injection line contains two check valves that are welded together (hence they cannot be individually leak tested), a normally closed motor operated valve and the HPI pump discharge check valve. The four lines are identified by the associated MOV, namely HP-2A (B, C, D). The HPI A-line, is also used for normal RCS makeup by the Make-Up and Purification system (MU&P). The MU&P connects to the HPI A-line between the two check valves and the normally closed MOV (HP-2A).

3.2. Possible ISLOCA Sequences

Sequences were developed through examination of the system interfaces and plant operational information by a team of PRA and human factory specialists. In some cases, for example the LPI injection lines, the sequences are strictly hardware driven, that is the ISLOCA potential is a function of the hardware failure rates of the ressure isolation boundary (PIB) valves. In other cases, for example the DHR letdown lines, the possible ISLOCA sequences are initiated by human errors. The table below summarizes the ISLOCA sequences identified for the Davis-Besse analysis.

Table 3.2-1. List of ISLOCA Interface Sequences.

Interface System	Sequence Description	Sequence ID
LPI (two lines)	Hardware failure of two check valves	LPI
DHR-letdown (shutdown)	Premature opening of letdown MOVs during shutdown	DHR-SD
DHR-letdown (startup)	Startup with letdown MOVs left open	DHR-SU
HPI (B, C, and D legs)	Hardware failure of two check valves and stroke test of MOV	HPI
HPI (A-leg)	stroke test of HP-2A and failure of two check valves	MU&P

3.2.1. LPI Sequences.

Only a single ISLOCA sequence was identified for the LPI interface. Because of the absence of routine operations associated with the LPI system, this sequence comprises a series of hardware related check valve failures that characterizes the classical V-sequence.

3.2.2. DHR Sequences.

When Davis-Besse operates in one of its shutdown modes (i.e. modes 4 or 5), the DHR system is used for removing core decay heat. It operates via a 12-inch pipe connected to one of the RCS hot legs and is isolated by two 12-inch motor operated gate valves in series (DH-12 and DH-11). There is also an 8-inch line that bypasses DH-11 and DH-12 that has two locally-manually operated gate valves in series.

There are two sequences that relate to possible ISLOCAs. First, is 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). The second scenario involves leaving the DHR letdown line open after the DHR pressure limit has been exceeded during plant startup. 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.

3.2.3. HPI Sequences.

During most operating modes, the MU&P system operates to provide cleanup of the RCS water and to provide seal injection to the reactor coolant pumps. The normal makeup flows from the MU&P system through the HPI A-header via check valves HP-57 and HP-59.

Several unique features, specific to the Davis-Besse HPI interface, create the potential for an ISLOCA related scenario, these are:

1. The HPI pressure isolation check valves (HP-57/59, HP-56/58, HP-48/50, and HP-49/51) are welded together. This arrangement prevents leak testing each valve individually. Therefore, a successful leak test does not necessarily confirm that both of the check valves are properly seated. Because of this untestable design, it is possible that the one of the valves might have been installed incorrectly or might not have received proper maintenance.

2. The normally closed HPI-MOVs (HP-2A,B,C, and D) are stroke tested quarterly. When the HPI A-header valve (HP-2A) is stroke tested, the MU&P system continues to provide RCS makeup through that line. When HP-2A is opened, high pressure makeup water back-flows all the way to the HP-pump discharge check valve (HP-23). Once the test is completed, the MOV is closed, and the HPI line is vented by opening a recirculation line to the BWST. This process presents an opportunity to allow RCS water to flow into the BWST or to allow RCS water to backflow through the HP pump, either of which could result in an ISLOCA.

4. DAVIS-BESSE RESULTS

Because of the unique nature of the ISLOCA sequence, a detailed understanding of the capabilities of the plant hardware and the personnel are needed to accurately analyze the ISLOCA challenge. For this report, an ISLOCA is considered to involve a loss of reactor coolant outside containment. Since the supply of water available for cooling the core is limited, a high priority item for the control room operators should be isolating the rupture and terminating the leak. Although the BWST inventory is maintained at about 480,000 gallons, even a small ISLOCA (equivalent to a 2 inch line) will result in a leak rate of about 1,000 gpm, which would deplete the BWST in about 8 hours. (The BWST makeup system, which is capable of 150 gpm, would not significantly affect this scenario.) Most postulated ruptures, particularly those associated with the DHR system, would result in much larger leakage rates. However, if the rupture were isolated in a timely manner and the leak terminated, the plant could, in all likelihood, be safely cooled down using the auxiliary feedwater system (AFW) and steam generator (SG) cooling. This is particularly significant in most sequences, since the likely rupture location would disable one or both trains of the DHR system, preventing direct cooldown of the primary system.

4.1. Davis-Besse Event Trees

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, as reflected in the frequency of the initiating event. 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 initiating event is only slightly conservative, since the event trees are based on the plant operating all four quarters per year but include one outage (during which manual values DH-21 and DH-23 are opened to allow MCVATS 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. 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. Finally, each event tree end-state was assigned to one of the consequence bins listed below.

OK - No overpressurization of the low pressure system occurred.

OK-op - Scenario results in overpressurization of the interfacing system but the system does not rupture or leak.

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

REL-mit - An ISLOCA with core damage occurs but the radioactive release is mitigated through some accident management strategy.

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

The REL-mit and REL-lg bins are sometimes subdivided according to failure location, with the new bins identified as RL1, RL2, etc. These bins are described further in the appropriate sequence description.

4.1.1. 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 4. The base case ISLOCA event tree for this system is shown in Figure 5. During most operating modes, 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: 1) 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. Furthermore, it is possible that the redundant valve could have been installed incorrectly at the time the plant was built, with the fault having since gone undetected. 2) 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 back-flows 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 a recirculation line to the BWST. This process presents an opportunity for RCS water to flow into the BWST and for RCS water to backflow through the HP pump. shows the doubl[e] check valves

The MU&P event tree nodes are defined as follows. Also listed are the base case branch probabilities.

M1 - 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 initiating event is quantified based on four quarters per year to obtain a

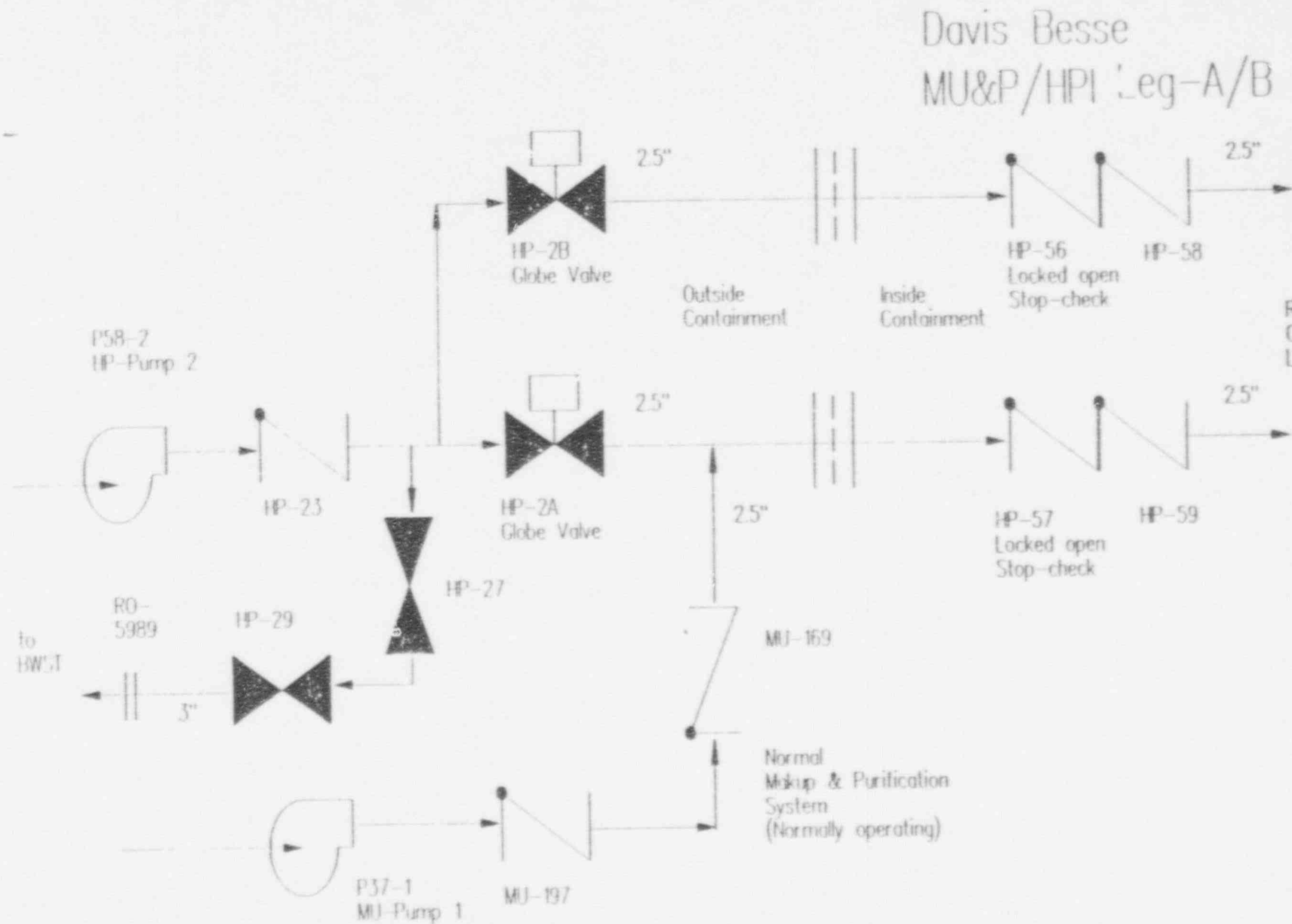


Figure 4. Schematic Diagram of the Makeup and Purification System Interface.

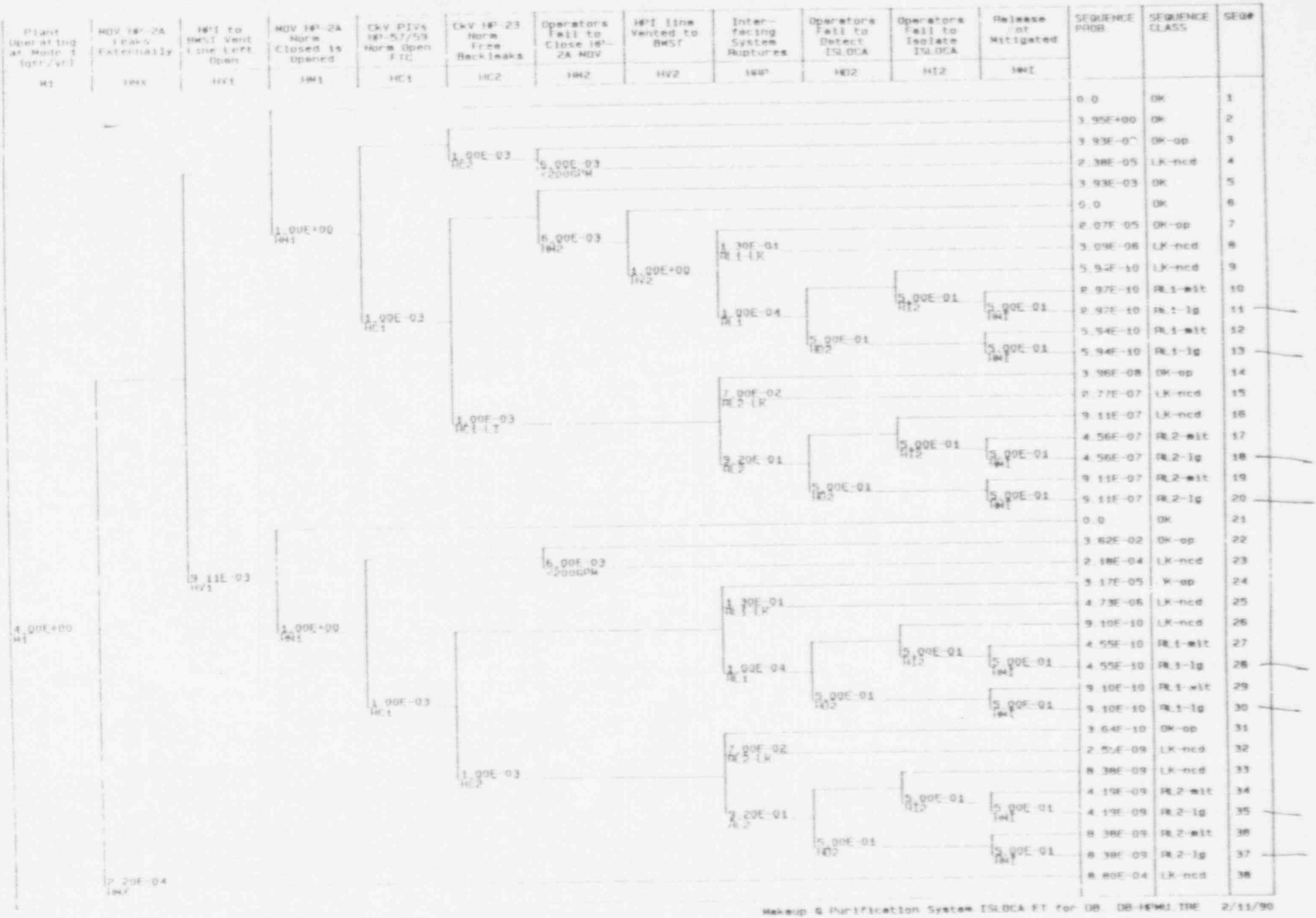


Figure 5. Makeup and Purification System ISLOCA Sequence Event Tree.

yearly estimate of the accident frequency. The subsequent events are quantified under the assumption that the MU&P system is operating through HPI leg A.

HMX - MOV HP-2A Leaks Externally. 2.2E-4

The nodal probability value for this event is calculated by taking the product of the hourly failure rate of 1.0E-7 (see Appendix B), calculated from the LER aggregations, and the number of hours per quarter (2,190). This event results in an RCS leak outside containment but the expected leak rate is small enough that core cooling is not threatened.

HV1 - HPI to BWST Vent Line Left Open. 0.009

The normal procedure for executing the stroke test of HP-2A includes venting the HPI line to the BWST after the test is complete. This is done in order to relieve the pressure in the HPI line between the HPI pump discharge check valve (HP-23) and HP-2A. This event accounts for the possibility that the operators could inadvertently leave the vent line open during the previous stroke test of MOV HP-2A. The value used is based on an HRA task analysis of the stroke test procedure (see Appendix E).

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

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

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

This is a demar[illegible]ilure rate, for one valve, which is based on data in the NUCLARR d[illegible]se (see Appendix B). Since these valves are welded together and [illegible]ly be leak tested as a pair, the probability is very high that ove[illegible]he years one valve will have entered a failed state. This failure will not be detected during leak testing, since the test only verifies that one of the two valves is positively seated. 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). However, the MU&P flow will be diverted from the RCS and the MU&P automatic control system will increase the makeup flow rate in response to the resulting decrease in pressurizer level. With the valves closed, the net leakage rate out of any resulting rupture would likely be limited to the capacity of single MU&P pump (about 150 gpm).

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

This is the probability of the valve failing to close after a quarterly test of the HP-pump. The value of 1.0E-3/demand is from the NUCLARR database. For more details, see Appendix B.

HM2 - Operators Fail to Close HP-2A MOV. 6.0E-3

During the quarterly stroke test of HP-2A, the valve is opened and the time required for the valve to transit from fully closed to fully opened is measured. The valve is then returned to its normal closed state. This event models the possibility that the operators fail to reclose the valve. The probability used is based on the combination of both hardware failure (from Appendix B) and human error (from Appendix E) probabilities (3.0E-3 plus 3.0E-3, respectively).

HV2 - Operators Vent High Pressure Injection line to Borated Water Storage Tank. 1.0

Opening HP-2A while the MU&P system is providing normal makeup to the reactor coolant system pressurizes the HPI line to the discharge pressure of the MU&P pump (about 2200 psi). After HP-2A is reclosed, the HPI line remains pressurized. This pressure is vented during the routine performance of the HP-2A stroke test by opening the HPI pump recirculation line back to the BWST (i.e., by opening HP-27 and HP-29).

HRP - Interfacing System Ruptures. 1.0/1E-4

This event is evaluated in a separate analysis that utilizes 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 obtained from a structural analysis performed by IMPELL Corporation (Appendix G). The rupture probability for various components is 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 then just the fraction of the Monte Carlo sample observations in which system pressure exceeded 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 rupture locations. The first is in the recirculation line to the BWST, downstream from manual valve HP-35, at which point the pipe schedule changes from 1500 psi rated to 150 psi rated. Since the BWST contains both an overflow line and a vent line, overpressurization of the BWST is not a credible scenario. The second likely rupture 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 on demand (see event HC2, above). The BWST recirculation line and the HPI pump suction line are identified as rupture locations RL1 and RL2, respectively. A rupture in either location would likely disable one train of each ECC system, including HPI, LPI, and CSS, but excluding the MU&P system.

HD2 - Operators Fail to Detect ISLOCA. 0.5) too high

A number of indicators of an interfacing system rupture are available to the control room operators. These indicators are primarily pressure, temperature, and computer alarms. The probability that the operators will detect an overpressure/ISLOCA situation is estimated through the use of screening values for knowledge-based behavior (see Appendix E). Note that this event does not include the process by which operators diagnose the situation. All that is included here is the detection of an overpressure rupture in an interfacing system, not the identification of the cause or the corrective actions.

HI2 - Operators Fail to Isolate ISLOCA. 0.5

After the operators have become aware of an abnormal situation, they must diagnose the cause and initiate some corrective actions. This event models the probability that they fail to do so successfully. The probabilities used for this event were derived from screening values developed for knowledge-based actions (Appendix E).

HMI - Release Not Mitigated. 0.5

Once an accident sequence progresses to core damage and a radioactive release is imminent, there are steps the operators could take to reduce the severity of the release. Specifically, actuation of the fire protection sprinkler system would provide some scrubbing of fission products, which would mitigate the offsite consequences of the release. Because there are no procedures or training for this action, a knowledge-based screening value of 0.5 (see Appendix E) is used for the probability that the operators will fail to initiate mitigative actions, given that core damage has occurred and a radioactive release is about to occur.

4.1.2. High Pressure Injection System Interface Event Tree - HPI.

Figure 6 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 7. Each of the two HPI pump trains branch into two injection legs, with each injection leg discharging into one of the RCS cold legs. As mentioned in the description of the MU&P event tree, the pressure isolation boundary is maintained by two check valves that are welded together, a normally closed MOV that is stroke tested quarterly, and the HPI pump discharge check valve. Because the MU&P system provides normal

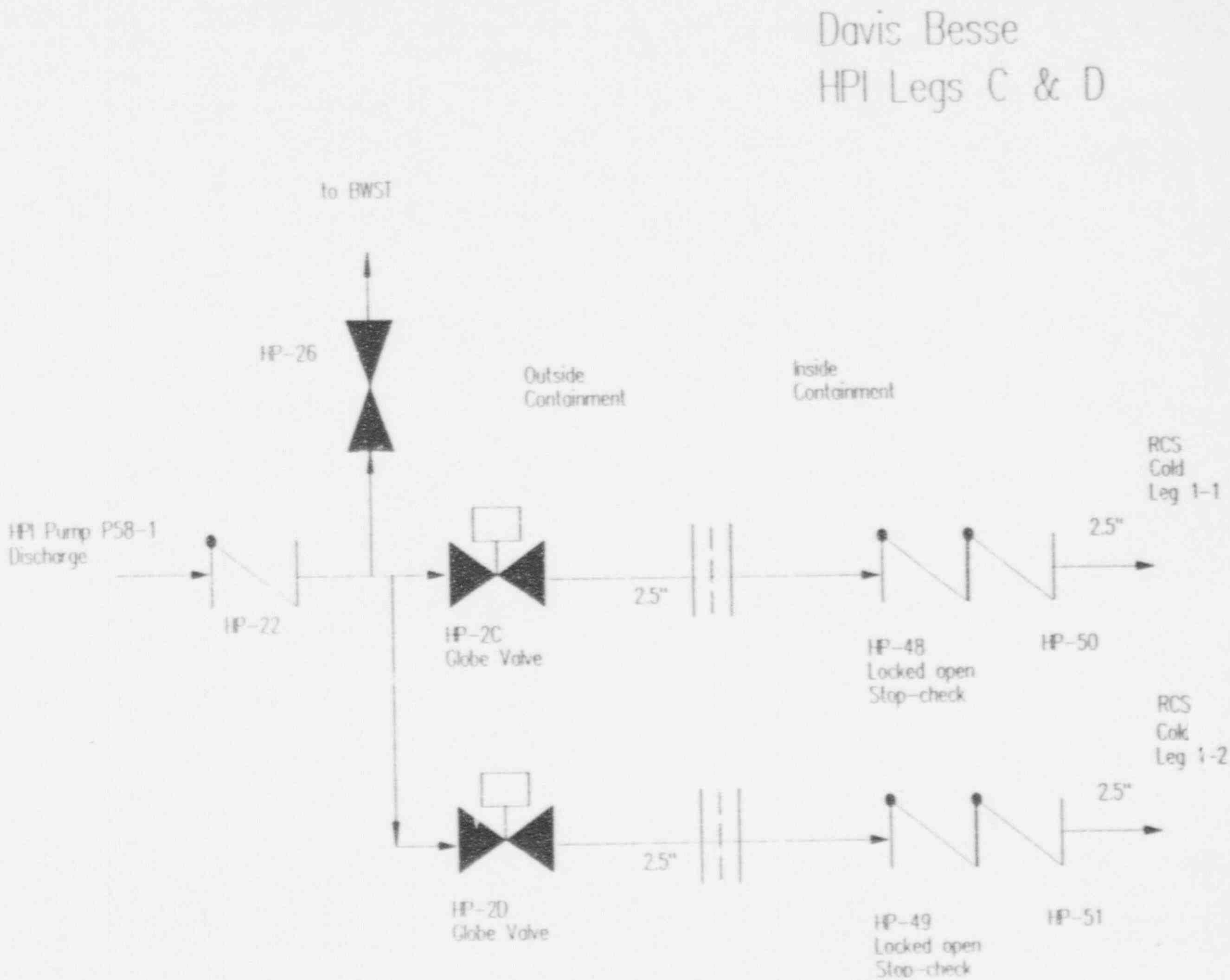


Figure 6. Schematic Diagram of the High Pressure Injection Interface.

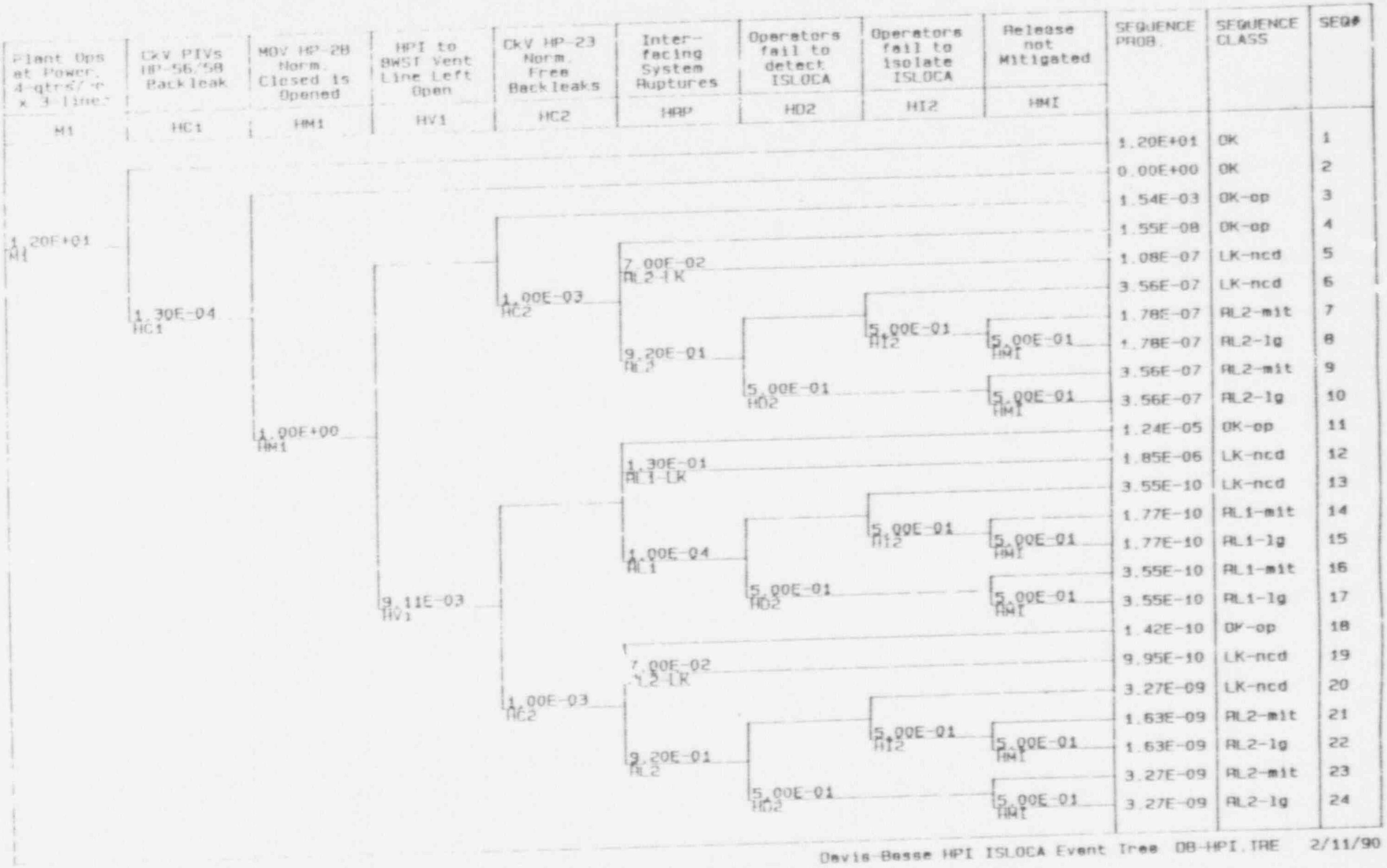


Figure 7. High Pressure Injection ISLOCA Sequence Event Tree.

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 - 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 of this is that the pressure boundary check valves are normally closed with a 2200 psi differential pressure across them.

HC1 - 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 - MOV HP-2B(C,D) Normally Closed is Opened. 1.0

The nodal probability value is based on the routine quarterly stroke tests of MOVs HP-2B, C, and D.

HV1 - HPI to BWST Vent Line Left Open. 9.11E-3

This event models the possibility that the 3-inch 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 quarterly flow tests of the HPI pumps. It is therefore possible that this line could be left open after the pump test and, along with the preexisting failure of the PIV check valves (HP-58 and HP-56), could allow RCS water to flow back to the BWST when the HPI discharge MOV (HP-2B) is stroke tested. This event is quantified using an HRA task analysis (see Appendix E).

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

If the PIV check valves fail open, and 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 - Interfacing System Ruptures. 0.92/0.007 - 1.0E-4/0.13

This event models the conditional probability that, 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 prrbabil'ties are taken as the fraction of Monte Carlo observa*ions that resulted in large, small, or no ruptures in the HPI system (see Append'x H for the details of this calculation).

HD2 - Operators fai ⌄ detect ISLOCA. 0.5

A number of indicators of a rupture in an interfacing system are available to the control room operators. These indicators are primarily pressure, temperature,, and computer alarms. The probability that the operators will detect an overpressure/ISLOCA situation is estimated through the use of screening values for knowledge-based behavior (see Appendix E). Note that this event does not include the process by which the operators diagnose the situation. All that is included here is detection of overpressurization of an interfacing system, not identification of the cause or the corrective actions.

HI2 - Operators Fail to Isolate ISLOCA. 0.5

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 fail to do so successfully. The probabilities used were derived from screening values that were in turn generated from knowledge-based actions (from Appendix E).

HM1 - Release Not Mitigated. 0.5

Once an accident sequence progresses to core damage and a radioactive release is imminent, there are steps that the operators could take to reduce the severity of the release. Specifically, actuation of the fire protection sprinkler system would provide some scrubbing of fission products, which would mitigate the offsite consequences of the release. Since there are no procedures or training for this action, a knowledge-based screening value of 0.5 (from Appendix E) is used for the probability that the operators will fail to initiate mitigative actions, given that core damage has occurred and a radioactive release is about to occur.

4.1.3. 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 during the shutdown operation in order to ensure adherence to the limits and requirements governing shutdown (e.g., at Davis Besse the cooldown rate is limited to $50^{o}F/hr$), and to be aware of when to initiate DHR operation. Figure 8 shows a schematic diagram of the interface between the DHR Letdown and the RHR The ISLOCA event tree for this interface is shown in Figure 9. The scenario of concern here begins with the premature opening of the DHR letdown line (MOVs DH-11 and DH-12) and is based on the premise that shutdown has begun and 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 $600^{o}F$ to 300 psi and $300^{o}F$. The lower end of the range would seem more likely in those cases where plant shutdown proceeds expeditiously, while the high end of the range might be more probable if the plant has spent an unusually long amount of time in hot standby. One area of concern 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. However, if DH-12 will not open, the procedure instructs the operators to jumper-out the relays in order to bypass the interlock. The danger here is that an operator who has routinely bypassed these types of protective safety features may be more inclined to do so even when such 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 susceptible to the human error of commission of entering DHR cooling prematurely (i.e., when RCS pressure is still above 300 psi).

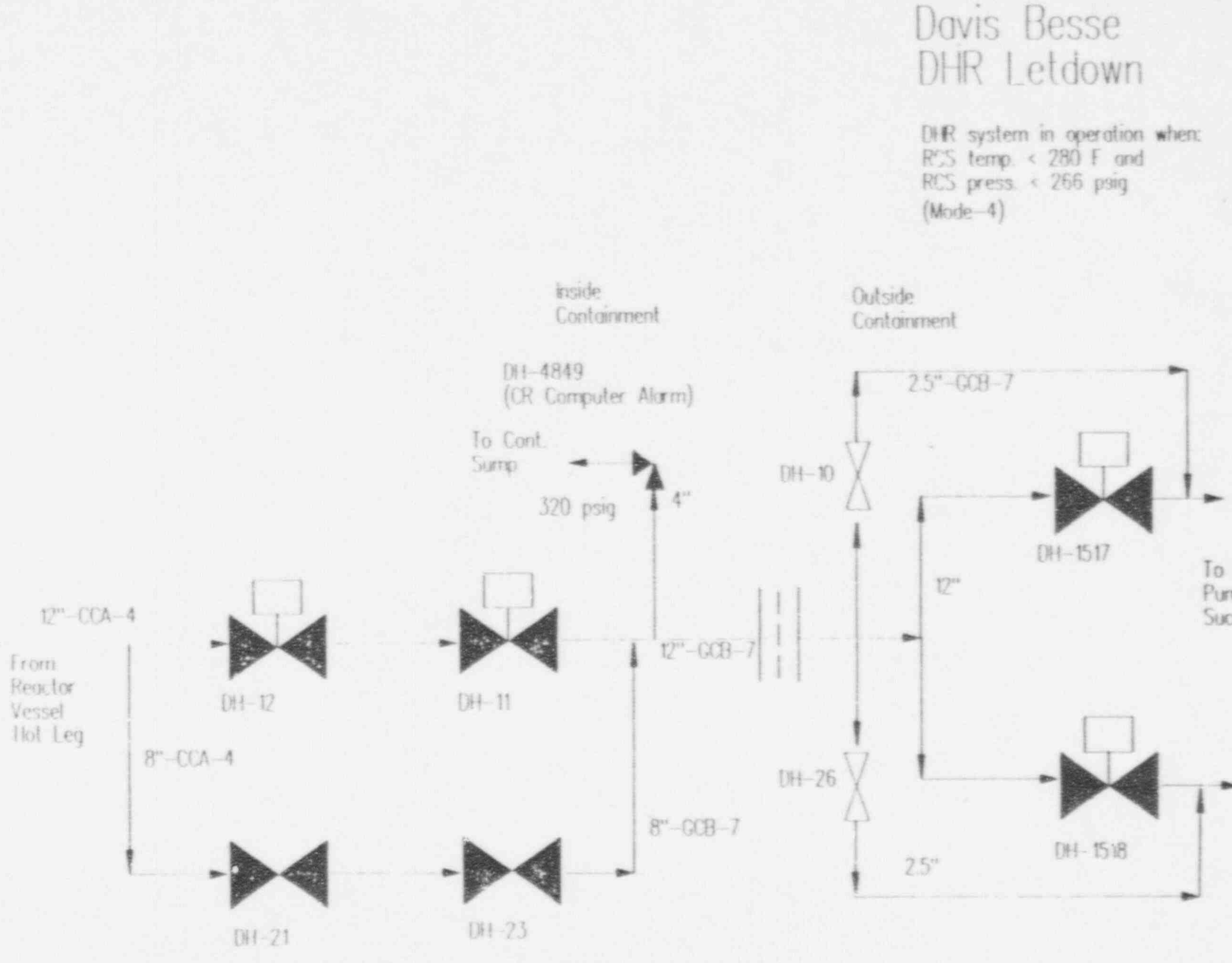
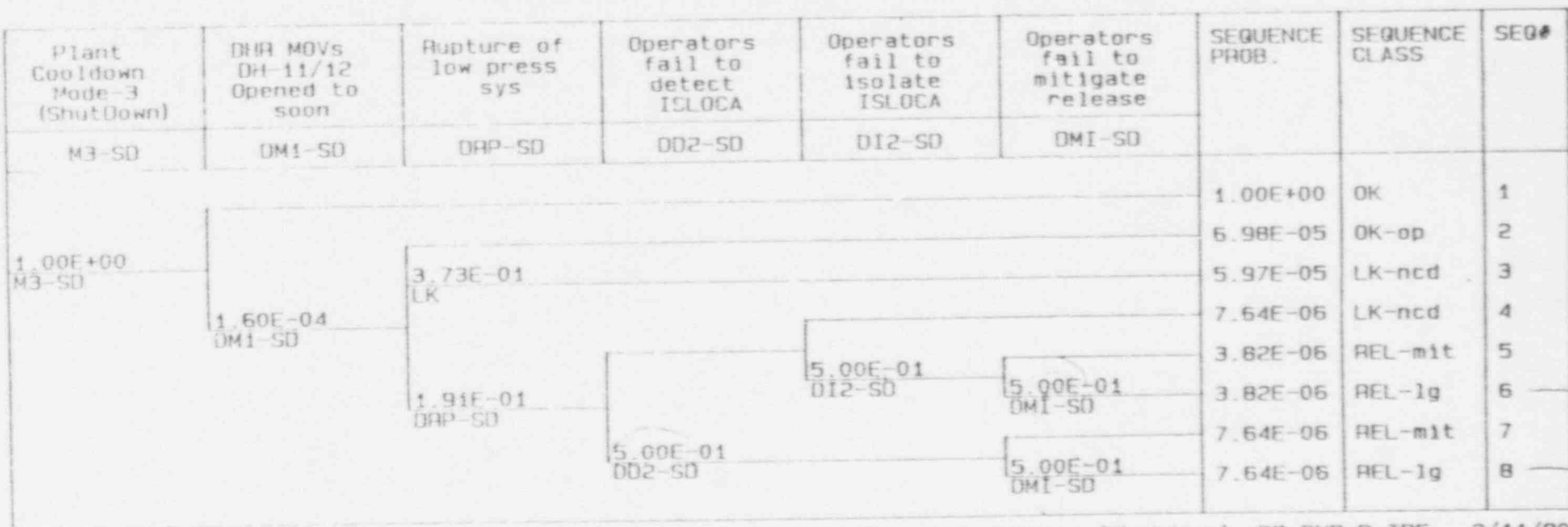


Figure 8. Schematic Diagram of the DHR Letdown Interface (Shutdown).



ISLOCA E.T. for Davis-Besse DHR Letdown (Shutdown) DB-DHR-D.TRE 2/11/90

Figure 9. DHR Letdown System (Shutdown) ISLOCA Sequence Event Tree.

DM1-SD - DHR Motor Operated Isolation Valves are Opened Prematurely. 1.6E-4

This action represents a series of events that begins with the incorrect decision, by a control room operator, to prematurely initiate DHR cooling (i.e., before RCS pressure has dropped to 300 psi) and includes inappropriately bypassing the MOV DH-11 and DH-12 interlocks discussed above. This event is quantified through the use of an HRA task analysis. The nodal probability used is the cumulative HEP for opening MOV DH-11 and 12 before the RCS pressure is reduced to 300 psi. Therefore, the value used in the event tree is the cumulative probability that DH-11 and 12 are prematurely opened when the RCS pressure is between 2150 and 350 psi (see Appendix E).

DRP-SD - Rupture of Low Pressure System Components. 0.1/0.46

This event 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. The values listed are the probabilities that the rupture will be either a large rupture or a small leak, respectively. 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 for various RCS pressures from 400-2100 psig in 100 psig increments (see Appendix F). These local system pressures were then compared to the estimated failure pressures, which were calculated in an independent analysis by IMPELL Corporation (see Appendix G) 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. The rupture pressures for the system components were randomly sampled from the postulated distributions (normal for RCS pressure and lognormal for the failure pressure). The two 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 (0.1, 0.46, and 0.44) represent the fraction of the 10,000 Monte Carlo observations that resulted in large, small, and no ruptures, respectively, weighted by the probability that the valves are opened by the operators (see section 4.4.1 for further discussion on this).

DD2-SD - Operators fail to detect ISLOCA. 0.5

The probability of this event was estimated using an HRA screening value for knowledge-based actions (see Appendix E).

DI2-SD - Operators fail to isolate ISLOCA 0.5

The probability of this event was estimated using an HRA screening value for knowledge-based actions (see Appendix E). This event also includes the diagnostic process.

DMI-SD - Operators fail to mitigate release 0.5

This event was quantified using an HRA screening value for the probability that the control room operators will mitigate the radioactive release by actuating the fire protection sprinkler system. Because there are no procedures or training for this action, a knowledge-based screening value was used (see Appendix E).

4.1.4. 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 10 and the ISLOCA event tree for this system is shown in Figure 11. 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, which are local-manually 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 to prevent inadvertent operation, thus defeating the closure interlock.

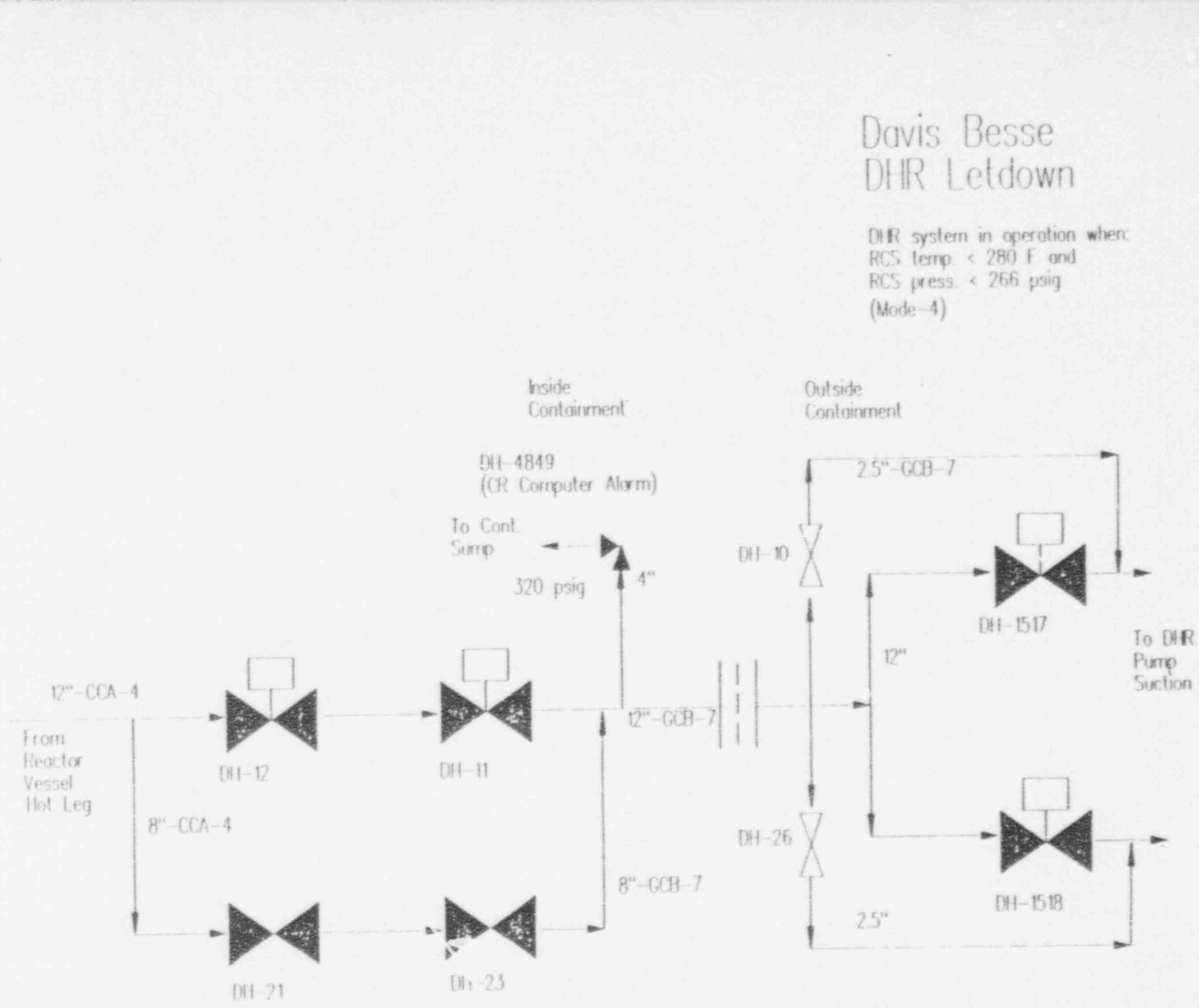


Figure 10. Schematic Diagram of the DHR Letdown Interface (Startup).

Figure 11. DHR Letdown (Startup) ISLOCA Sequence Event Tree.

M3-SU - Plant Heatup. 1.0/yr

This event represents the occurrence of plant heatup, which takes place with the reactor subcritical. Mode 3 operations cover the period from approximately 280°F and 200 psig, to about 500°F and 2200 psig. Heatup is primarily accomplished by 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. Since 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. 4.0E-3

This event models the probability that the DHR system letdown isolation valves, DH-11 and DH-12, are inadvertently left open during plant startup and the RCS is pressurized above 300 psig. DH-11 and DH-12 are 12-inch motor operated gate valves that are interlocked to automatically close when the RCS pressure reaches 300 psi. However, normal plant procedure at Davis-Besse is to maintain the valves in a disabled state by removing their control power. This is done during power operation to prevent inadvertent opening and during plant shutdown to prevent inadvertent closure that would isolate the DHR system. The only time valve control power is energized is when the valves are to be operated. This event is quantified using an HRA task analysis described in Appendix E.

DIL-SU - Pressurizer Heater Interlock Fails. 9.5E-5

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. This will prevent the further pressurization of the RCS above 300 psig. This event models the probability that the interlock fails to disable the pressurizer heaters, and is quantified using a fault tree development that accounts for both hardware and miscalibration faults. The fault tree is shown on Figure D-9 in Appendix D and is quantified using data from Appendix B.

DM2-SU - DHR Bypass Manual Valves DH-21 and DH-23 Left Open. 1.8E-2

This event models the probability that valves DH-21 and DH-23 are left open following their use during a shutdown. Opening these valves is necessary to stroke test valves DH-11 and DH-12. These valves have no remote position indication or hardware control (they are administratively controlled) and are not mentioned in the plant startup procedures. This event is quantified using an HRA task analysis model described in Appendix E.

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 (a 4-inch R/V on the 12-inch letdown line) but would provide a highly visible mechanism for informing the control room operators that the situation was not normal. In addition to the outlet temperature indicator located in the control room, the R/V discharges to the containment sump, which is also instrumented. Furthermore, upon opening at its setpoint of 320 psig, the R/V will pass approximately 1800 gpm that cannot be replenished by the make-up system, producing a drop in pressurizer level. Therefore, if the R/V 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 - Operators Fail to Detect Overpressure in the DHR System. 1.0E-5/1.5E-2

There are two situations of interest for this event, depending on whether or not relief valve DH-4849 opens. If the R/V opens, the RCS will lose approximately 1,800 gpm to the containment sump. In addition, the R/V contains a thermocouple on its discharge such that the outlet temperature can be monitored from the control room (this is also a computer alarm point, but given the lack of importance associated with the Davis-Besse computer alarm system, very little benefit is ascribed to it). Therefore, given the successful operation of the DH-4849 relief valve, the probability of the operators failing to detect an abnormal situation is estimated at 1.0E-5 (or 1 failure in 100,000 opportunities). The second situation examines the cases when DH-4849 fails to open when demanded. In these scenarios the operators must rely on less obvious indications to detect the abnormal valve lineup. These indications are primarily pressure and temperature (not alarmed) at various points in the DHR system. This case was quantified at 1.5E-2 failures per demand using an HRA task analysis. Both values are taken from Appendix E.

DI1-SU - Operators Fail to Isolate the RCS from the DHR System. 7E-4/7E-3 - 3E-4/3E-3

This event represents the probability that the operators will either fail to properly diagnosis the problem or, after successful diagnosis, will fail to properly perform the necessary corrective action. There are four cases analyzed, depending on which pair of DHR letdown valves is open and on whether or not relief valve DH-4849 opens properly. As mentioned earlier, DH-11 and 12 are motor-operated valves that normally have control power removed. However, the control circuits for these valves are wired such that even after control power has been removed, their position indicators function properly and valve position is always displayed in the control room. Conversely, DH-21 and 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 as an open DHR letdown line. This event is quantified using an HRA task analysis as described in Appendix E.

DRP-SU - Rupture of the Interfacing System. 1.0

The previous events were evaluated based on the scenario that RCS pressurization would continue until the sequence was terminated by closing the DHR letdown valves or a rupture occurred in the DHR/LPI system. Therefore, by definition, 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 - Operators fail to detect ISLOCA. 0.5

This event is quantified using an HRA screening value for a knowledge-based action (see Appendix E).

DI2-SU - Operators fail to isolate ISLOCA. 0.5

The probability of this event is estimated from an HRA screening value for a knowledge-based action that also includes the process of diagnosing the situation (see Appendix E).

DMI-SU - Operators fail to mitigate release. 0.5

An HRA screening value (knowledge-based action) is used to estimate the probability that the control room operators will mitigate the radioactive release by actuating the fire protection sprinkler system.

4.1.5. 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 12. The ISLOCA event tree for this system is shown in Figure 13. This interface represents the classical V-sequence configuration of two check valves in series, forming the pressure

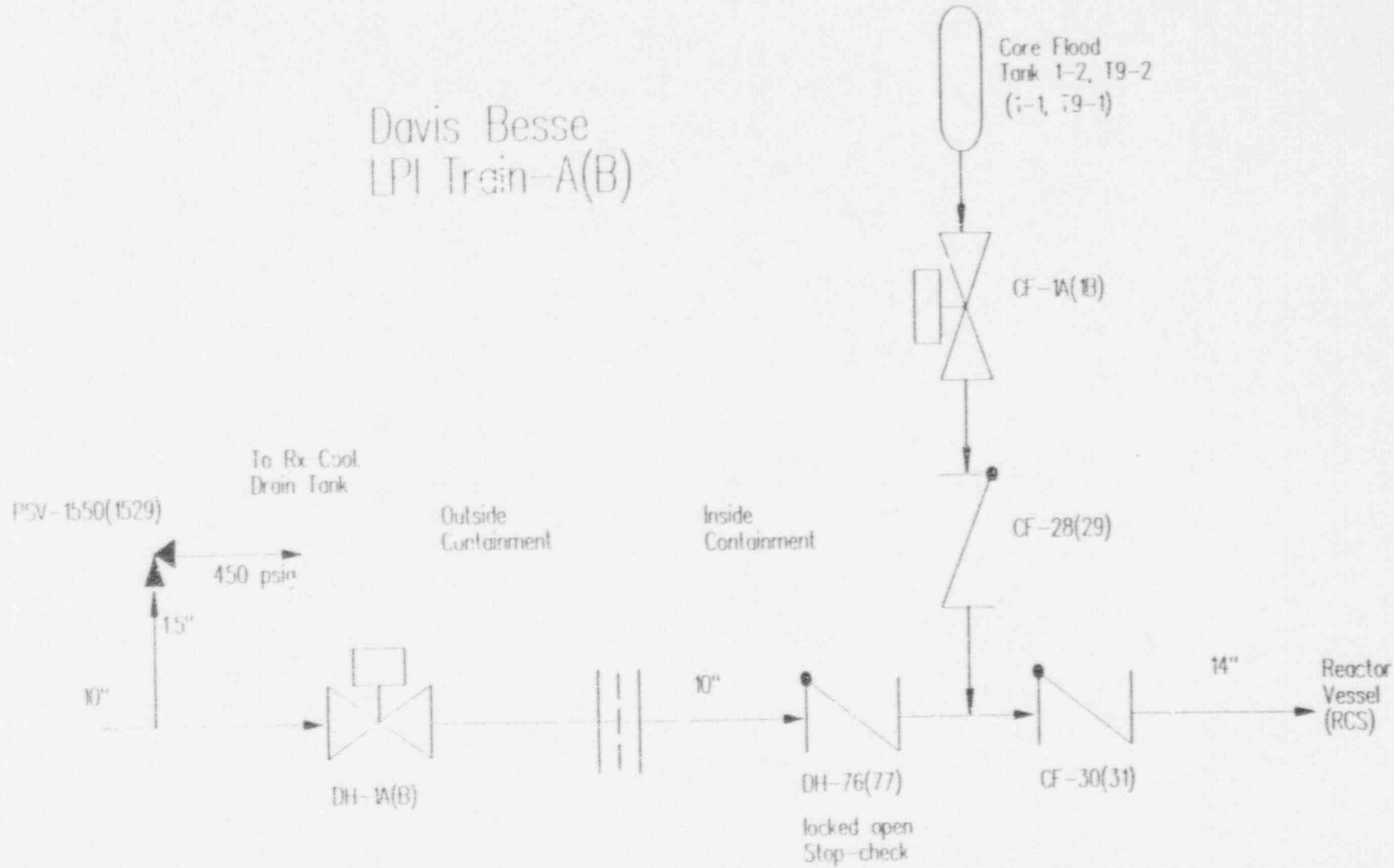


Figure 12. Schematic Diagram of the Low Pressure Injection Interface.

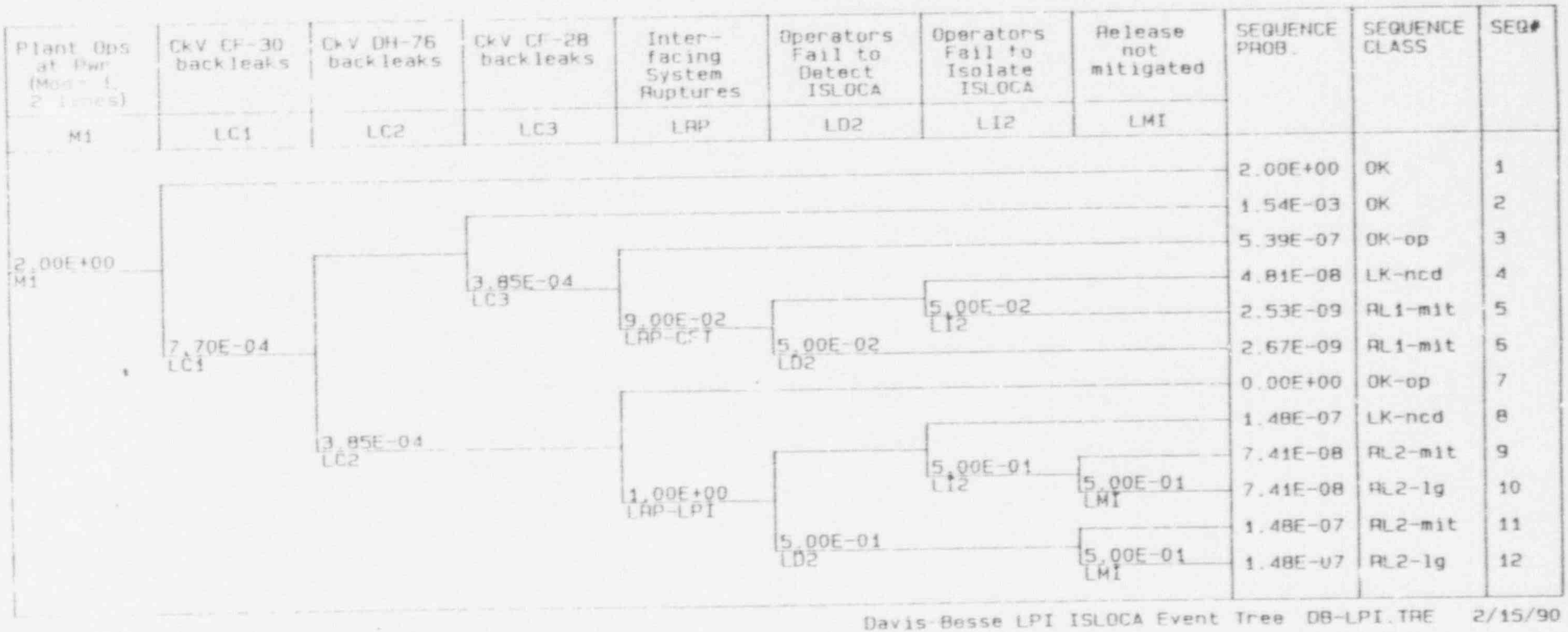


Figure 13. Low Pressure Injection ISLOCA Sequence Event Tree.

isolation boundary between the RCS and LPI system. The system is comprised of 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 the two LPI system injection lines.

LC1 - Backleakage of Pressure Isolation Check Valve CF-30. 7.7E-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 discharge check valves is used to quantify this event. Because of the environment and service the CFT discharge check valves see, they experience 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, since such leak rates overpressurize the interfacing system slowly, resulting in a very high likelihood of detection and correction 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.

LC2 - Check Valve DH-76 Backleaks. 3.9E-4

Check valve DH-76 is also leak tested; therefore, this event is quantified using the same data as event LC1. However, assuming event LC1 is the initiating event, subsequent failures should be modeled as unavailabilities, which are calculated as 1/2(lambda)t, instead of the probability equation of (lambda)t. Therefore, the value used for event LC1 is divided by two.

LC3 - Check Valve CF-28 Backleaks. 3.9E-4

Since 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 Core Flood Tanks. 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 calculated below.

The CFT has two likely failure modes, these being cylinder rupture and plastic collapse head buckling (see Appendix G, Table 2-11), which at $600^{\circ}F$ have associated failure pressures 3130 psi and 3330 psi, and uncertainty factors of 0.24 and 0.27, respectively. Assuming the failure pressure is lognormally distributed, the natural logarithm of the failure pressure generates a normal distribution, which can then be standardized to a mean of zero and a variance of one. This allows the probability that the failure pressure is below 2200 psi (the RCS system pressure) to be calculated from tabulated standard normal curve areas (see Appendix D).

LD2 - Operators Fail to Detect ISLOCA. 0.05/0.5

The location of the failure determines the likelihood that it will be detected in a timely manner. The CFTs are well instrumented and regularly monitored, as required by the plant's Tech. Specs. Since there are procedures to address abnormal conditions in the CFTs, this event becomes a rule-based action. Therefore, a screening value of 0.05 is used for scenarios that involve overpressurization of the CFTs. Otherwise, in a scenario that overpressurizes the LPI system, because of the absence of either procedures or training, event LD2 is assigned a knowledge-based value of 0.5 (see Appendix E)

LI2 - Operators Fail to Isolate ISLOCA. 0.05/0.5

As described above for event LD2, scenarios that involve the CFTs are quantified at 0.05 and scenarios that involve the LPI system use a value of 0.5.

LMI - Release not Mitigated. 0.5

This event only applies to the case where the failure occurs in the LPI system. Again, a knowledge-based screening value of 0.5 (from Appendix E) is used for the probability that the control room operators fail to take action that will mitigate the radioactive release (e.g., actuation of the fire protection sprinkler system). For the case where the rupture occurs in the CFT, the failure is located inside containment; therefore, all releases will be mitigated. In addition, this scenario results in an anticipated design basis LOCA in which the ECCS would likely be effective in preventing core damage. However, given its relatively low frequency (about 1.0E-8/Rx-yr) this scenario was not developed further.

4.2. Quantification of ISLOCA Model

Based on the event trees described in section 4.1 (and in more detail in Appendix-D), the total ISLOCA core damage frequency for Davis-Besse is estimated at 3.8E-5/Rx-yr. Table 4.2-1 provides a breakdown of this frequency by sequence and release category. The dominant sequence is the premature opening of DH-11 and DH-12 during shutdown (identified as the DHR-SD sequence). The results show that, the human error initiated sequences (i.e. DHR-SD, DHR-SU, and MU&P) contribute much more to the core damage frequency than the multiple passive hardware failure sequences (i.e. LPI and HPI). The hardware dominated sequences are similar to the classical V-sequence category that are typically examined in current

PRAs. The relative insignificance of the hardware dominated sequences are supported by historical experience, which indicates that improper valve lineups are potentially a more severe event. Further, the historical experience indicates that sudden, catastrophic failures of valves are virtually nonexistent. Consequently, the use of less conservative hardware failure probabilities and inclusion of human error contributions to ISLOCA sequences, produces an overall ISLOCA CDF significantly higher than those reported in past PRAs.

Table 4.2-1. Davis-Besse Plant Damage State Frequency from ISLOCA Sequences (Frequency per Reactor-year).

Sequence	Lg Release	Mit Release	Leak-ncd	OK-op
MU&P	1.4E-6	1.4E-6	1.1E-3	4.0E-2
HPI	5.4E-7	5.4E-7	2.3E-6	1.5E-3
LPI	2.2E-7	2.3E-7	2.0E-7	5.4E-7
DHR-SD	1.2E-5	1.2E-5	6.7E-5	7.0E-5
DHR-SU	5.2E-6	5.2E-6	1.8E-2	5.3E-5
TOTAL	1.9E-5	1.9E-5	1.9E-2	4.2E-2

Total Core Damage Frequency: 3.8E-5/Rx-yr. (Sum of large and mitigated Release frequencies).

Plant Damage State Definitions:

Lg Release - Core damage with a large unmitigated radioactive release.

Mit Release - Core damage, but radioactive release is mitigated.

Leak-ncd - Reactor coolant is lost, but is either too small to be significant or is isolated before core damage occurs (no core damage).

OK-op - Interfacing system is overpressurized, but does not rupture.

4.3. Risk Assessment

As described previously, the consequences of core damage producing, ISLOCA sequences, were estimated utilizing the containment bypass source term from the Oconee PRA. Although, these results were developed using somewhat dated source term technology, and may not be comparable to the current generation of source term estimates. Based on the Oconee results, conditional consequences were calculated, which when combined with the release category frequencies produces an estimate of the ISLOCA risk. The conditional consequences for a range of decontamination factors is listed on Table 4.3-1. Two release categories were used for binning the event tree end states. These categoı [illegible] ɒresent the mitigated and unmitigated (i.e. large) releases. Because tɦ [illegible] :onee containment bypass source term assumed no decontamination factor (DF), DFs were estimated for the two release categories. Based on information from the NUREG-1150 program that estimated DFs for both dry and wet containment bypass releases, a DF=1 is assumed for the auxiliary building release (large or dry release) and a DF=10 for the mitigated release (fire protection sprays or wet release). (NUREG-1150 used a weighted average DF for the w[illegible] release, DF=10 represents the median value of weighted av[illegible] The ISLOCA risks for Davis-Besse are shown on Table 4.3-2.

Table 4.3-1. MACCS Consequence Results for a range of possible Decontamination Factors (Oconee source term, scaled to Davis-Besse power, and the Surry site).

Consequence Measure	DF=1	DF=5	DF=10	DF=100
Population Dose (person-rem, 50-mi.)	2.8E+6	1.3E+6	9.7E+5	2.9E+5
Latent Cancers (total grid)	4.5E+3	1.5E+3	8.9E+2	1.4E+2
Early Fatalities	3.6E-2	3.0E-4	5.8E-5	1.2E-6

Table 4.3-2. ISLOCA Risk for Davis-Besse (Oconee source term, scaled to Davis-Besse power, and the Surry site).

Risk Measure	REL-1g DF=1	REL-mit DF=10	Total
Population Dose (person-rem, 50-mi.)	53	18	71
Latent Cancers (total grid)	8.4E-2	1.7E-2	1.0E-1
Early Fatalities	6.7E-7	1.1E-9	6.8E-7

4.4. Sensitivity Study Results

Because human errors dominate the results for Davis-Besse, the major effort in evaluating the effects of uncertainty, and the sensitivity of major issues on risk, was devoted to the human reliability analysis. The one exception to this is an analysis of the effects of the uncertainty in pipe rupture pressures on the core damage frequency of the DHR-shutdown sequence, which will be described first.

4.4.1. Pipe Rupture Pressure Uncertainty.

The DHR-SD sequence was chosen for evaluating the effect of the uncertainty in pipe rupture pressures on the core damage frequency, for two reasons. First, it represents the dominant core damage sequence, and second it is analyzed on a weighted average of the range of possible system pressures. That is, system rupture calculations were performed for RCS pressures ranging from 400 to 2100 psig. The rupture probabilities were then weighted by the probability the control room operators would prematurely open the DHR-letdown isolation valves (DH-11 and DH-12). The human error probabilities, in turn are dependent on the RCS pressure such that it is 100 times less likely that the valves would be prematurely opened at 2100 psig compared to 400 psi. This process produced a conditional large leak probability of 0.19, conditional on the premature opening of DH-11 and DH-12 at some unspecified RCS pressure between 400 and 2100 psig.

The uncertainty in the pipe rupture pressure is expressed as the logarithmic standard deviation. The failure pressure is postulated to be lognormally distributed and the logarithmic standard deviation describes the spread in the corresponding normal distribution [i.e. the distribution of log(failure pressure)]. The best estimate of this parameter, which is used as the base case, is 0.36. The sensitivity case was calculated assuming the uncertainty in the failure pressure could be reduced such that the logarithmic standard deviation would be 0.1. Tables 4.4-1 and 4.4-2 show the pressure dependent system rupture probabilities for the two cases. The results on these tables show that for sequences associated with normal RCS operating pressures, the system rupture probability is indistinguishable. Indeed, at an RCS pressure of about 1400 psig, the rupture probabilities are esentially the same. Table 4.4-3 compares the effects on the DHR-SD sequence core damage frequency for the two cases. Overall, the sensitivity case produced a CDF that is reduced to 50% of its base case value. This difference is not considered to be significant based on the assumed change in the standard deviation.

Table 4.4-2. DHR System Rupture Probabilities (Normalized to HEP of prematurely opening DH-11/12) as a Function of RCS Pressure. (Pipe failure pressure log(std dev) = 0.1)

RCS Pressure (psig)	HEP	System Rupture Prob large	small	no-leak	HEP Norm'zed Sys Rupt Prob large	small	no-leak
2100	4.0E-07	1	0	0	4.0E-07	0.0E+00	0.0E+00
2000	5.2E-07	0.9999	0.0001	0	5.2E-07	5.2E-11	0.0E+00
1900	6.8E-07	0.9999	0.0001	0	6.8E-07	6.8E-11	0.0E+00
1800	8.9E-07	0.9998	0.0002	0	8.9E-07	1.8E-10	0.0E+00
1700	1.2E-06	0.9993	0.0007	0	1.2E-06	8.2E-10	0.0E+00
1600	1.5E-06	0.99466	0.00534	0	1.5E-06	8.2E-09	0.0E+00
1500	2.0E-06	0.97576	0.02424	0	1.9E-06	4.8E-08	0.0E+00
1400	2.6E-06	0.89134	0.10866	0	2.3E-06	2.8E-07	0.0E+00
1300	3.4E-06	0.69044	0.30956	0	2.4E-06	1.1E-06	0.0E+00
1200	4.5E-06	0.3867	0.6133	0	1.7E-06	2.7E-06	0.0E+00
1100	5.8E-06	0.14036	0.85963	0	8.2E-07	5.0E-06	0.0E+00
1000	7.6E-06	0.03086	0.96894	0.0002	2.3E-07	7.4E-06	1.5E-09
900	9.9E-06	0.00522	0.99398	0.0008	5.2E-08	9.9E-06	8.0E-09
800	1.3E-05	0.0008	0.9849	0.0143	1.0E-08	1.3E-05	1.9E-07
700	1.7E-05	0.00012	0.898	0.10188	2.0E-09	1.5E-05	1.7E-06
600	2.2E-05	0.00006	0.61286	0.38708	1.3E-09	1.4E-05	8.6E-06
500	2.9E-05	0.00004	0.198	0.80196	1.2E-09	5.7E-06	2.3E-05
400	3.8E-05	0.00002	0.0125	0.98748	7.6E-10	4.7E-07	3.7E-05
Total	0.000160				0.091567	0.463764	0.444667

Table 4.4-3. Sensitivity of Pipe Rupture Pressure Uncertainty on DHR-SD Sequence Core Damage Frequency (per Rx-yr). Base case, logarithmic std.dev. = 0.36; sensitivity case, logarithmic std.dev. = 0.1.

Sequence Class	Base Case	Sensitivity Case
OK-op	6.98E-5	7.11E-5
LK-ncd	6.73E-5	7.79E-5
REL-mit	1.15E-5	5.49E-6
REL-lg	1.15E-5	5.49E-6

4.4.2. HRA Method Sensitivity.

Two separate sensitivity studies were performed relative to the human reliability analysis performed for this program. First, a sensitivity study based on the method of calculating HRA values was performed. This case examined the effect on ISLOCA core melt frequency when a typical HRA screening analysis is utilized instead of the detailed task analysis, which was done for the base case quantification. The second sensitivity case studies the effect of optimizing the conditions at Davis-Besse with respect to operator performance. In this case, the performance shaping factors are assumed to result in positive influences and produce relatively low HEPs.

The sensitivity study comparing human error probabilities (HEPs) based upon the detailed plant information (the base case) with those that were derived from screening values showed that there was a statistically significant difference between the two sets of HEPs. Inspection of the screening analysis details in Appendix-E, reveals that the sensitivity case HEPs were more conservative for 56% of the actions, less conservative for 20% of the actions, and for 24% of the HEPs there was no difference. In addition, the DHR-SD sequence (premature opening of DH11 & DH12 during shutdown) shows no screening values for the sensitivity case because this sequence would not have been identified without the benefit of detailed walkdowns and operator interviews at the plant. This in conjunction with the differences cited above makes a strong case for why the detailed HRA is necessary for a complete picture of an ISLOCA.

The final step for this sensitivity analysis was the calculation of core melt frequencies for the screening data set and the base case data set. The end result of the analysis shows a substantial difference in risk measures (see Table 4.4-4). Given that the base case data set is the true best estimate case, these core melt frequencies indicate the differences which can be attained with the benefit of a more detailed HRA.

The second HRA sensitivity analysis was conducted to determine if modifications to the human machine system (performance shaping factors) would result in significant gains in operator performance. Since many of the sequences analyzed were not covered by procedures, and were not specifically trained on, it seemed reasonable that these types of modifications might improve the probability of successful task completion. Additionally, some sequences did not have the benefit of instrumentation which would indicate the status of valves, critical to avoiding ISLOCA sequences. Lastly, some instances of operator performance did not have the benefit of recovery factors in the form of a second individual independently checking and signing off on task completion. Therefore, the following performance shaping factors were optimized and the resulting change in the human error probability calculated for all scenarios.

1. Procedures- HEPs were calculated on the basis of the startup, shutdown, or quarterly stroke test being upgraded to reflect the appropriate cautions, notes, or warning. An example would be noting the importance of correct valve line-ups for HP27 & HP29, and the correct line-up for DH21 and DH23 in terms of the potential for ISLOCA.
2. Instrumentation- HEPs were calculated on the basis of the presence of a valve status board in the control room and on the presumption that ambiguous information on pressure, temperature, level, and flow were available to the crew.
3. Training- HEPs were calculated on the basis of two significant improvements; the existence of an ISLOCA procedure formally trained to by control room and EO personnel and that similarly, there were training and procedures for the handling of computerized alarms on the control room CRT.
4. Recovery- HEPs are calculated so that all tasks are covered by procedures and second operator, shift supervisor , or I&C or maintenance foreman, where appropriate, signs on task performed.

Using the base case and sensitivity study HEPs, core melt frequency was recalculated and is shown in Table 4.4-4. It should be noted that a significant reduction in CDF and risk resulted from optimization of the HEPs even though the changes to procedures, training and instrumentation are not considered to be major. The information presented in this table also shows that the use of HEPs based on screening values, rather than HEPs developed through plant specific analysis, would produce significantly higher core melt frequencies and risk values.

Table 4.4-4. Sensitivity of HRA technique on both CDF (per Rx-yr) and risk.

Risk Measure	Base Case	Optimum HEPs	Screening HEPs[a]
REL-lg	1.9E-5	8.1E-9	1.4E-2
REL-mit	1.9E-5	8.0E-7	1.4E-2
Lk-ncd	1.9E-2	7.2E-3	3.3E-2
OK-op	4.2E-2	4.2E-2	2.0E-1
Pop-Dose	71	0.8	53,000
Lat-Cancers	0.1	7.5E-4	75
Early-Fat	6.8E-7	3.4E-10	5.0E-4

a. Note that the screening evaluation would not have identified the DHR-SD sequence, which involves a human error of commission. This sequence is not included in the Screening HEP totals.

5. CONCLUSION AND RECOMMENDATIONS

A methodology for evaluating an ISLOCA has been developed and has been applied to the Davis-Besse Nuclear Power Station. This methodology has been successful in providing insights on the relative contribution to core melt frequency and risk, of both hardware faults and human actions. The results indicate that human errors of commission, latent faults of equipment and normal procedural tasks can combine in an ISLOCA sequence to produce potentially serious consequences. However, the methodology was also used to identify one potential means of reducing these contributions to risk. In addition, the method has identified significant ISLOCA sequences applicable to non-power producing operating modes. Following are the conclusions and recommendations relating to Davis-Besse followed by a preliminary discussion of the relationship of these results to the general population of nuclear power plants.

5.1. Davis-Besse Specific

Since a PRA for Davis-Besse is not publicly available, the ISOLCA results can not be directly compared with a previously developed plant specific study. The best material that could be found for comparison purposes were results from the Brookhaven ISLOCA study for Oconee, which indicated that the expected core melt frequency from ISLOCA was about $1x10^{-6}$. The Davis-Besse results are about a factor of forty larger. This difference is attributed primarily to the inclusion of the influence of human errors on the initiation and progression of an ISLOCA. ISLOCA sequences that comprise only hardware failures (i.e. V-sequence), were found to be relatively insignificant with respect to the total ISLOCA risk. This is attributed to the exclusion of small leak rate failures (i.e. < 200 gpm) in the failure rate data used in the present study.

In the pressure fragility analysis of the interfacing systems, existing relief valves provide virtually no protection against ISLOCA events. Typically, relief valves in the interfacing systems are designed to mitigate the occasionally pressure transient associated with routine valve realignments and pump starts and stops. The pressures generated in ISLOCA events simply overwhelms the relatively small relief capacity of these valves. In the RELAP5 runs without relief valves, equilibrium pressure was usually reached in less than 10 seconds. With the relief valve the system would required approximately 30 seconds to reach equilibrium. A relief valve might be effective in protecting the portion of a system downstream from a restricting orifice, provide the orifice size was comparable to the relief valve size and the relief valve was also downstream of the orifice.

Based on the pressure fragility and rupture analyses, the DHR heat exchanger was identified as having a relatively low pressure fragility. The large diameter, low pressure pipe was also a very likely candidate for rupture. Specifically, the schedule 10S and 20 pipe on the suction side of the DHR pumps was estimated to have a very high rupture probability in the DHR sequences.

In regard to the potential for human errors at Davis-Besse, the lack of awareness about interfacing system LOCAs and the casual attitude of plant personnel in dealing with the pressure isolation boundaries appear to be the most significant contributors. This is demonstrated by the approach taken by the plant on some routine operations performed relating to the RCS pressure isolation boundary. These are:

- The routine, quarterly stroke testing of HPI MOV HP-2A, results in the tripping of pressure switch PSH-2883A that actuates a computer alarm in the control room. The operators "don't worry about it" anymore since they know it is caused by HP-2A test. If it were actually needed, this computer alarm is the only direct indication that RCS was backleaking into the HPI lines.
- In order to deal with a large "dead-band" in the interlock on DH-12 that prevents the valve from being opened when RCS pressure is above 266 psig, the plant shutdown procedures instruct the operators to

jumper out the DH-12 interlock when initiating DHR cooling.

- Because of past problems at Davis-Besse with DHR letdown motor operated valves DH-11 and DH-12 automatically closing while the plant was shutdown and isolating the DHR system, the plant has proceduralized the practice of maintaining DH-11 and DH-12 in a perpetual disabled state. That is, DH-11/12 always have their control power removed. Consequently, although these valves are interlocked to automatically close when the RCS pressure rises to 300 psig, the valves will only operate when control power is restored, which is done only when the control room operators wish to change their position.

In summary, the ISLOCA frequency at Davis-Besse is strongly influenced by:

1) Poorly written procedures,
2) the lack of training on ISLOCA sequences, which contributes to the general unawareness of plant personnel about the possibility and consequences of ISLOCA type events, and
3) procedures that instruct personnel to routinely bypass or jumper out protective features or interlocks that otherwise could prevent the initiation of an ISLOCA.

As shown in the sensitivity case that estimated the effect on risk of optimizing the human performance, a significant improvement in risk is likely achievable with changes to procedures, training and instrumentation.

A number of general observations are worthwhile mentioning based on their insignificance with respect to ISLOCA risk. Area effects associated with interfacing system ruptures and the resulting water spray and accumulation, were assessed. At Davis-Besse, the emergency core cooling systems (ECCS) are adequately separated. Each postulated rupture was reviewed based on the premise that all equipment in the same room as the rupture was failed because of the leak. The worst situation occurs in the DHR sequences were both trains of DHR would likely fail. However, given that the leak is isolated, the plant would be able to rely on the operation of the power conversion system and/or the auxiliary feedwater system to cool the reactor and maintain it in a stable condition. This option was available for every ISLOCA sequence postulated. That is, provided the interfacing system rupture was isolated (and in every sequence examined valves were in place that could be used for this), there were systems available to prevent core damage. However, the converse was also estimated to be true. Because of the small BWST makeup capability (about 150 gpm) and the time expected for the ECCS to drain the BWST (approximately 4 to 8 hours) if the rupture could not be isolated, core damage is very likely. That is, without isolating the rupture and terminating the leak, successful operation of plant safety systems would delay but not prevent core damage.

5.2. Generalized Conclusions

Caution should be exercised when attempting to extrapolate the results of a single sample to estimate the performance of the entire commercial nuclear power industry. The Davis-Besse analysis has identified some potential ISLOCA issues, but the completeness and typicality of the results for other plants has not been determined. What can be said is, based on the experience gained from the analysis of Davis-Besse, the most important concern regarding ISLOCA risk centers on the reliability of the plant personnel. It is unwise to conclude that human reliability represents the entire potential for ISLOCA events, but for Davis-Besse, the human errors dominate the ISLOCA risk and this could be an issue for other plants as well. Therefore, a major emphasis in any evaluation of ISLOCA events should be the assessment of the potential for human errors. Specifically, this involves judging the adequacy of plant procedures, personnel training, and personnel awareness of the potential for and consequences from an ISLOCA event. For Davis-Besse there were no procedures or training on ISLOCA events. Furthermore, there was no significant awareness on the potential for, or the consequences of violating the RCS pressure isolation boundary. To generalize, the understanding by the plant personnel of the importance of maintaining the pressure isolation boundary, and recognizing the potential for an ISLOCA event and its consequences, has a dramatic effect on ISLOCA risk.

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Appendix A

Historical Experience Related to ISLOCA Events

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Historical Experience Related to ISLOCA Events

APPENDIX-A

HISTORICAL EXPERIENCE RELATED TO ISLOCA EVENTS

A computerized search of licensee event reports (LERs) was performed to collect and analyze those events that can loosely be categorized as ISLOCA precursors. On the succeeding pages is the resulting list such of events, along with a brief description of each event. After reviewing the LERs, a number of generalizations were made. Those events that resulted in an overpressurization and/or leak out of the RCS typically involve either a series of human errors, inadequate procedures, or existing hardware failures in combination with a human error or inadequate procedure. A few of the more applicable events are described in more detail below.

McGuire

While stroke timing a valve at McGuire Unit 2 (9/5/89), operators inadvertently released 200 gallons of primary coolant to the pressurizer relief tank (PRT) and 2000 gallons to the auxiliary building, bypassing containment, over a thirty second period. The 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 pre-test status, operators subsequently opened another valve which 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 approximately a 30 minute period of time. Control room personnel were notified of the flooding in the auxiliary building by Radwaste Chemistry personnel.

A year prior to this 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 ... the changes only addressed the operation of valves which were

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A year prior to this 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 which were involved in that particular event. The valves which were involved in the 9/5/89 event were overlooked by that procedural change. It was noted that operators' attention was focused on preventing the reoccurrence of the 1988 event, thereby ignoring other overpressurization and backleakage pathways.

In addition, the procedure required a review of system conditions prior to initiation of the test, it did not adequately address all conditions which should have existed to prevent this event. 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 which could arise as a result of using the procedure. Thus, a combination of procedural inadequacies, training which focused operator attention to prevent a specific event, operator's belief in the adequacy of procedures, and inattention to potential problems contributed to this event.

Biblis

Approximately two years prior to the McGuire Unit 2 event, while conducting restart operations at the Biblis plant in West Germany, operators established a high pressure pathway from the primary circuit to low pressure systems causing an uncontrolled release outside of containment. During restart, operators observed that one of the two isolation valves (a check valve) in the connecting lines between the primary circuit and LPIS was slightly open. The operator attempted to close the valve by opening the other isolation valve, intending to create sufficient differential pressure against the open check valve to cause it to close. The valve did not reseat as intended and, for a period of 7 seconds, coolant was discharged through a relief valve to a low pressure test line, and from there to the annulus and subsequently to the plant stack.

A position indicator alarm in the control room had alerted the operator to the condition of the check valve. The operators ignored the position indication instrument and alarm, believing them to be false. The shift supervisor was not informed of this condition and neither were incoming shifts during shift turnover as required. The open check valve was undetected for the next two shifts. With this check valve open, only 1 check valve prevented overpressurization, possible disablement of one train of the Emergency Core Cooling System (ECCS), and an unisolable leak outside of containment.

Since the approach that was used to close the open check valve was according to the operators' training, the operators were acting on the belief that the valve would respond as intended, and not on the immediate effects which might occur due to overpressurization of low pressure systems. This generic weakness in their operating procedures was resolved by retraining operators on the specific features of the event, by changing procedures for control room alarm review, and by categorizing the alarms associated with the specific valves of this event as high priority.

BWR Testable Check Valves

A study by AEOD (1985) identified 8 events which 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 which 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 of the 5 events were attributed to personnel errors during surveillance testing. The most serious of these events resulted in the contamination of thirteen workers after being sprayed by coolant from a relief valve after it was over-pressurized.

Farley

During a refueling outage at Farley Unit 2 (1987) test and maintenance personnel failed to refill a section of pipe which 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 a cause of the initiating event, administrative controls governing these types of tests and the operations-engineering planning interface also contributed to this event due to inadequate communication.

Trojan

At the Trojan plant (1989) during cold shutdown, one of two residual heat removal (RHR) isolation valves was determined to be inoperable after it was identified that the valve would not close automatically. The valve had been wired incorrectly due to inadequate as-built drawings. 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 during the interval between outages in this condition.

Hatch and Browns Ferry

Similarly, two events at Hatch Unit 2 (1983) and Browns Ferry (1984) were determined to be caused by incorrect installation or assembly of valves which were part of the pressure boundary between the high pressure (RCS) and low pressure ECCS systems. The events were also thought to be due, in part, to a failure to use and follow approved maintenance and assembly procedures.

ANO-1

During a complicated transient at Arkansas Nuclear One Unit 1 (1989), 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. This condition, high piping temperature or pressure, was not instrumented for. Detection was accomplished when taping attached to the pipe began to smoke and set off a local area smoke detector which caused an alarm in the control room. The backflow occurred for approximately 10 to 15 minutes before the fire alarm was observed and investigated.

It is worth noting that control room personnel were involved in an unusual post-trip condition due to several equipment failures which 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 which 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, backleakage 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 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.

Vogtle

While preparing for initial heatup at Vogtle Unit 2 (1989), 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 depressurize the RHR system by momentarily opening two locked-closed valves. 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 by a later shift, both RHR trains were declared inoperable.

The event 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 the condition which opening these valves placed the unit in. During this event, RCS coolant passed from the RHR system to the refueling water storage tank, and from there to the atmosphere. Since the unit had not achieved its initial criticality, however, no radiation was released.

Pilgrim

During preparation for the Reactor Core Isolation Cooling (RCIC) logic function system test (LFST) while at 25% rated power and ascending, 6 circuit breakers to motor operated valves were incorrectly positioned. An Instrument & Control technician, a Control Room Operator, and an Equipment Operator divided the task of positioning the breakers at the local area amongst themselves, 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 hi 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 LCO for RCIC recovery.

No pre-evaluation briefing was conducted by the operating shift prior to preparation for the RCIC LFST, although required by Technical Specifications. Two of the persons 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 some time prior to this event. In addition, the procedure did not include precautions to warn workers of the effect which incorrectly performing the steps would have on the safety system.

Limerick

At Limerick Unit 1 (1989) 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: 1) a lack of detailed procedures in performing the Safe Shutdown Analysis and; 2) a misunderstanding or misapplication of detailed regulatory requirements.

ISLOCA Precursors

Plant	Date	LER No.	Description
Vermont Yankee	12/12/75	75-24	Testable check valve did not seat properly allowing backleakage, during stroke testing of injection MOV, a normally open MOV is closed however it did not close completely. When the injection MOV was opened it allowed backleakage into the RHR system.
Cooper	01/21/77	77-04	Testable check valve in HPCI failed to fully close because of a broken sample probe wedged under the disk. The outboard isol. valve was opened, as required, for the HPCI System Turbine Trip and Init. Logic Surv. Test allowing FW backflow into HPCI sys.
LaSalle-1	10/05/82	82-115	Testable check valve was opened for a test, however when the air pressure was removed the valve failed to reclose.
LaSalle-1	06/17/83	83-067	HPCS testable check valve failed to close after quarterly test. Failure caused by insufficient spring tension of the actuator assembly.
LaSalle-1	09/14/83	83-105	During RHR System Relay Logic Test, injection valves were opened (per proceedure) leaving the injection check valve as the only isolation between RHR and RCS. This valve leaked because of improper timing and packing gland being too tight.
Pilgrim	09/29/83	83-048	Feedwater pressure backed-up thru a partially open injection check valve when a HPCI logic test mistakenly opened (because of miscommunication between the control room operators and the I&C technicians) both HPCI pump discharge valves.
Hatch-2	10/28/83	83-112	Incorrectly assembled air actuator held open a testable check valve for about 4 months. Indicator was apparently re-wired at the time the actuator was reassembled since valve position is indicated in the control room.
Susquehanna-2	05/28/84	84-006	Dual indication (both open and close) prompted CR operators to attempt to reseat testable check valve by opening injection valve (a normally open valve was closed). This allowed backleakage to the RHR heat exchanger.

Browns Ferry-1	06/14/84	84-032	Testable check valve solenoid valve had been reassembled with the air ports reversed probably in Dec. '83 resulting in the valve being held open. Subsequent logic test (operators failed to electrically disarm the MOV) opened a MOV allowing RC into CS
Pilgrim	02/11/86	86-003	During maintenance of electrical cable, a 480V safety-related bus was inadvertently de-energized resulting in the disablement of some primary containment isolation capability.
Biblis-A	11/15/87		Operators attempted to seat a stuck open check valve by opening an MOV on the low pressure side to increase the differential pressure across the check valve. Proceedure didn't work allowing RCS water to leak back into the RWST.
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.
D. C. Cook-1&2	10/28/88		Generic problem with Anchor Darling swing check valves.
ANO-1	01/20/89	89-002	Rx coolant backflowed outside containment thru one HPIS check valve,thru a crossover line, and back into the RCS. HPIS crossover line not designed to handle RCS temperatures. HPIS is used for normal make-up. High wear allowed back-leakage of CV.
Vogtle-2	03/09/89	89-003	Both PIV check valves leaked, and allowed RCS backleakage into RHR system (exceeding tech specs). Operators attempted to depressurize the RHR system by opening two manual valves to RWST.
Trojan	04/09/89	89-009	Auto close/open pressure interlock on RHR letdown isolation valve incorrectly wired
Pilgrim	04/12/89	89-014	During RCIC system logic test, injection valves opened (they were not disabled prior to test), the discharge check valve failed to properly seat and allowed backleakage into RCIC piping.
Sequoyah-2	04/20/89		Charging line flow control valve (62-93) blew its stem leakoff packing allowing normal makup to leak-out before reaching the RCS. This resulted in an abnormally high makup flow rate.

Limerick-1	06/01/89	89-039	PECo determined that a fire in certain plant areas could result in the spurious opening of high/low pressure interface valves.
ANO-2	06/26/89	89-012	During plant heatup, RCS backleaked 3-times thru a SI system check valve. Each time the valve was reseated by injecting water using HPI pump. While shutdown, the CV was inspected and valve disk was found to be not secured to the disk shaft.

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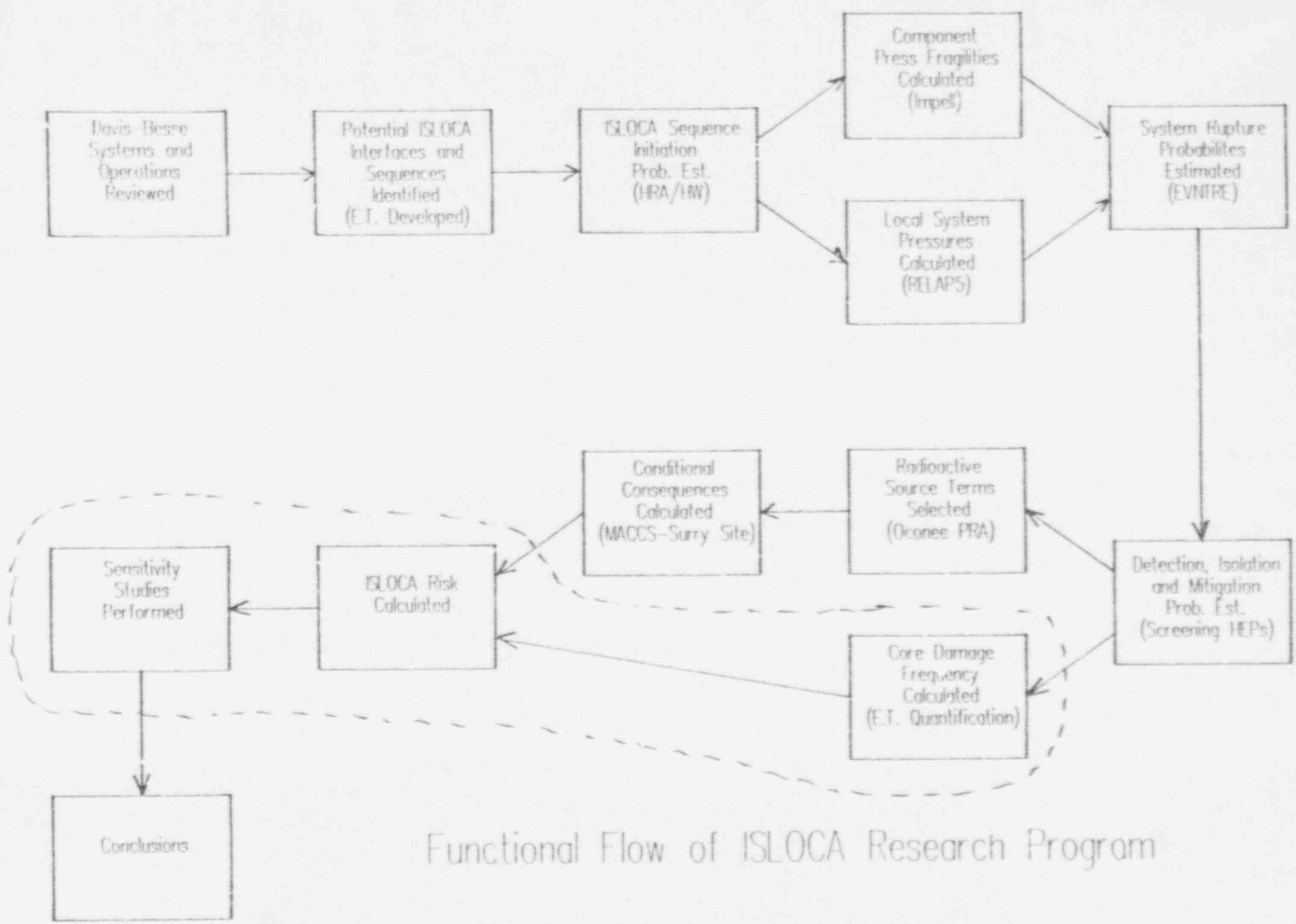
QUANTIFICATION AND HARDWARE SENSITIVITY STUDY RESULTS

FEBRUARY 26/27, 1990

W. J. GALYEAN

B-44

EG&G Idaho, Inc.



Functional Flow of ISLOCA Research Program

QUANTIFICATION OF RESULTS

- o Event trees produce plant damage state frequencies
  - Release-large
  - Release-mitigated
  - Leaks-no core damage
  - OK-overpressure

- o Core Damage Frequency sum of REL-lg and REL-mit

- o Risk Measures:
  - Early Fatalities
  - Latent Cancers (total grid)
  - Population Dose (50-mi.)

DAVIS-BESSE PLANT DAMAGE STATE FREQUENCY FROM ISLOCA SEQUENCES

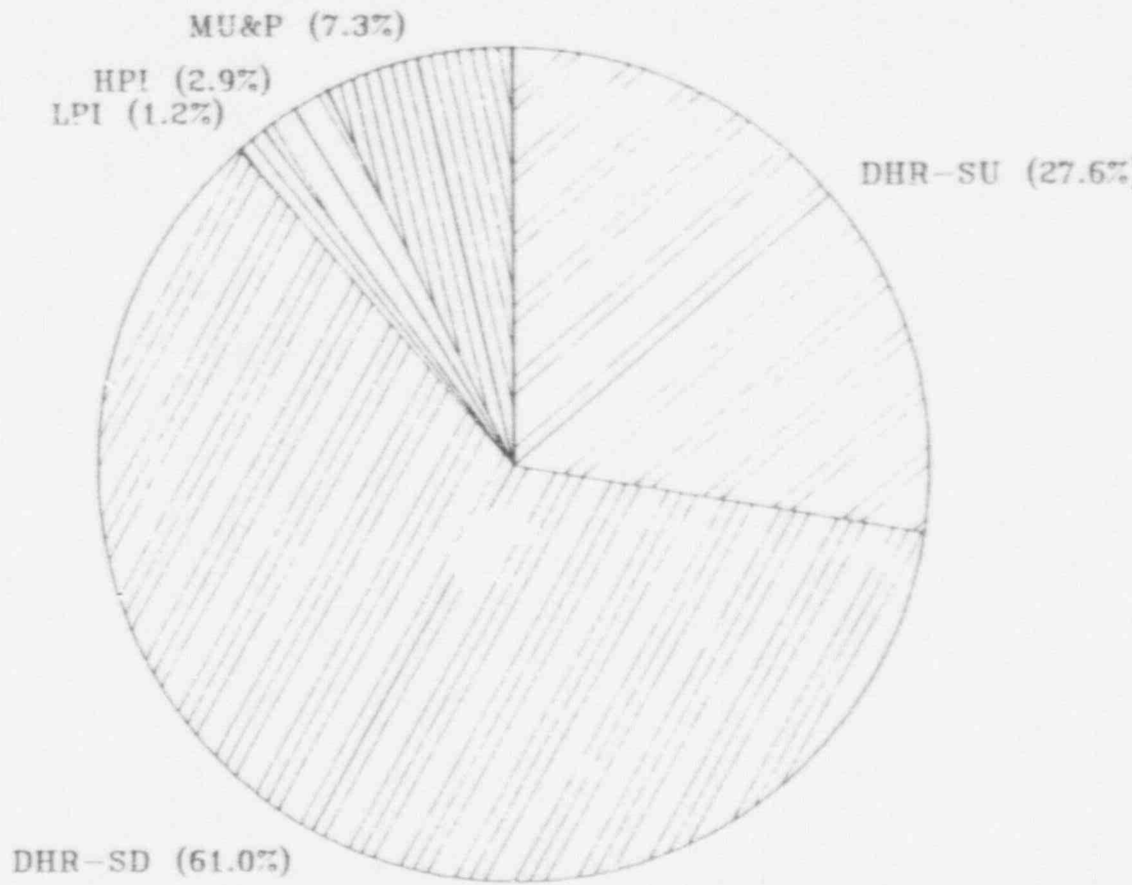
(FREQUENCY PER REACTOR-YEAR)

SEQUENCE	LG REL	MIT REL	LK-NCD	OK-OP
MU&P	1.4E-6	1.4E-6	1.1E-3	4.0E-2
HPI	5.4E-7	5.4E-7	2.3E-6	1.5E-3
LPI	2.2E-7	2.3E-7	2.0E-7	5.4E-7
DHR-SD	1.2E-5	1.2E-5	6.7E-5	7.0E-5
DHR-SU	5.2E-6	5.2E-6	1.8E-2	5.3E-5
TOTAL	1.9E-5	1.9E-5	1.9E-2	4.2E-2

TOTAL CORE DAMAGE FREQUENCY: 3.8E-5/RX-YR. (SUM OF LARGE AND MITIGATED RELEASE FREQUENCIES).

Davis-Besse CDF Distribution

from ISLOCA Sequences



ISLOCA RISK FOR DAVIS-BESSE
(OCONEE SOURCE TERM, SCALED TO DAVIS-BESSE POWER,
AND THE SURRY SITE)

RISK MEASUPE	REL-LG DF=1	REL-MIT DF=10	TOTAL
POPULATION DOSE (PERSON-REM, 50-MI.)	53	18	71
LATENT CANCERS (TOTAL GRID)	8.4E-2	1.7E-2	1.0E-1
EARLY FATALITIES	6.7E-7	1.1E-9	6.8E-7

SENSITIVITY STUDY RESULTS

- o PIPE RUPTURE PRESSURE UNCERTAINTY SENSITIVITY ON DHR-SHUTDOWN CDF
  - LOGARITHMIC STD DEV = 0.36 (BASE CASE)
  - LOGARITHMIC STD DEV = 0.1 (SENSITIVITY CASE)
- o HRA SENSITIVITY ON RISK
  - HRA METHOD (DETAILED ANALYSIS VS. SCREENING)
  - OPTIMIZED PSFs

SENSITIVITY OF PIPE RUPTURE PRESSURE UNCERTAINTY ON DHR-SD SEQUENCE

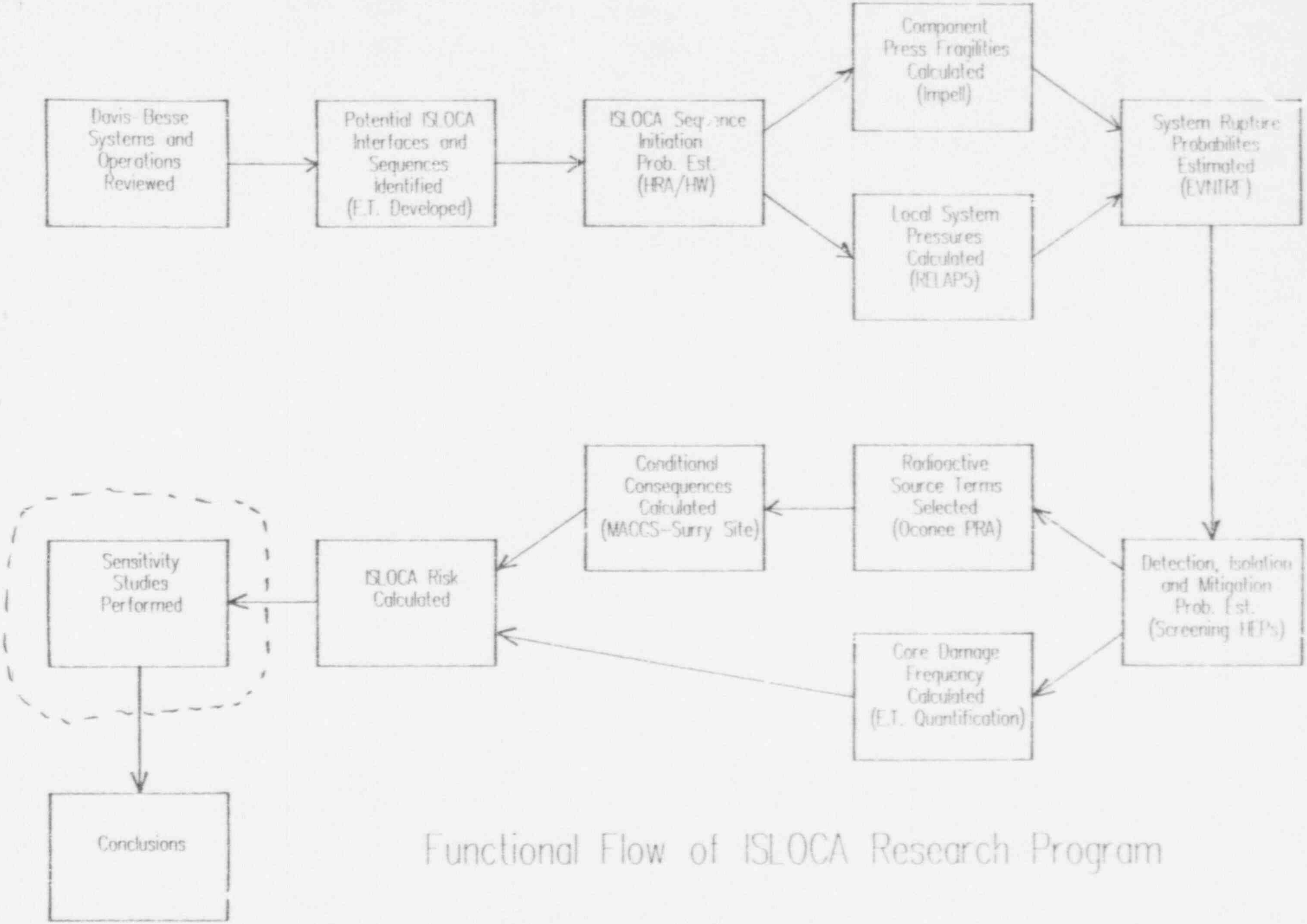
Sequence Class	Base Case	Sensitivity Case
OK-op	6.98E-5	7.11E-5
LK-ncd	6.73E-5	7.79E-5
REL-mit	1.15E-5	5.49E-6
REL-lg	1.15E-5	5.49E-6

SENSITIVITY OF HRA TECHNIQUE ON BOTH CDF AND RISK.

Risk Measure	Base Case	Optimum HEPs	Screening HEPs
REL-LG	1.9E-5	8.1E-9	1.4E-2
REL-MIT	1.9E-5	8.0E-7	1.4E-2
LK-NCD	1.9E-2	7.2E-3	3.3E-2
OK-OP	4.2E-2	4.2E-2	2.0E-1
Pop-Dose	71	0.8	53,000
Lat-Cancers	0.1	7.5E-4	75
Early-Fat	6.8E-7	3.4E-10	5.0E-4

o Note that the screening evaluation would not have identified the DHR-SD sequence, which involves a human error of commission.

This sequence is not included in the Screening HEP totals.



Functional Flow of ISLOCA Research Program

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EVENT TREE ANALYSIS SUPPORT

FEBRUARY 26/27, 1990

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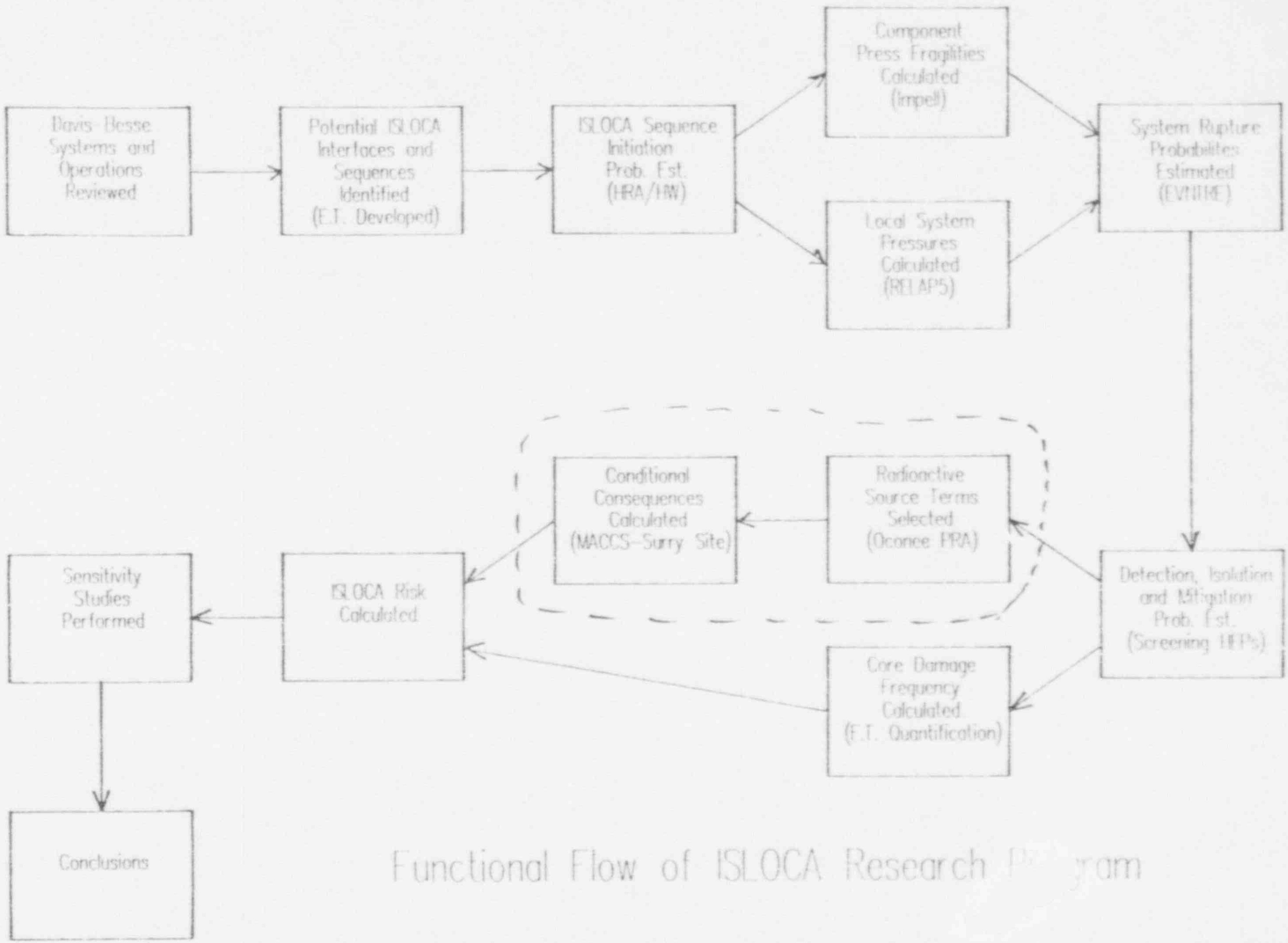
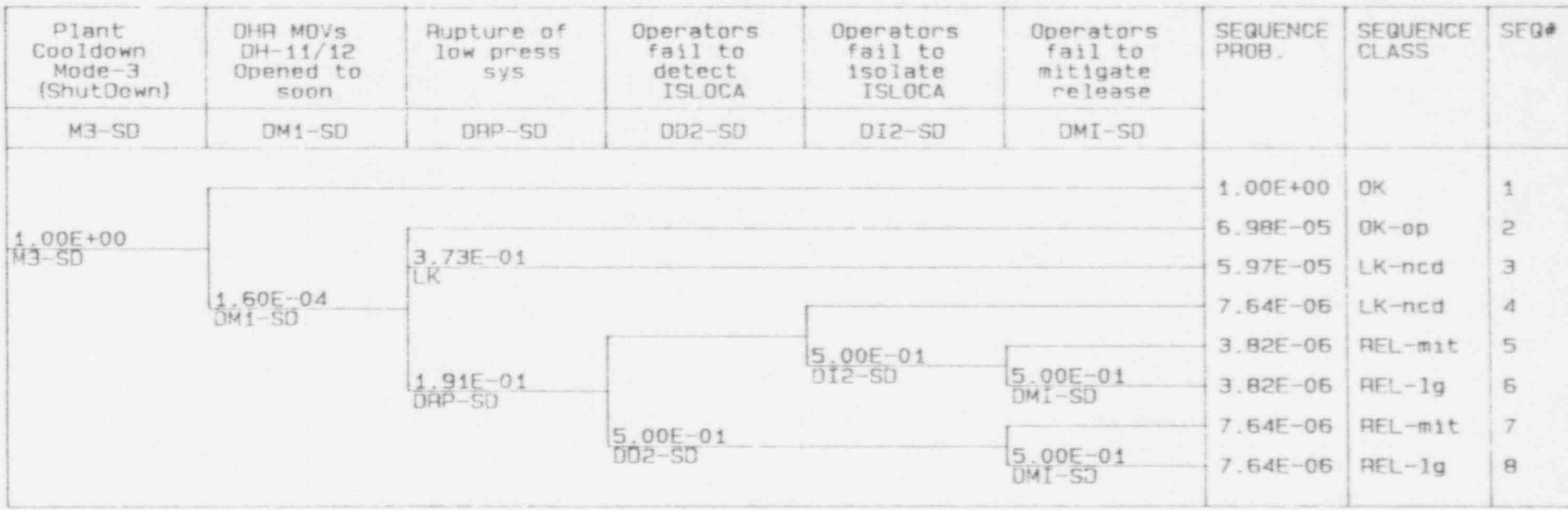
EG&G Idaho, Inc.



Functional Flow of ISLOCA Research Program



ISLOCA E.T. for Davis-Besse DHR Letdown (Shutdown) DB-DHR-D.TRE 2/11/90

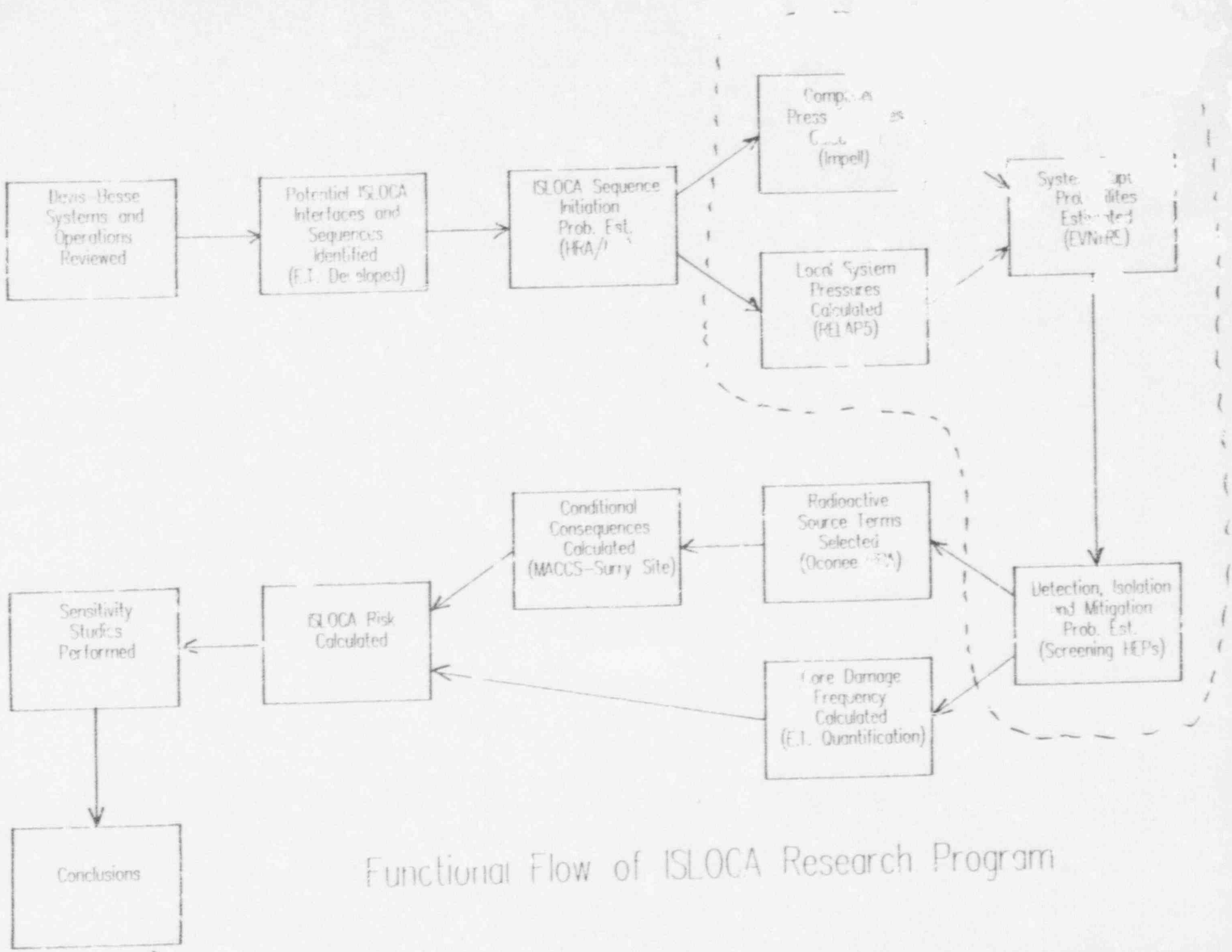
IDENTIFICATION OF ERRORS

- o LATENT ERRORS

- o ERRORS OF COMMISSION

  - ERRORS OF EXECUTION ARE GENERALLY MODELED AND HAVE HIGH ACCEPTANCE

  - ERRORS OF INTENTION ARE NOT GENERALLY MODELED AND HAVE LOWER ACCEPTANCE



Functional Flow of ISLOCA Research Program

REALISTIC COMPONENT PRESSURE FRAGILITIES WERE ESTIMATED

- o ANALYSIS PERFORMED BY IMPELL CORPORATION
  - STATE OF THE ART TECHNIQUES WERE UTILIZED
  - AVAILABLE TEST DATA REVIEW
  - ALL RELEVANT COMPONENTS EXAMINED, INCLUDING: PIPES, VALVES, FLANGES, VESSELS, TANKS, HEAT EXCHANGERS, ETC.
  - UNCERTAINTY IN MEDIAN VALUES ESTIMATED

MANY LOW PRESSURE RATED COMPONENTS NOT CAPABLE OF WITHSTANDING RCS PRESSURES

- o MEDIAN FAILURE PRESSURES (LARGE RUPTURE):

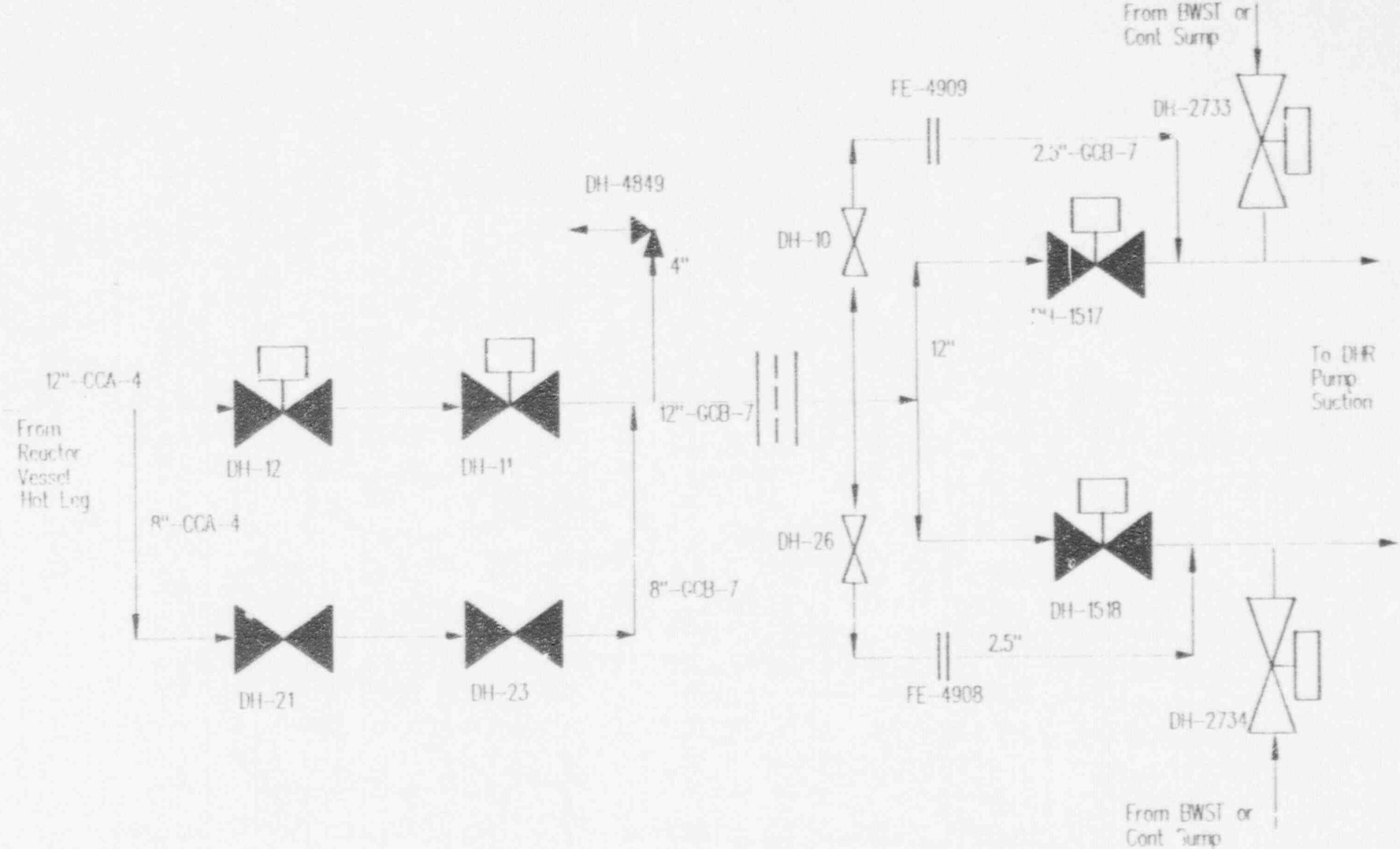
- 12" SCH20 PIPE	1660 PSIG
- 18" SCH10 PIPE	843 PSIG
- 12" 300# FLANGE	2250 PSIG

- o DHR HEAT EXCHANGER:

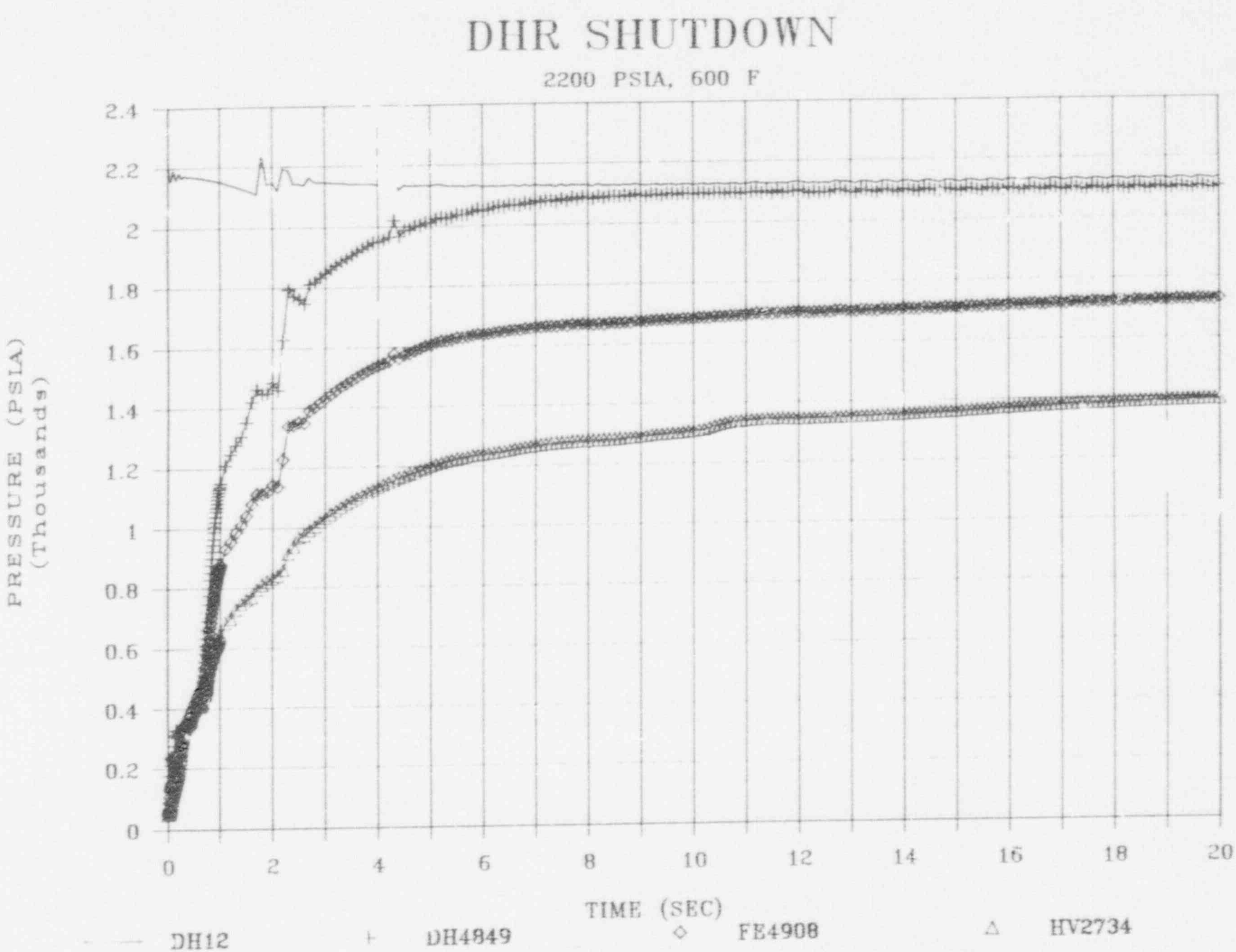
- TUBE SHEET FLANGE	893 PSIG
- PLASTIC COLLAPSE HEAD BUCKLING	1030 PSIG
- CYLINDER RUPTURE	1630 PSIG

DAVIS-BESSE DHR SYSTEM RUPTURE DATA (after screening)

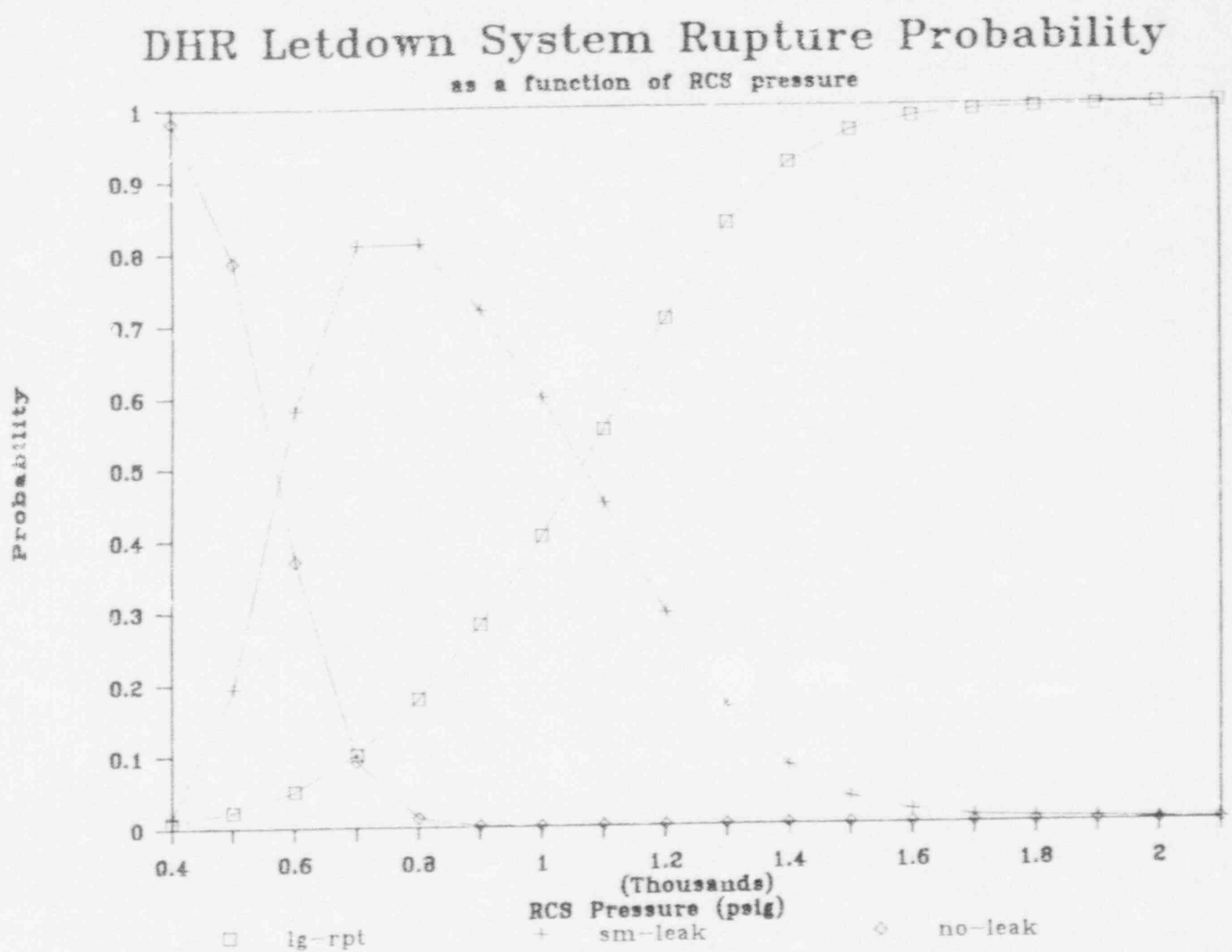
Component	Description	Median Failure Press	Log Std Dev	Leak Size at Failure	Large Leak Press
12"-GCB-7	Pipe sch20	1660	0.36	Lg	1660
DH-1517	12" MOV Gate 300#	1704	0.2	Sm	>2500
18"-GCB-8	Pipe sch20	1488	0.36	Lg	1488
DH-2733	18" MOV Gate 300#	2277	0.2	Sm	>2500
18"-HCB-1	Pipe sch10S	843	0.36	Lg	843
14"-HCB-1	Pipe sch10S	1090	0.36	Lg	1090
DH-81	14" Sw Check 150#	1445	0.2	Sm	>2500
12"-GCB-8	Pipe sch20	1660	0.36	Lg	1660
12GCB8a	Flange-a 300#	2250	0.12	Lg	2250
12GCB8b	Flange-b 300#	2250	0.12	Lg	2250
12GCB8c	Flange-c 300#	2250	0.12	Lg	2250
P42-1	DHR pump 1-1	2250	0.2	Sm	>2500
10"-GCB-1	Pipe sch20	1984	0.36	Lg	1984
10GCB1a	10" Flange-a 300#	2485	0.12	Lg	2485
DH-43	10" Sw Check 300#	2016	0.2	Sm	>2500
DH-45	10" HW Gate 300#	2170	0.2	Sm	>2500
E271T	DHR Hx Tube sh flg fail	432	0.12	Sm	893
E271P	DHR Hx Plastic col hd bk	1030	0.23	.2Lg	1030
E271C	DHR Hx Cylinder Rupt.	1630	0.27	Lg	1630
E271A	DHR Hx Asym. head bucklg	2030	0.23	.2Sm	n/a
E271a	10" out-flg E27-1 300#	2485	0.12	Lg	2485
E271b	10" in-flg E27-1 300#	2485	0.12	Lg	2485
6"-GCB-10	Pipe sch10S	1585	0.36	Lg	1585
10"-GCB-10	Pipe sch20	1984	0.36	Lg	1984
8"-GCB-10	Pipe sch20	2503	0.36	Lg	2503
DH-128	8" Sw Check 300#	1242	0.2	Sm	>2500
4"-GCB-2	Pipe sch10S	2075	0.36	Lg	2075
FE-DH2B	Flow El. 10" 300# flg.	2485	0.12	Lg	2485



From BWST or Cont Sump
FE-4909
DH-2733
2.5"-GCB-7
DH-4849
DH-10
4"
H-1517
12"
To DHR Pump Suction
12"-CCA-4
From Reactor Vessel Hot Leg
DH-12
DH-1
12"-GCB-7
8"-CCA-4
DH-26
8"-GCB-7
DH-1518
2.5"
DH-21
DH-23
FE-4908
DH-2734
From BWST or Cont Sump



DHR SHUTDOWN
2200 PSIA, 600 F
PRESSURE (PSIA)
(Thousands)
2.4
2.2
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
2
4
6
8
10
12
14
16
18
20
TIME (SEC)
DH12
DH4849
FE4908
HV2734


DHR Letdown System Rupture Probability
as a function of RCS pressure
Probability
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
(Thousands)
RCS Pressure (psig)
lg-rpt
sm-leak
no-leak

COMPONENT FAILURE PROBABILITIES CAN BE CALCULATED UTILIZING SEISMIC FAILURE EQUATION

- REF: R.P.KENNEDY ET AL, NUCLEAR ENGINEERING AND DESIGN, VOL 59, NO.2, AUGUST 1980.
- PROBABILITY OF FAILURE AT 2100 PSIG FOR A 12-INCH SCH20 PIPE (MED. = 1660 PSIG, LOGARITHMIC STD DEV = 0.36):
- PROB(FAIL PRESS < 2100 PSIG) = PHI((LN(2100)-LN(1660))/0.36)

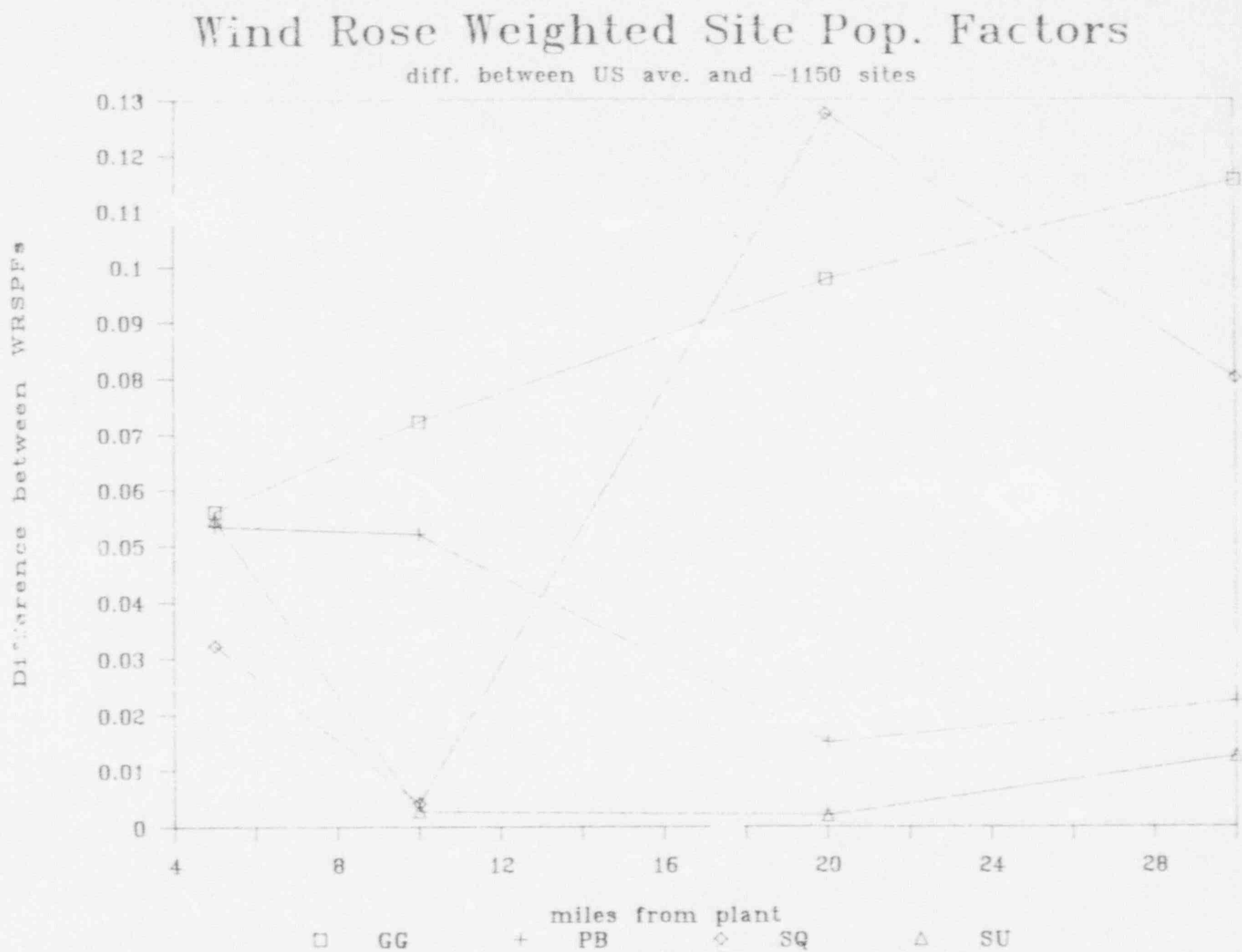
  = PHI(0.65)

  = 0.742

DIFFERENCE BETWEEN NUREG-1150 SITES AND U.S. AVERAGE SITE

	Site Population Factors			
	5mi.	10mi.	20mi.	30mi
Grand Gulf	0.065	0.069	0.091	0.110
Peach Bottom	0.056	0.043	0.007	0.021
Sequoyah	0.002	0.004	0.054	0.012
Surry	0.065	0.012	0.016	0.007
Zion	0.637	0.618	0.471	0.423

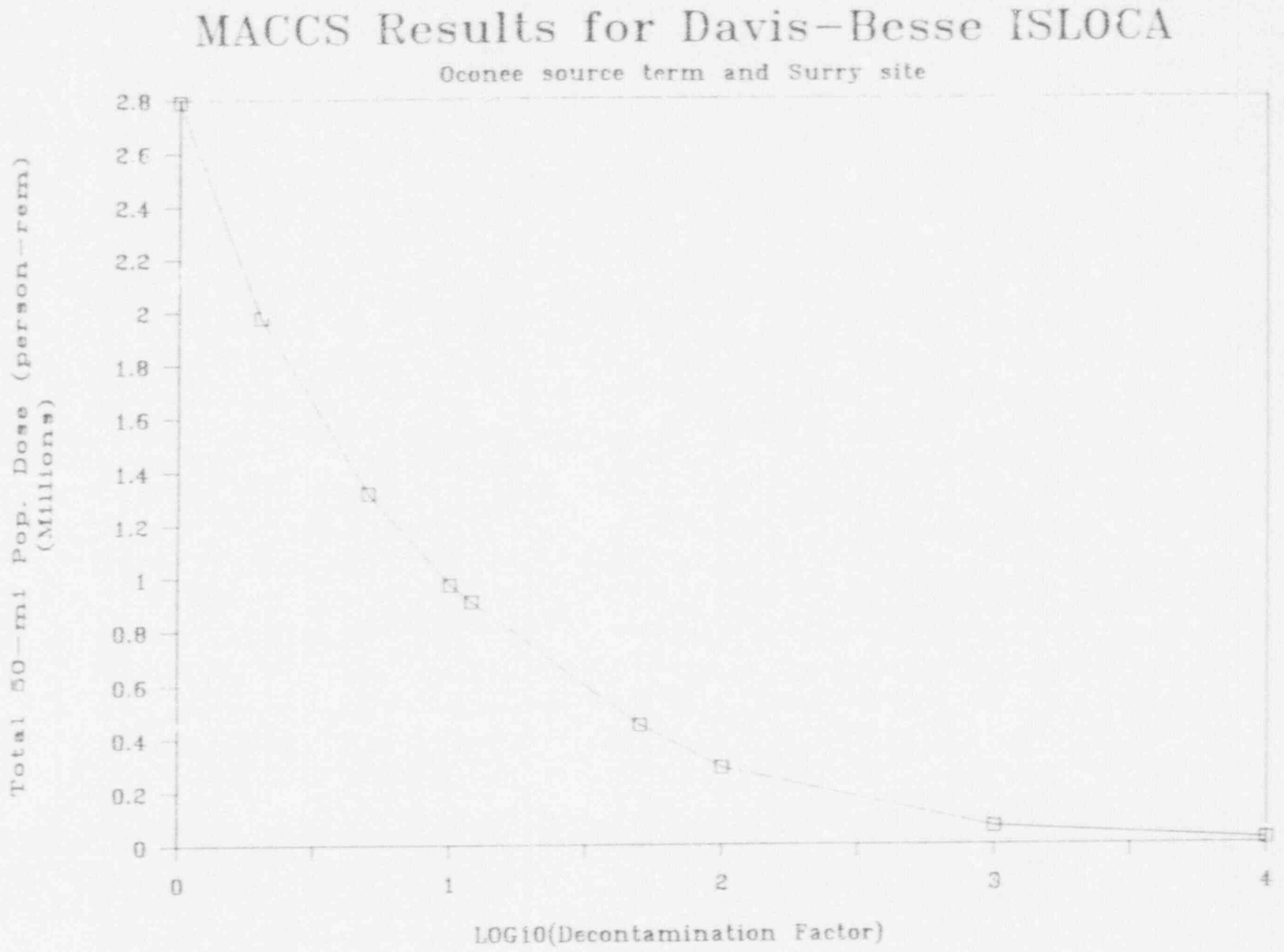
	Wind Rose Weighted Site Population Factors			
	5mi.	10mi.	20mi.	30mi
Grand Gulf	0.056	0.072	0.098	0.115
Peach Bottom	0.053	0.052	0.015	0.022
Sequoyah	0.032	0.004	0.127	0.080
Surry	0.055	0.003	0.002	0.013
Zion	0.809	0.745	0.567	0.517



Wind Rose Weighted Site Pop. Factors
diff. between US ave. and -1150 sites
Di'ference between WRSPFs
0.13
0.12
0.11
0.1
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
4
8
12
16
20
24
28
miles from plant
GG
PB
SQ
SU

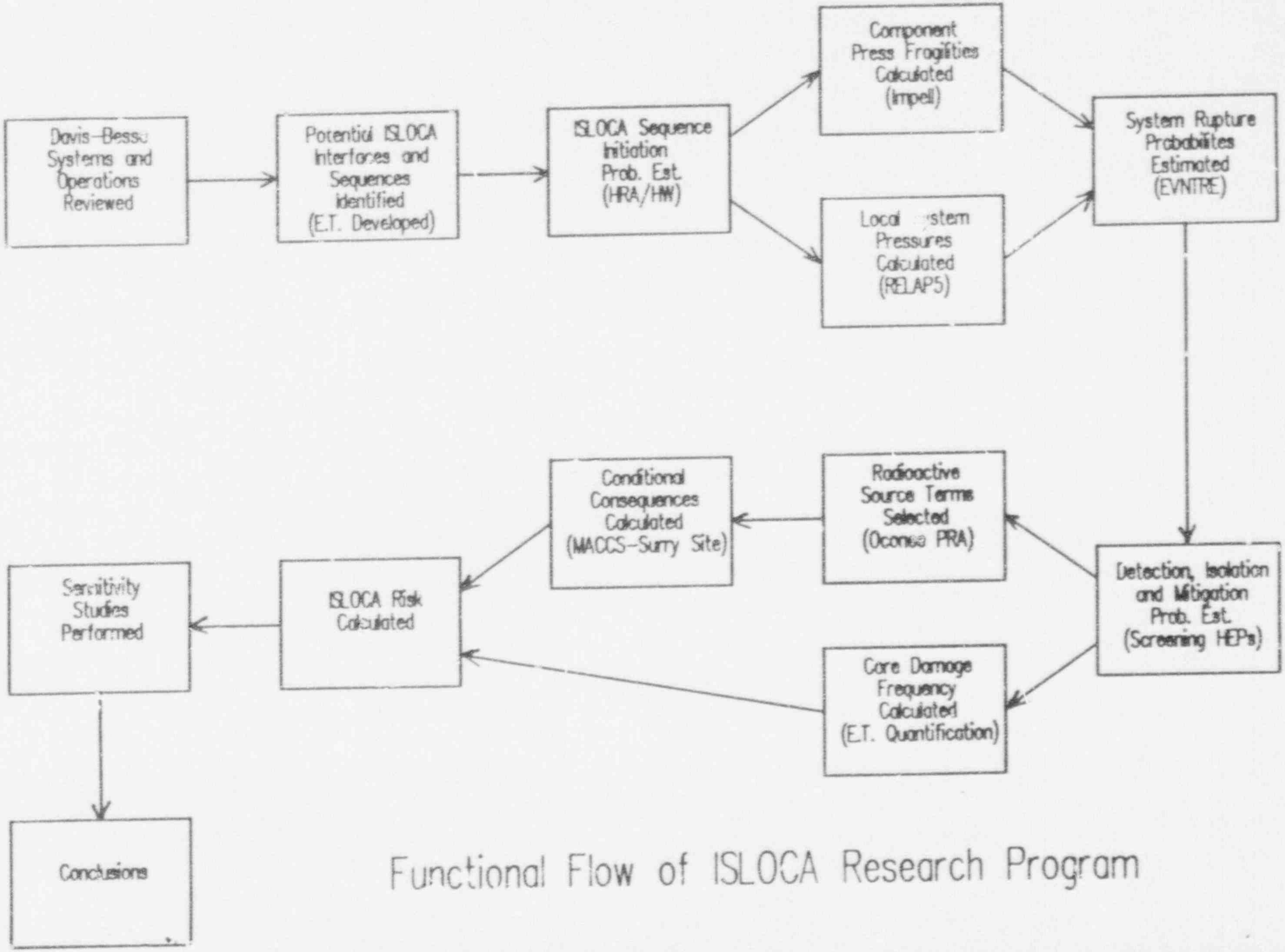
CONDITIONAL CONSEQUENCES CALCULATED USING MACCS

- Second draft NUREG-1150 utilized:
  - MACCS-pc version 1.5.11
  - Surry evacuation strategy
  - Decontamination Factors:
    - Large Release DF=1
    - Mitigated Rel DF=10
  - Consequence Measures:
    - Early Fatalities
    - Latent Cancers (total grid)
    - Population Dose (50-mi.)


MACCS Results for Davis-Besse ISLOCA
Oconee source term and Surry site
Total 50-mi Pop. Dose (person-rem)
(Millions)
2.8
2.6
2.4
2.2
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
1
2
3
4
LOG10(Decontamination Factor)

MACCS CONSEQUENCE RESULTS FOR A RANGE OF POSSIBLE DFs
(Oconee source term, scaled to Davis-Besse power, and the Surry site)

Consequence Measure	DF=1	DF=5	DF=10	DF=100
Population Dose (person-rem, 50-mi.)	2.8E+6	1.3E+6	9.7E+5	2.9E+5
Latent Cancers (total grid)	4.5E+3	1.5E+3	8.9E+2	1.4E+2
Early Fatalities	3.6E-2	3.0E-4	5.8E-5	1.2E-6



Functional Flow of ISLOCA Research Program

AN INTEGRATED APPROACH TO HRA WAS USED

- o ENSURE ALL TYPES OF ACTIONS ARE REPRESENTED
- o IDENTIFY AND SCREEN HUMAN INTERACTIONS WHICH MAY BE RISK SIGNIFICANT
- o DEVELOP A DETAILED DESCRIPTION OF IMPORTANT HUMAN ACTIONS
- o SELECT AND APPLY APPROPRIATE MODELING TECHNIQUES
- o EVALUATE THE IMPACT OF SIGNIFICANT HUMAN ACTIONS
- o QUANTIFY THE PROBABILITIES FOR THE VARIOUS HUMAN ACTIONS
- o DOCUMENT THE INFORMATION FOR TRACEABILITY

ASSESSMENT OF THE POTENTIAL FOR ISLOCA AT DAVIS-BESSE -

Idaho National Engineering Laboratory

EG&G Idaho, Inc.

ISLOCA PROGRAM OBJECTIVES

- DEVELOP A FRAMEWORK FOR EVALUATING ISLOCA RISK
  - HARDWARE
  - HUMAN ERROR
  - POTENTIAL FOR RISK REDUCTION

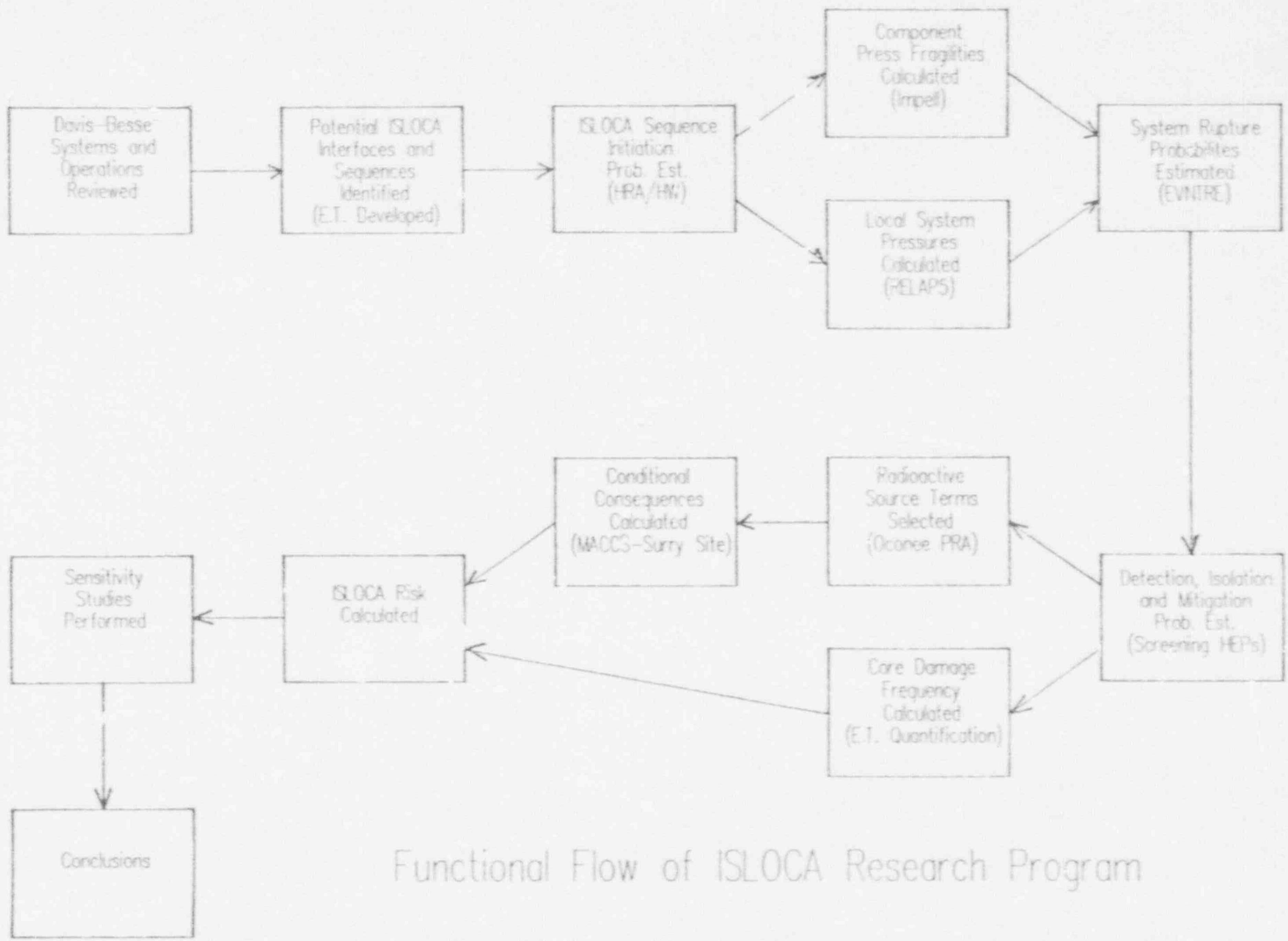
- DETERMINE THE EFFECTS OF HUMAN ERROR AND THEIR CONTRIBUTORS

- EVALUATE THE FRAGILITY OF LOW PRESSURE SYSTEMS
  - FAILURE LOCATIONS
  - FAILURE PROBABILITIES

ISLOCA PROGRAM OBJECTIVES (CONT.)

- Identify and describe potential ISLOCA sequences
  - Timing
  - Accident management strategies
  - Effect on other equipment

- Estimate consequences of potential ISLOCA sequences
  - Core melt frequency
  - Offsite consequences
  - Recommend consequence reduction actions



Functional Flow of ISLOCA Research Program

IMPORTANT CONSIDERATIONS FOR FOLLOWING PRESENTATIONS

- EFFECT OF HUMAN ACTIONS AS INITIATORS FOR AN ISLOCA
- RELATIVE EFFECT OF HUMAN ERRORS AND HARDWARE FAILURES AS CONTRIBUTORS TO ISLOCA CORE MELT FREQUENCY AND RISK
- COMPONENTS THAT WOULD FAIL WHEN EXPOSED TO OVERPRESSURE
- INFLUENCE OF PROCEDURES, TRAINING, AND INSTRUMENTATION ON THE CAPABILITIES OF PLANT PERSONNEL

LER REVIEWS AND DETAILED PLANT EXAMINATIONS IDENTIFIED LIKELY ISLOCA SEQUENCES

- LERs WERE USED TO IDENTIFY POSSIBLE "TYPES" OF HUMAN ERRORS AND FAILURES.
- LERs WERE NOT USED TO GENERATE FAILURE RATES, BECAUSE OF DIFFERENCES IN CONTEXT, SITUATION SPECIFICS, AND EXPOSURE.
- DETAILED PLANT REVIEW IDENTIFIED OPPORTUNITIES FOR POSSIBLE ISLOCA SEQUENCES.

ISLOCA SEQUENCES INITIATED BY MULTIPLE HUMAN ERRORS OR COMBINATIONS OF HUMAN ERRORS AND HARDWARE FAULTS

- HISTORICAL EXPERIENCE INDICATES:
  - IMPROPER VALVE LINEUP AND OPERATOR ERRORS IN

    MISPOSITIONING VALVES - RELATIVELY LIKELY.
  - RANDOM AND CATASTROPHIC FAILURES OF REDUNDANT VALVES

    IN STANDBY - NOT SUPPORTED.

REVIEW OF D-B SYSTEMS AND OPERATIONS LEADS TO IDENTIFICATION OF ISLOCA INTERFACES AND SEQUENCES

- 1-inch and smaller lines, and <200 gpm deemed risk insignificant.
- Three ISLOCA interfaces identified: HPI, LPI, and DHR letdown.
- Five possible ISLOCA sequences identified:
  - HPI
  - MU&P/HPI
  - LPI
  - DHR-Startup
  - DHR-Shutdown

AN INTEGRATED APPROACH TO HRA WAS USED

- Ensure all types of actions are represented
- Identify and screen human interactions which may be risk significant
- Develop a detailed description of important human actions
- Select and apply appropriate modeling techniques
- Evaluate the impact of significant human actions
- Quantify the probabilities for the various human actions
- Document the information for traceability

Ashok → Screening base approach
Does not provide
Issue awareness

IDENTIFICATION OF ERRORS

- LATENT ERRORS

- ERRORS OF COMMISSION

  - ERRORS OF EXECUTION ARE GENERALLY MODELED AND HAVE HIGH ACCEPTANCE

  - ERRORS OF INTENTION ARE NOT GENERALLY MODELED AND HAVE LOWER ACCEPTANCE

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

HRA FINDINGS AND CONCLUSIONS

- A COMBINATION OF HRA MODELING AND QUANTIFICATION TECHNIQUES WERE USED TO REPRESENT THE WIDE VARIETY OF HUMAN ACTIONS IDENTIFIED

- ERRORS OF COMMISSION AND LATENT ERRORS PROVED TO BE RISK SIGNIFICANT

- SCREENING VALUES WERE USED FOR DETECTION, ISOLATION, AND MITIGATION BUT ADDITIONAL ANALYSES ARE UNDER CONSIDERATION

REALISTIC COMPONENT PRESSURE FRAGILITIES WERE ESTIMATED

- ANALYSIS PERFORMED BY IMPELL CORPORATION
  - STATE OF THE ART TECHNIQUES WERE UTILIZED
  - AVAILABLE TEST DATA REVIEW
  - ALL RELEVANT COMPONENTS EXAMINED, INCLUDING: PIPES, VALVES, FLANGES, VESSELS, TANKS, HEAT EXCHANGERS, ETC.
  - UNCERTAINTY IN MEDIAN VALUES ESTIMATED

MANY LOW PRESSURE RATED COMPONENTS NOT CAPABLE OF WITHSTANDING RCS PRESSURES

- MEDIAN FAILURE PRESSURES (LARGE RUPTURE):

- 12" SCH20 PIPE	1660 PSIG
- 18" SCH10 PIPE	843 PSIG
- 12" 300# FLANGE	2250 PSIG

- DHR HEAT EXCHANGER:

- TUBE SHEET FLANGE	893 PSIG
- PLASTIC COLLAPSE HEAD BUCKLING	1030 PSIG
- CYLINDER RUPTURE	1630 PSIG

LOCAL INTERFACING SYSTEM PRESSURES PREDICTED USING SYSTEM SPECIFIC MODELS

- RELAP5 Model Built and Run.
  - Pressure equilibrium established very quickly - dead ended (closed) systems pressurize virtually instantaneously
  - Small relief valves in combination with flow restrictions may protect portions of system

SYSTEM RUPTURE PROBABILITIES CALCULATED USING MONTE CARLO SIMULATION

- EVNTRE Computer Code (NUREG-1150) used to perform Simulation
  - Local system pressure sampled from postulated distribution e.g. Normally distributed with mean 2100 psi, std dev 50 psi
  - Component failure pressure sampled from postulated dist. e.g. 12-inch sch20 pipe, lognormally dist. with median 1660 psig, log std dev 0.36.

SYSTEM RUPTURE PROBABILITIES CALCULATED USING MONTE CARLO SIMULATION (CONTINUED)

- TWO SAMPLES COMPARED:
  - SYS PRESS > FAIL PRESS, THEN COMPONENT FAILED
  - SYS PRESS < FAIL PRESS, THEN NO FAILURE
- RUPTURE PROBABILITY IS FRACTION OF MONTE CARLO OBSERVATIONS RESULTING IN RUPTURES

SOURCE TERMS AND SITE DATA ESTIMATED UTILIZING EXISTING INFORMATION

- INFORMATION ON B&W PLANTS IS LIMITED, SOURCE TERM AND RELEASE TIMING TAKEN FROM OCONEE PRA (NSAC/60)
- INDUSTRY-WIDE AVERAGE SITE POPULATION ESTIMATED USING SANDIA SITING STUDY (NUREG/CR-2239)
- NUREG-1150 SITES COMPARED TO AVERAGE POPULATION, SURRY SELECTED AS REPRESENTING AVERAGE SITE (FOR MACCS INPUT)

CONDITIONAL CONSEQUENCES CALCULATED USING MACCS

- Second draft NUREG-1150 utilized:
  - MACCS-pc version 1.5.11
  - Surry evacuation strategy
  - Decontamination Factors:
    Large Release DF=1
    Mitigated Rel DF=10
  - Consequence Measures:
    Early Fatalities
    Latent Cancers (total grid)
    Population Dose (50-mi.)

MACCS CONSEQUENCE RESULTS FOR A RANGE OF POSSIBLE DFs
(OCONEE SOURCE TERM, SCALED TO DAVIS-BESSE POWER, AND THE SURRY SITE)

CONSEQUENCE MEASURE	DF=1	DF=5	DF=10	DF=100
POPULATION DOSE (PERSON-REM, 50-MI.)	2.8E+6	1.3E+6	9.7E+5	2.9E+5
LATENT CANCERS (TOTAL GRID)	4.5E+3	1.5E+3	8.9E+2	1.4E+2
EARLY FATALITIES	3.6E-2	3.0E-4	5.8E-5	1.2E-6

DAVIS-BESSE PLANT DAMAGE STATE FREQUENCY FROM ISLOCA SEQUENCES

(FREQUENCY PER REACTOR-YEAR)

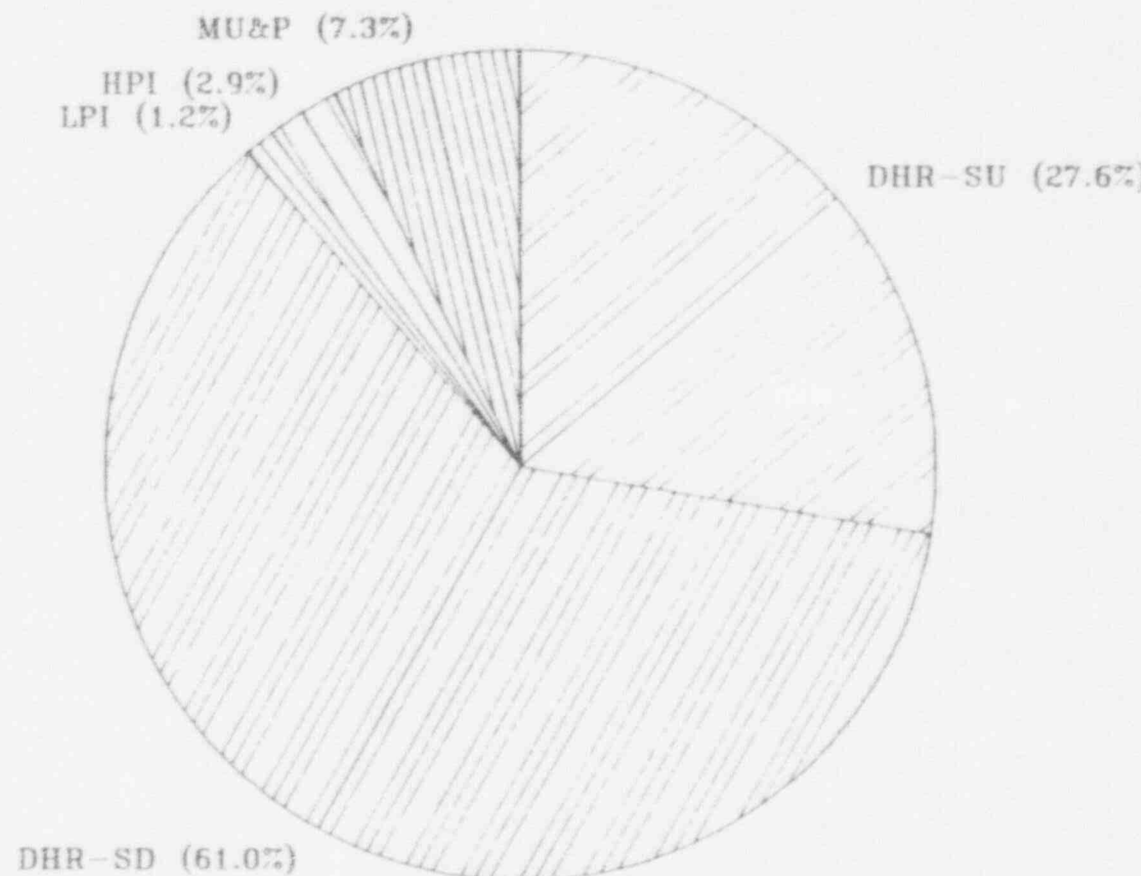
SEQUENCE	LG REL	MIT REL	LK-NCD	OK-OP
MU&P	1.4E-6	1.4E-6	1.1E-3	4.0E-2
HPI	5.4E-7	5.4E-7	2.3E-6	1.5E-3
LPI	2.2E-7	2.3E-7	2.0E-7	5.4E-7
DHR-SD	1.2E-5	1.2E-5	6.7E-5	7.0E-5
DHR-SU	5.2E-6	5.2E-6	1.8E-2	5.3E-5
TOTAL	1.9E-5	1.9E-5	1.9E-2	4.2E-2

← Plant Damage STATES

TOTAL CORE DAMAGE FREQUENCY: 3.8E-5/RX-YR. (SUM OF LARGE AND MITIGATED RELEASE FREQUENCIES).

Davis-Besse CDF Distribution

from ISLOCA Sequences



ISLOCA RISK FOR DAVIS-BESSE

(OCONEE SOURCE TERM, SCALED TO DAVIS-BESSE POWER, AND THE SURRY SITE)

RISK MEASURE	REL-LG DF=1	REL-MIT DF=10	TOTAL
POPULATION DOSE (PERSON-REM, 50-MI.)	53	18	71
LATENT CANCERS (TOTAL GRID)	8.4E-2	1.7E-2	1.0E-1
EARLY FATALITIES	6.7E-7	1.1E-9	6.8E-7

SENSITIVITY STUDY RESULTS

- PIPE RUPTURE PRESSURE UNCERTAINTY SENSITIVITY ON DHR-SHUTDOWN CDF
  - LOGARITHMIC STD DEV = 0.36 (BASE CASE)
  - LOGARITHMIC STD DEV = 0.1 (SENSITIVITY CASE)
- HRA SENSITIVITY ON RISK
  - HRA METHOD (DETAILED ANALYSIS VS. SCREENING)
  - OPTIMIZED PSFs

SENSITIVITY OF PIPE RUPTURE PRESSURE UNCERTAINTY ON DHR-SD SEQUENCE

Sequence Class	Base Case	Sensitivity Case
OK-op	6.98E-5	7.11E-5
LK-ncd	6.73E-5	7.79E-5
REL-mit	1.15E-5	5.49E-6
REL-lg	1.15E-5	5.49E-6

SENSITIVITY OF HRA TECHNIQUE ON BOTH CDF AND RISK.

Risk Measure	Base Case	Optimum HEPs	Screening HEPs
REL-LG	1.9E-5	8.1E-9	1.4E-2
REL-MIT	1.9E-5	8.0E-7	1.4E-2
LK-NCD	1.9E-2	7.2E-3	3.3E-2
OK-OP	4.2E-2	4.2E-2	2.0E-1
Pop-Dose	71	0.8	53,000
Lat-Cancers	0.1	7.5E-4	75
Early-Fat	6.8E-7	3.4E-10	5.0E-4

o Note that the screening evaluation would not have identified the DHR-SD sequence, which involves a human error of commission.

This sequence is not included in the Screening HEP totals.

- Comparison of Base Case HRA Values to Optimized HRA Values

This analysis was conducted to determine if modifications in the Human Machine interface would result in significant gains in Operator Performance.

MODIFICATIONS TO THE HUMAN MACHINE SYSTEM

- PROCEDURES
  - CAUTIONS, NOTES, AND WARNINGS ADDED
  - CREATE PROCEDURE FOR ISLOCA

- INSTRUMENTATION
  - ADDITION OF VALVE STATUS BOARD
  - PRESENTATION OF INFORMATION ON PRESSURES, TEMPERATURES, LEVEL, AND FLOW

- TRAINING
  - FORMAL TRAINING ON ISLOCA, ASSOCIATED ALARMS, NEW PROCEDURES

- RECOVERY
  - ALL TASKS COVERED BY PROCEDURES, CHECKOFFS, AND SECOND CHECKERS

Conclusions

$\frac{4\ -5}{8\ -7} \rightarrow 0.5\ km^2$

$\rightarrow 100$

- Modifications in procedures, instrumentation, training, and recovery result in a significant reduction of ISLOCA Core Damage Frequency (from 3.9×10^{-5} to 8.1×10^{-7}) and risk

- Making plant personnel aware of ISLOCA through these modifications will eliminate ISLOCA as a significant contributor to risk

CONCLUSIONS FOR ISLOCA EVALUATION OF DAVIS-BESSE

- HUMAN ERRORS DURING STARTUP AND SHUTDOWN WERE SIGNIFICANT CONTRIBUTORS TO CORE MELT FREQUENCY AND RISK

- HARDWARE FAILURES WERE RELATIVELY SMALL CONTRIBUTORS TO CORE MELT FREQUENCY AND RISK

- ALTHOUGH HARDWARE WOULD BE AVAILABLE TO ISOLATE ISLOCA BREAKS, ADEQUATE PROCEDURES AND TRAINING ARE NOT AVAILABLE TO ENSURE THIS HARDWARE IS USED

CONCLUSIONS FOR ISLOCA EVALUATION OF DAVIS-BESSE (CONT.)

- RELATIVELY SIMPLE CHANGES TO PROCEDURES, TRAINING, AND INSTRUMENTATION WOULD SIGNIFICANTLY REDUCE PLANT RISK

- DAMAGE BY FLOODING OR SPRAYING OF ADJACENT EQUIPMENT IS NOT RISK SIGNIFICANT OWING TO ADEQUATE EQUIPMENT SEPARATION

- HEAT EXCHANGERS AND LARGE DIAMETER, LOW PRESSURE PIPING WOULD MOST LIKELY RUPTURE

CONCLUSIONS FOR ISLOCA EVALUATION OF DAVIS-BESSE (CONT.)

- Relatively simple changes to procedures, training, and instrumentation would significantly reduce plant risk
- Damage by flooding or spraying of adjacent equipment is not risk significant owing to adequate equipment separation
- Heat exchangers and large diameter, low pressure piping would most likely rupture

GENERIC CONCLUSIONS

- ISLOCA ANALYSES TYPICALLY FOUND IN PRAs ARE LIKELY INCOMPLETE DESCRIPTIONS OF ISLOCA RISK
- HUMAN RELIABILITY ISSUES (INCLUDING ERRORS OF COMMISSION) ARE POTENTIALLY DOMINANT CONTRIBUTORS TO ISLOCA RISK

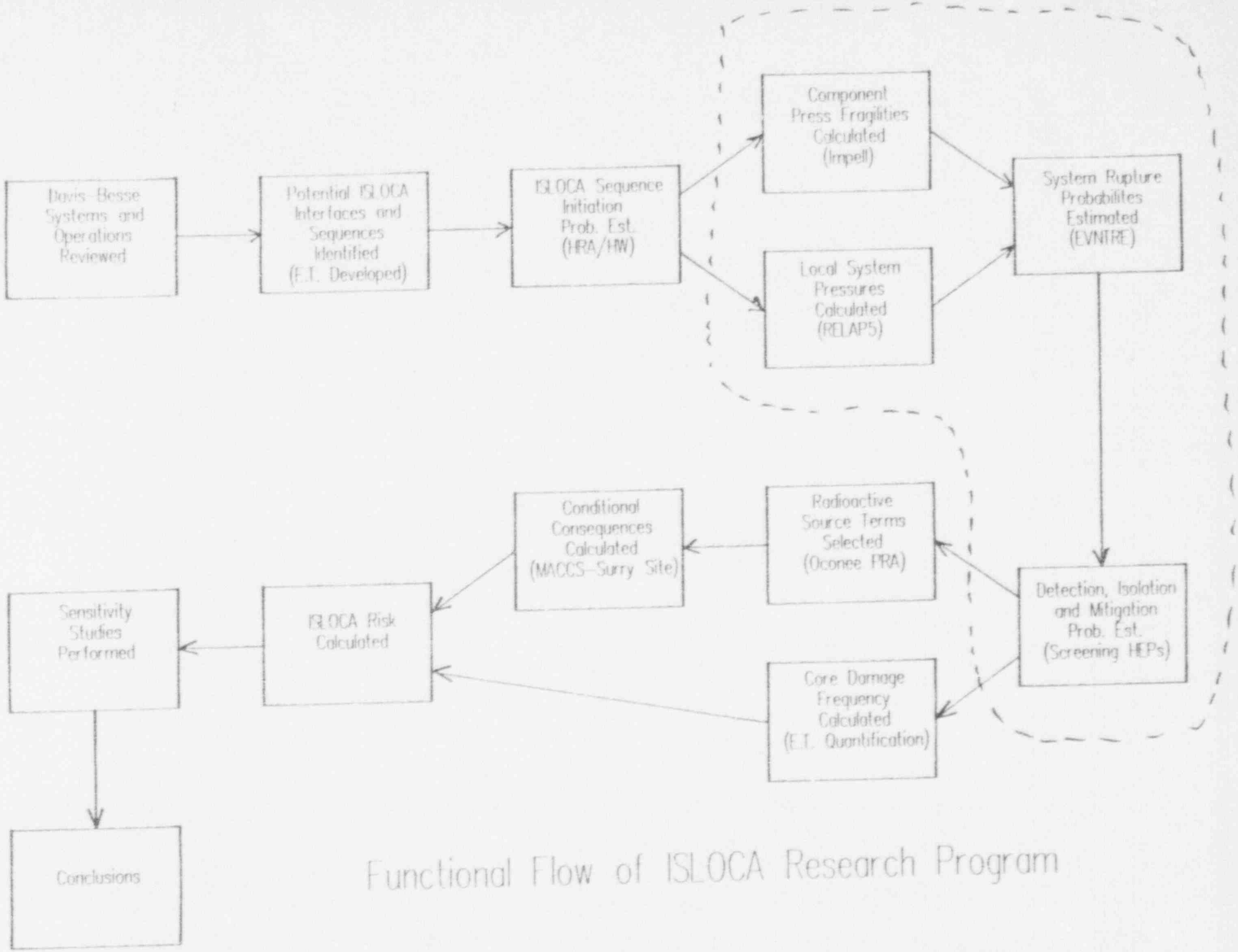
Idaho
National
Engineering
Laboratory

EVENT TREE ANALYSIS SUPPORT

FEBRUARY 26/27, 1990

W. J. GALYEAN

EG&G Idaho, Inc.



Functional Flow of ISLOCA Research Program

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- 18" SCH10 PIPE	843 PSIG
- 12" 300# FLANGE	2250 PSIG

- DHR HEAT EXCHANGER:

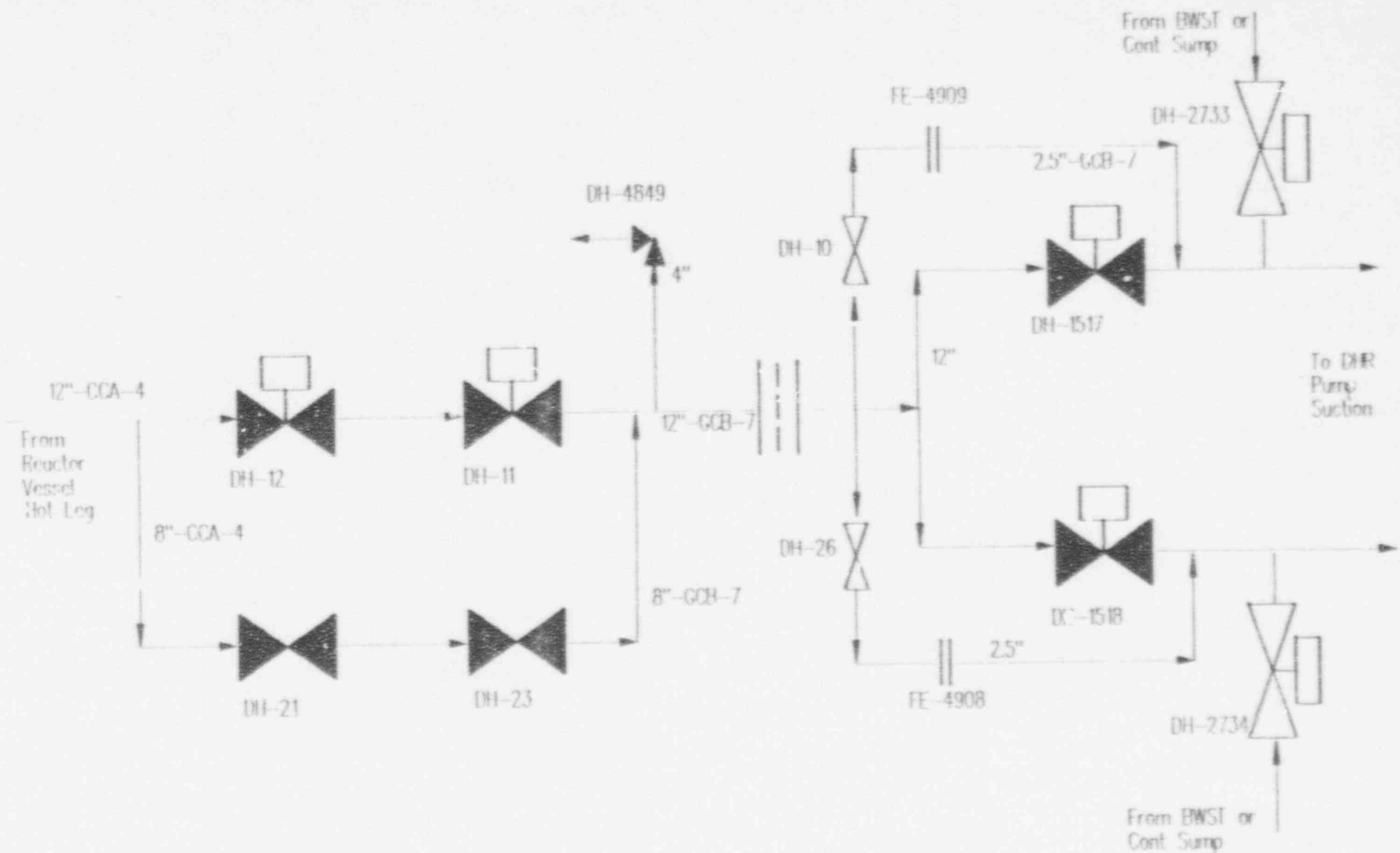
- TUBE SHEET FLANGE	893 PSIG
- PLASTIC COLLAPSE HEAD BUCKLING	1030 PSIG
- CYLINDER RUPTURE	1630 PSIG

DAVIS-BESSE DHR SYSTEM RUPTURE DATA (after screening)

Component	Description	Median Failure Press	Log Std Dev	Leak Size at Failure	Large Leak Press
12"-GCB-7	Pipe sch20	1660	0.36	Lg	1660
DH-1517	12" MOV Gate 300#	1704	0.2	Sm	>2500
18"-GCB-8	Pipe sch20	1488	0.36	Lg	1488
DH-2733	18" MOV Gate 300#	2277	0.2	Sm	>2500
18"-HCB-1	Pipe sch10S	843	0.36	Lg	843
14"-HCB-1	Pipe sch10S	1090	0.36	Lg	1090
DH-81	14" Sw Check 150#	1445	0.2	Sm	>2500
12"-GCB-8	Pipe sch20	1660	0.36	Lg	1660
12GCB8a	Flange-a 300#	2250	0.12	Lg	2250
12GCB8b	Flange-b 300#	2250	0.12	Lg	2250
12GCB8c	Flange-c 300#	2250	0.12	Lg	2250
P42-1	DHR pump 1-1	2250	0.2	Sm	>2500
10"-GCB-1	Pipe sch20	1984	0.36	Lg	1984
10GCB1a	10" Flange-a 300#	2485	0.12	Lg	2485
DH-43	10" Sw Check 300#	2016	0.2	Sm	>2500
DH-45	10" HW Gate 300#	2170	0.2	Sm	>2500
E271T	DHR Hx Tube sh flg fail	432	0.12	Sm	893
E271P	DHR Hx Plastic col hd bk	1030	0.23	.2Lg	1030
E271C	DHR Hx Cylinder Rupt.	1630	0.27	Lg	1630
E271A	DHR Hx Asym. head bucklg	2030	0.23	.2Sm	n/a
E271a	10" out-flg E27-1 300#	2485	0.12	Lg	2485
E271b	10" in-flg E27-1 300#	2485	0.12	Lg	2485
6"-GCB-10	Pipe sch10S	1585	0.36	Lg	1585
10"-GCB-10	Pipe sch20	1984	0.36	Lg	1984
8"-GCB-10	Pipe sch20	2503	0.36	Lg	2503
DH-128	8" Sw Check 300#	1242	0.2	Sm	>2500
4"-GCB-2	Pipe sch10S	2075	0.36	Lg	2075
FE-DH2B	Flow El. 10" 300# flg.	2485	0.12	Lg	2485

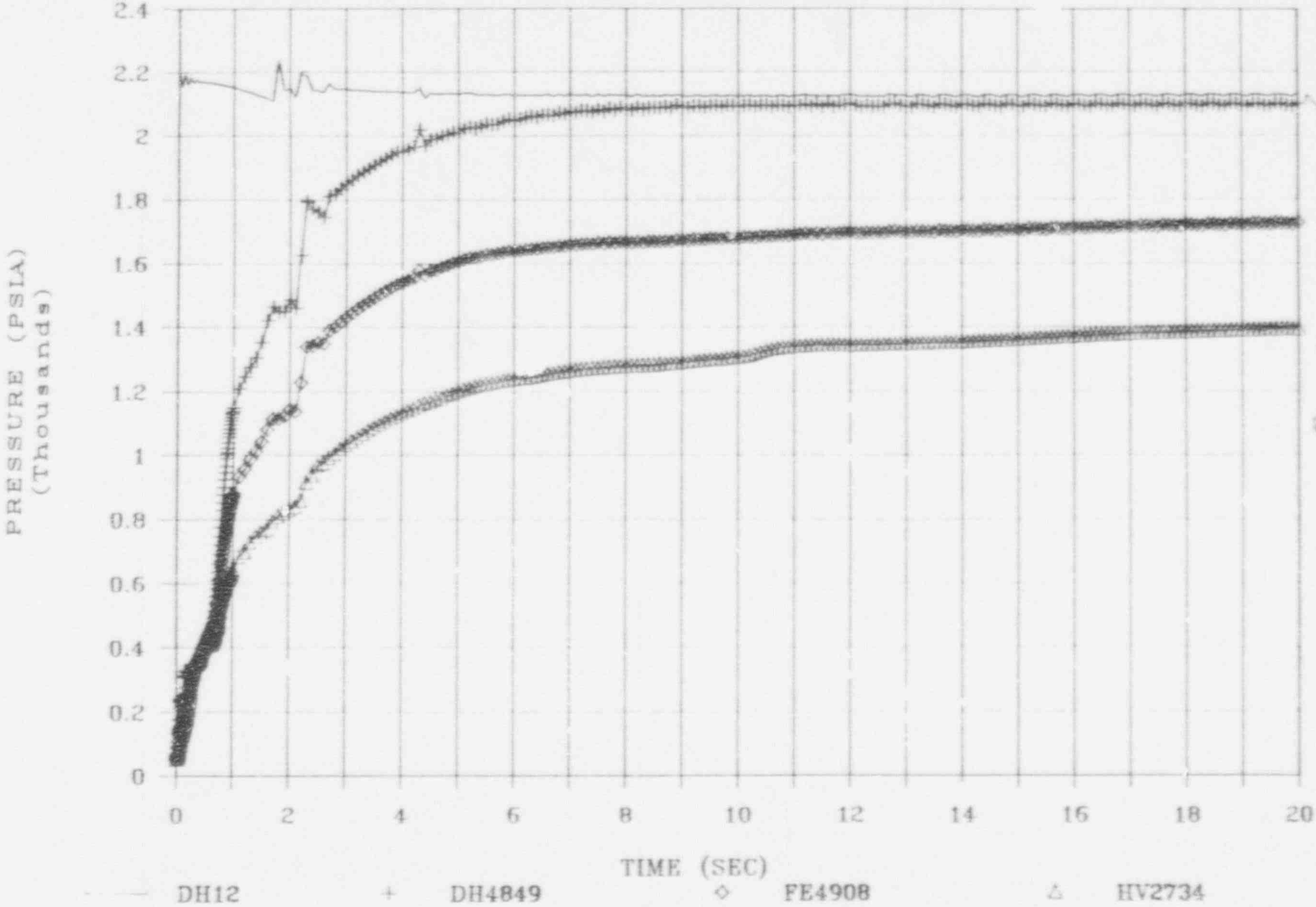
LOCAL INTERFACING SYSTEM PRESSURES ARE DIFFICULT TO PREDICT

- RELAP5 Model Built and Run.
  - Pressure equilibrium established very quickly - Dead ended (closed) systems pressurize virtually instantaneously
  - Small relief valves in combination with flow restrictions may protect portions of system


From BWST or
Cont Sump
FE-4909
DH-2733
2.5"-GCB-7
DH-4849
DH-10
4"
DH-1517
12"
To DHR
Pump
Suction
12"-CCA-4
From
Reactor
Vessel
Hot Leg
DH-12
DH-11
12"-GCB-7
8"-CCA-4
DH-26
8"-GCB-7
DC-1518
2.5"
DH-21
DH-23
FE-4908
DH-2734
From BWST or
Cont Sump

DHR SHUTDOWN

2200 PSIA, 600 F

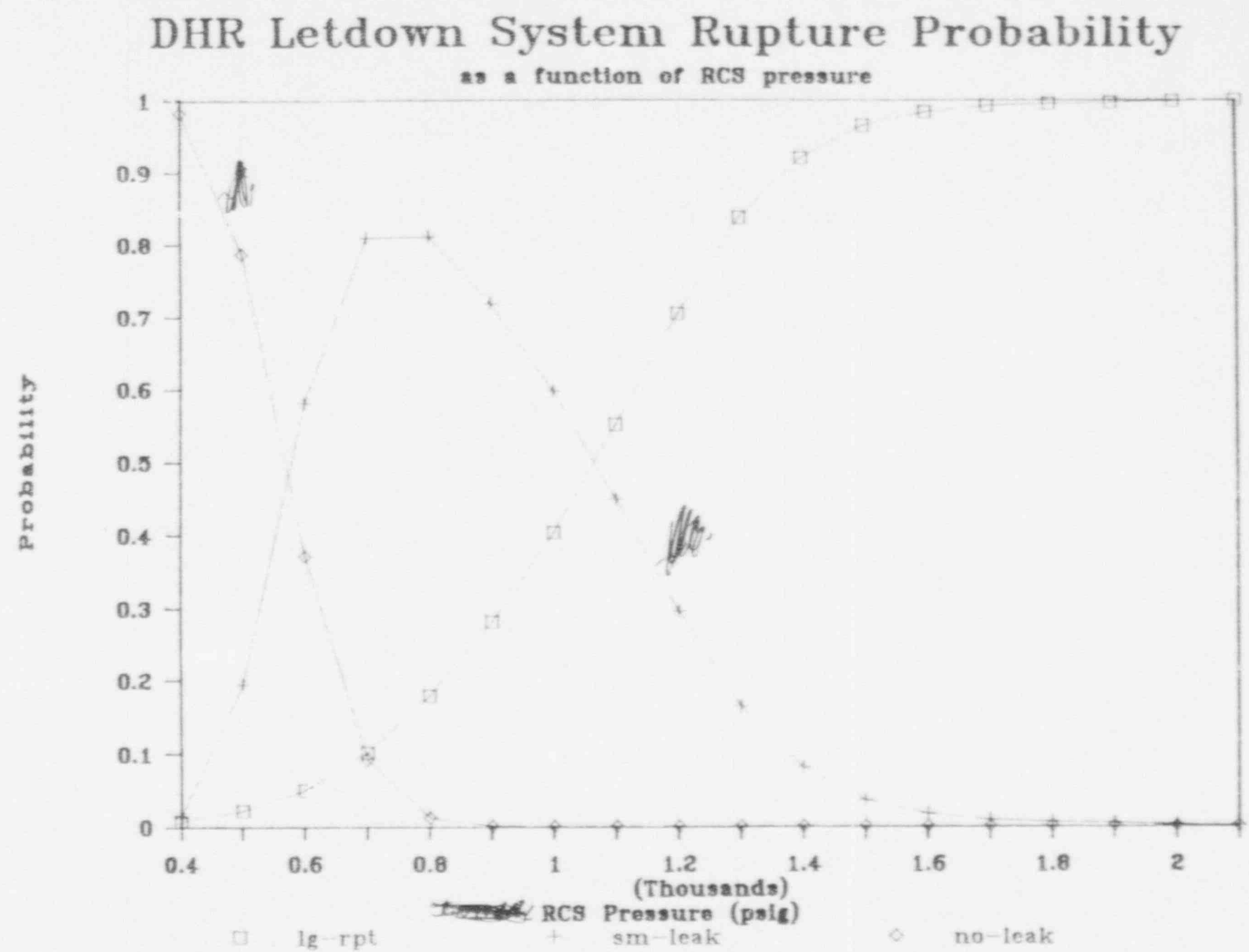


SYSTEM RUPTURE PROBABILITIES CALCULATED USING MONTE CARLO SIMULATION

- o EVNTRE Computer Code (NUREG-1150) used to perform Simulation
  - Local system pressure sampled from postulated distribution e.g. Normally distributed with mean 2100 psi, std dev 50 psi
  - Component failure pressure sampled from postulated dist. e.g. 12-inch SCH20 pipe, lognormally dist. with median 1660 psig, log std dev 0.36.

SYSTEM RUPTURE PROBABILITIES CALCULATED USING MONTE CARLO SIMULATION (CONTINUED)

- TWO SAMPLES COMPARED:
  - SYS PRESS > FAIL PRESS, THEN COMPONENT FAILED
  - SYS PRESS < FAIL PRESS, THEN NO FAILURE
- RUPTURE PROBABILITY IS FRACTION OF MONTE CARLO OBSERVATIONS RESULTING IN RUPTURES


DHR Letdown System Rupture Probability
as a function of RCS pressure
Probability
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
(Thousands)
RCS Pressure (psig)
lg-rpt
sm-leak
no-leak

COMPONENT FAILURE PROBABILITIES CAN BE CALCULATED UTILIZING SEISMIC FAILURE EQUATION

- REF: R.P.KENNEDY ET AL, NUCLEAR ENGINEERING AND DESIGN, VOL 59, NO.2, AUGUST 1980.

- PROBABILITY OF FAILURE AT 2100 PSIG FOR A 12-INCH SCH20 PIPE (MED. = 1660 PSIG, LOGARITHMIC STD DEV = 0.36):

- PROB(FAIL PRESS < 2100 PSIG) = PHI((LN(2100)-LN(1660))/0.36)
  = PHI(0.65)
  = 0.742

$\phi\left(\frac{\ln 2100-\ln 1660}{0.36}\right)$

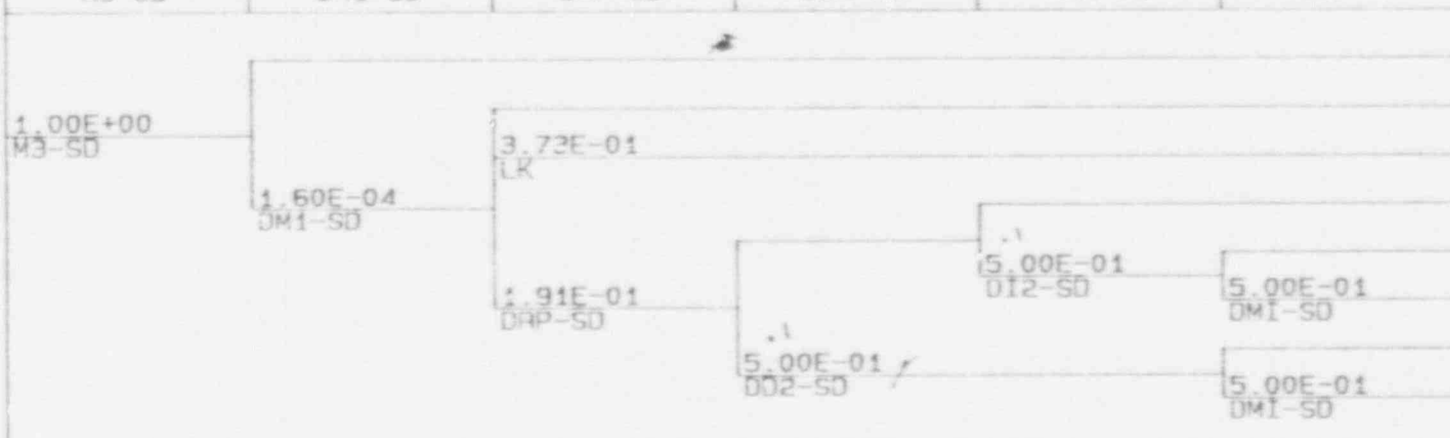
UNCERTAINTIES IN HARDWARE AND OPERATOR CAPABILITIES JUSTIFIES USE OF HEP SCREENING VALUES FOR POST-RUPTURE EVENTS

- GIVEN RAPIDNESS OF INTERFACING SYSTEM PRESSURIZATION, RECOVERY BEFORE RUPTURE VERY UNLIKELY.
- LACK OF PROCEDURES, TRAINING AND AWARENESS OF ISLOCA NECESSITATES SCREENING FOR POST RUPTURE RECOVERY AND MITIGATION.
- GIVEN LARGE UNCERTAINTY IN HUMAN PERFORMANCE, EQUIPMENT AVAILABILITY ISSUES NOT ADDRESSED IN DETAIL.

Instrumentation
MAN/MACHINE INTERFACES

PROCEDURES

Plant Cooldown Mode-3 (ShutDown)	DHR MOVs DH-11/12 Opened too soon	Rupture of low press sys	Operators fail to detect ISLOCA	Operators fail to isolate ISLOCA	Operators fail to mitigate release	SEQUENCE PROB.	SEQUENCE CLASS	SEQ#
M3-SD	DM1-SD	DRP-SD	DD2-SD	DI2-SD	DMI-SD			



SEQUENCE PROB.	SEQUENCE CLASS	SEQ#
1.00E+00	OK	1
6.98E-05	OK-op	2
5.97E-05	LK-ncd	3
7.64E-06	LK-ncd	4
3.82E-06	REL-mit	5
3.82E-06	REL-lg	6
7.64E-06	REL-mit	7
7.64E-06	REL-lg	8

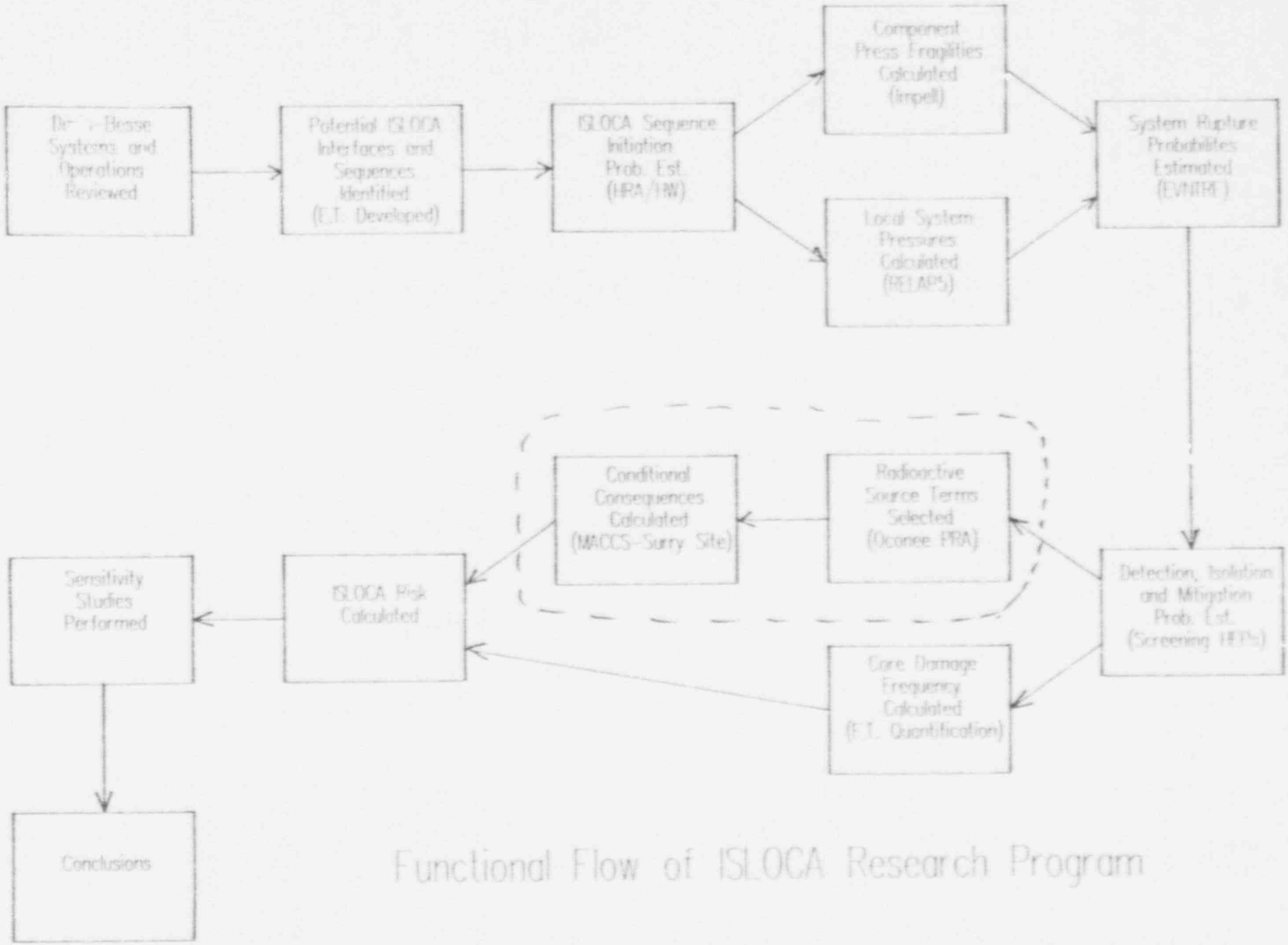
ISLOCA E.T. for Davis-Besse DHR .etdown (Shutdown) DB-DHR-D.TRE 2/11/90

Need to consider more in this regime. What can be done in area of mitigation, recovery, etc.?

.01 / 2.1

3.82 x10^-7 / 2.1

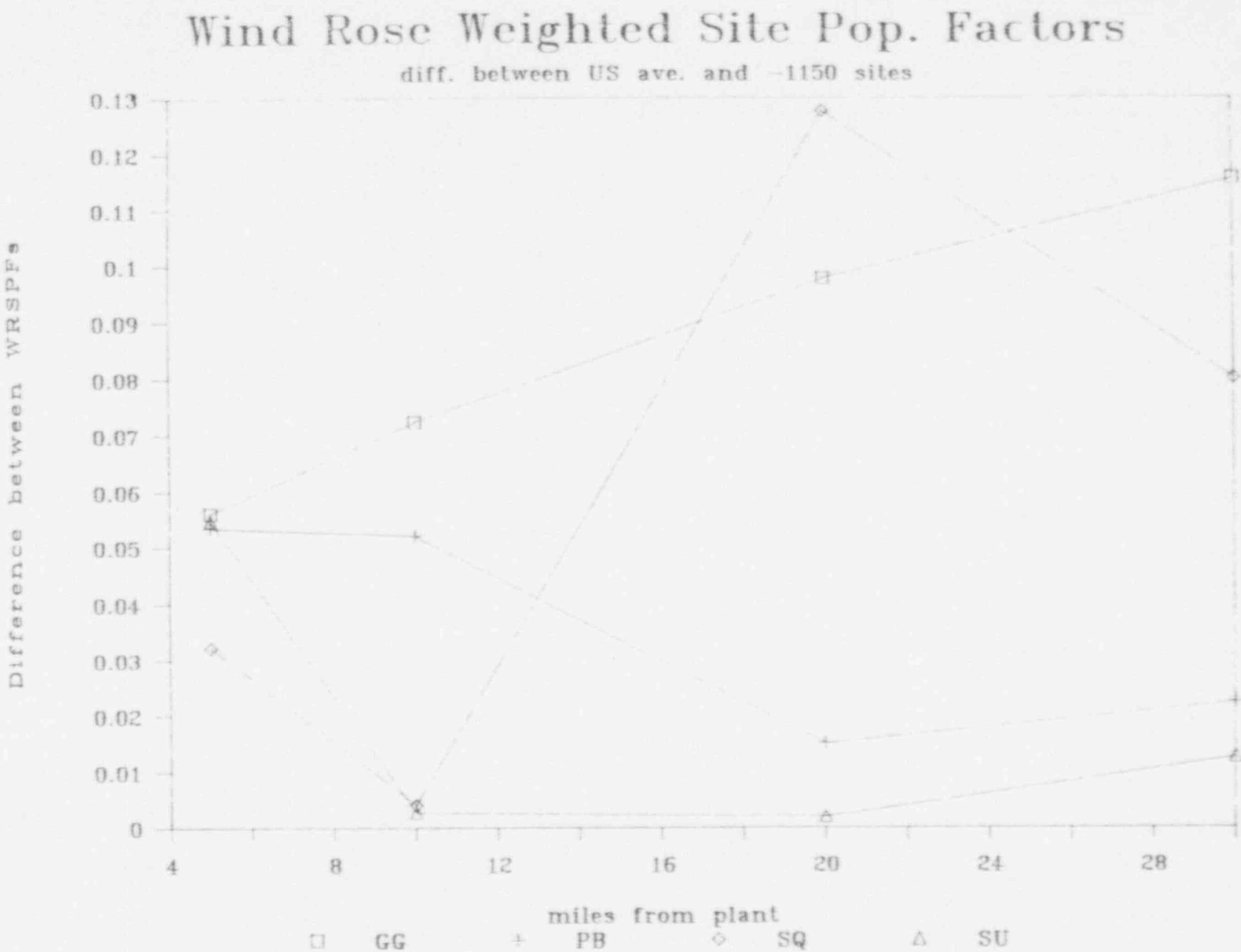
1.6 x10^-7



Functional Flow of ISLOCA Research Program

SOURCE TERMS AND SITE DATA ESTIMATED UTILIZING EXISTING INFORMATION

- Information on B&W plants is limited, source term and release timing taken from Oconee PRA (NSAC/60.
- Industry-wide average site population estimated using Sandia Siting Study (NUREG/CR-2239)
- NUREG-1150 sites compared to average population, Surry selected as representing average site (for MACCS input)


Wind Rose Weighted Site Pop. Factors
diff. between US ave. and ~1150 sites
Difference between WRSPFs
0.13
0.12
0.11
0.1
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
4
8
12
16
20
24
28
miles from plant
GG
PB
SQ
SU

DIFFERENCE BETWEEN NUREG-1150 SITES AND U.S. AVERAGE SITE

Site Population Factors

	5mi.	10mi.	20mi.	30mi
Grand Gulf	0.065	0.069	0.091	0.110
Peach Bottom	0.056	0.043	0.007	0.021
Sequoyah	0.002	0.004	0.054	0.012
Surry	0.065	0.012	0.016	0.007
Zion	0.637	0.618	0.471	0.423

Wind Rose Weighted Site Population Factors

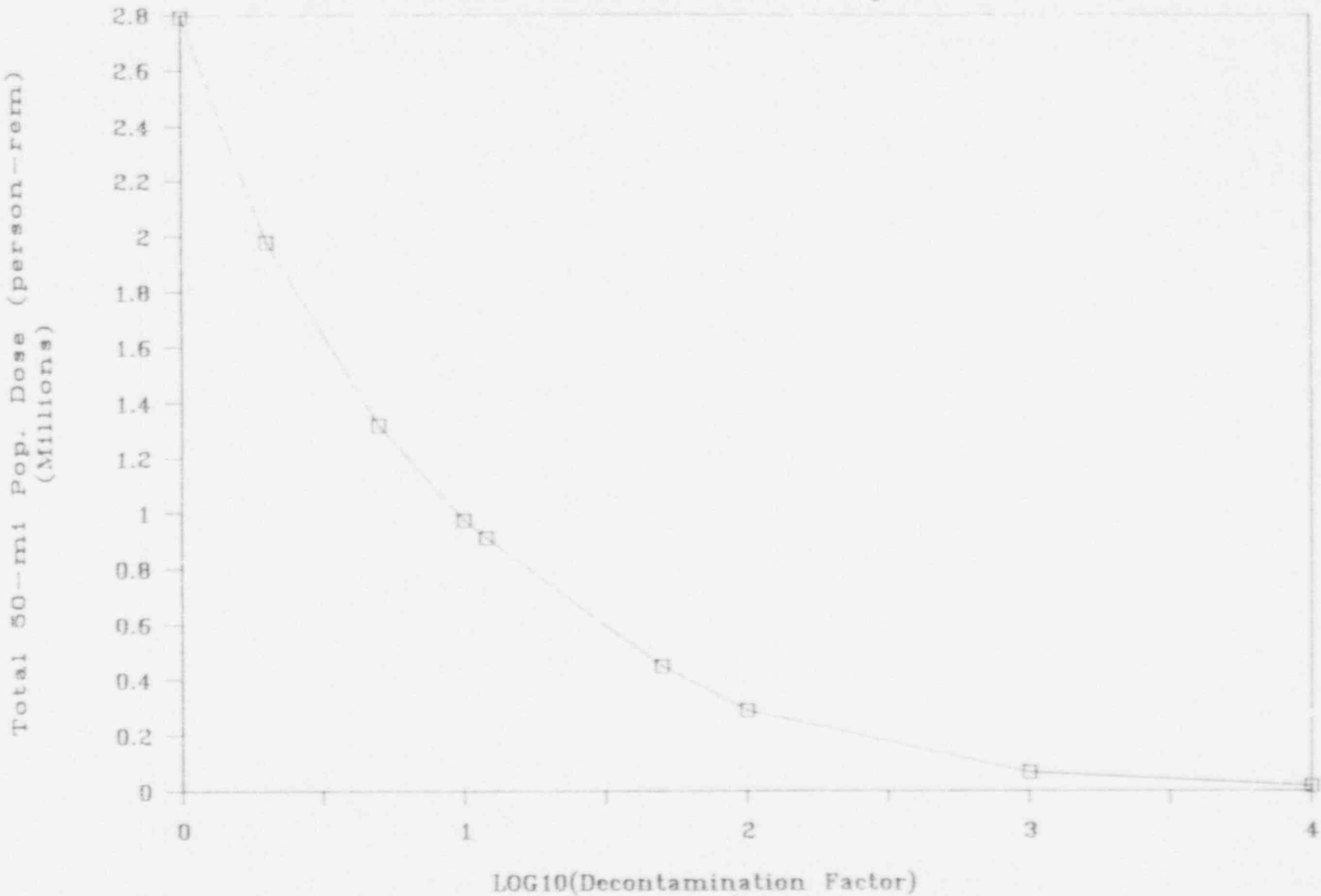
	5mi.	10mi.	20mi.	30mi
Grand Gulf	0.056	0.072	0.098	0.115
Peach Bottom	0.053	0.052	0.015	0.022
Sequoyah	0.032	0.004	0.127	0.080
Surry	0.055	0.003	0.002	0.013
Zion	0.809	0.745	0.567	0.517

CONDITIONAL CONSEQUENCES CALCULATED USING MACCS

- o Second draft NUREG-1150 utilized:
  - MACCS-pc version 1.5.11
  - Surry evacuation strategy
  - Decontamination Factors:
    - Large Release DF=1
    - Mitigated Rel DF=10
  - Consequence Measures:
    - Early Fatalities
    - Latent Cancers (total grid)
    - Population Dose (50-mi.)

MACCS Results for Davis-Besse ISLOCA

Oconee source term and Surry site



MACCS CONSEQUENCE RESULTS FOR A RANGE OF POSSIBLE DFs

(OCONEE SOURCE TERM, SCALED TO DAVIS-BESSE POWER, AND THE SURRY SITE)

CONSEQUENCE MEASURE	DF=1	DF=5	DF=10	DF=100
POPULATION DOSE (PERSON-REM, 50-MI.)	2.8E+6	1.3E+6	9.7E+5	2.9E+5
LATENT CANCERS (TOTAL GRID)	4.5E+3	1.5E+3	8.9E+2	1.4E+2
EARLY FATALITIES	3.6E-2	3.0E-4	5.8E-5	1.2E-6

Idaho National Engineering Laboratory

HUMAN RELIABILITY ANALYSIS (HRA) METHODS IN SUPPORT OF ISLOCA

D. I. GERTMAN

EG&G Idaho, Inc.

AN INTEGRATED TEAM APPROACH TO HRA WAS USED

LERs
Bayesian [illegible] info

- ENSURE ALL THE TYPES OF ACTIONS ARE REPRESENTED

Walkdowns
(qualitative)

- IDENTIFY AND SCREEN HUMAN INTERACTIONS WHICH MAY BE RISK SIGNIFICANT

- DEVELOP A DETAILED DESCRIPTION OF IMPORTANT HUMAN ACTIONS

- SELECT AND APPLY APPROPRIATE MODELING TECHNIQUES

- EVALUATE THE IMPACT OF THESE SIGNIFICANT HUMAN ACTIONS

- QUANTIFY THE PROBABILITIES FOR THE VARIOUS HUMAN ACTIONS

- DOCUMENT THE INFORMATION FOR UNDERSTANDABILITY AND TRACEABILITY.

MAJORITY OF THE DETAILED HRA EFFORT CENTERED UPON ASSESSING FOUR EVENT SEQUENCES

- Start/up Involving DH11/12 or DH 21/23 left open
- Low Pressure Injection ISLOCA Scenario
- Shutdown (cooldown) with Premature Opening of DH11 and 12
- HPI Involving the Quarterly Stroke Test of HP B, C, and D (no MU&P Flow)
- HPI Scenario Involving the Quarterly Stroke Test of HP2A, MU&P Flow

What Was Modeled

The Type of Errors Modeled for ISLOCA Sequences Included Latent Errors and Errors of Commission as well as Errors of Omission

HRA Modeling Techniques Employed

- THERP type HRA Event Trees
- HRA type Fault Trees
- Modified THERP Tree for instances when errors actually provide Plant Protection (COMET)

Quantification Techniques or Sources Used

- NUCLARR
- Direct Expert Estimation (Seaver and Stillwell, 1981)
- Human Cognitive Reliability (HCR) Estimations
- THERP Tables
- Data collection forms were designed for recording PSF information

Questions on how to include some of the failure modes that have occurred or are perceived to be able to occur.

ISLOCA Inspection Team Findings were Used As Performance Shaping Factors

- The absence of ISLOCA Procedures
- The lack of training for ISLOCA
- Lack of valve status indication for HP27/29 and DH 21/23
- Lack of ISLOCA awareness
- Lack of procedures for response to computer alarm presentation (alarms could be ignored for quite some time before being responded to)

Inspection Team Findings (Continued)

- Procedures were Deficient suggest a different term for this.

  -Failure of a procedure to mention one of the values required to open the HP Vent Line

  -Lack of Warnings, Cautions, or Notes related to the potential of ISLOCA

- Ergonomics

  -Local Control Station Tagging for a Gutted Valve "DH1556" had no tags indicating it was inoperable

  -Lighting levels for operation of local valves was uneven

  -Control room ergonomics were good (except for absence of valve status board)

- Motivation

  -Many personnel interviewed seemed well motivated

This appears to be vague. How do we measure motivation?

ERRORS OF COMMISSION WERE ASSESSED

TO DETERMINE THEIR CONTRIBUTION TO ISLOCA

DEFINITION OF ERRORS OF COMMISSION

ERRORS OF COMMISSION ARE THOSE WHICH ARE COMMITTED AS A RESULT OF AN INTENTIONAL ACT OR AS A RESULT OF IMPROPER EXECUTION

ERRORS OF COMMISSION DO OCCUR

- Auto Accidents
  - -Gas pedal pressed instead of brake in accident situation
- Fires
  - -Flour poured on grease fires
  - -Water poured on gasoline fires
- Aircraft
  - -Crashes due to Navigation Errors
  - -Flaps left down at takeoff

IDENTIFICATION OF ERRORS OF COMMISSION IS DIFFICULT

- ERRORS IN EXECUTION LESS DIFFICULT TO IDENTIFY, AND HAVE HIGH ACCEPTANCE

- ERRORS IN INTENTION ARE DIFFICULT TO IDENTIFY, AND HAVE LOW ACCEPTANCE

MODELING AND QUANTIFICATION OF ERRORS OF COMMISSION

- HRA EVENT TREES AND FAULT TREES CAN HANDLE ERRORS OF COMMISSION IF IDENTIFIED
- QUANTIFICATION TECHNIQUES CAN HANDLE "EXECUTION TYPE" ERRORS OF COMMISSION WELL
- QUANTIFICATION TECHNIQUES HAVE DIFFICULTY HANDLING "INTENTIONAL TYPE" ERRORS OF COMMISSION
- HRA ANALYSIS SOUGHT TO IDENTIFY ERRORS OF COMMISSION THROUGH STRUCTURED QUESTIONING AND QUANTIFIED THROUGH EXPERT JUDGMENT TECHNIQUES.

STRUCTURED QUESTIONING CONSISTED OF FOUR ITEMS

- DO PATHWAYS OF ERROR EXIST AROUND EXISTING PROCEDURES AND ADMINISTRATIVE CONTROLS?
- CAN INDICATORS BE READ INCORRECTLY, AND WHAT AIDS EXIST TO PREVENT THIS?
- IS TIME A CONTRIBUTING FACTOR IN TASK EXECUTION?
- CAN INDIRECT CONTROL ACTIONS BE TAKEN AND WHAT AIDS EXIST TO PREVENT THIS?

AN EXAMPLE ERROR OF COMMISSION PATHWAY

- During Shutdown a potential pathway for ISLOCA can be established by opening DH11 & 12 at higher than acceptable pressures

- This was deemed probable as:

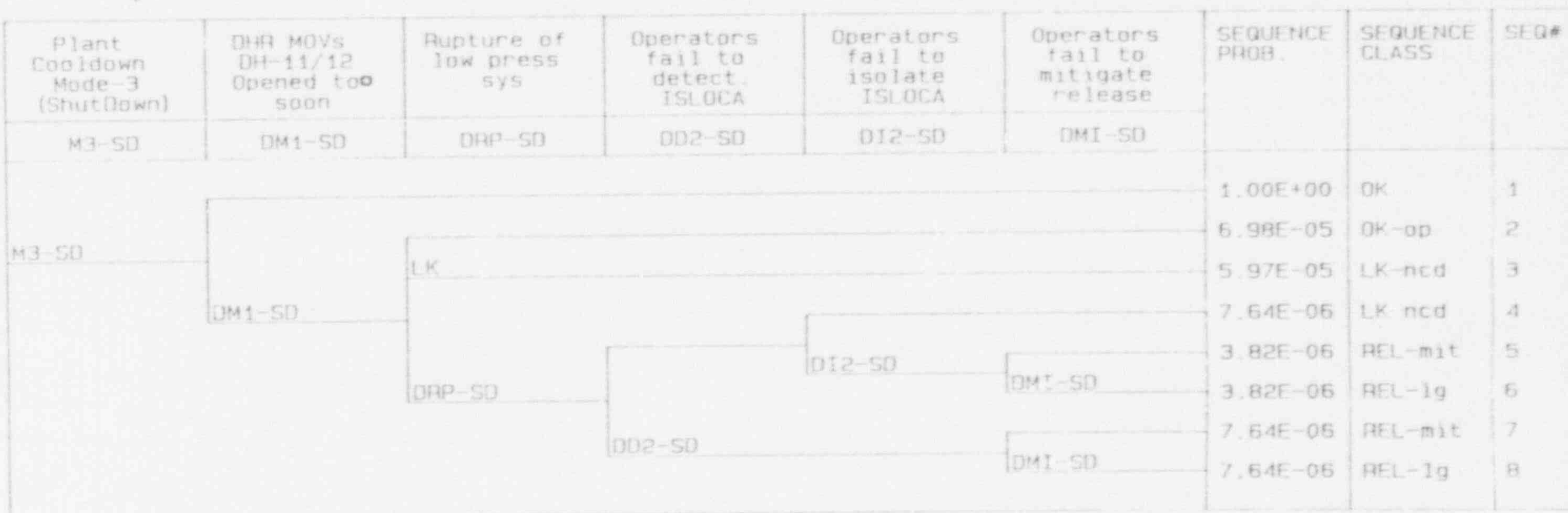
  - Procedure routinely allows jumpering of DH12 due to large dead band

  - Timing could become a motivational factor to to minimize time in hot standby

  - Interlocks can and are disabled on a routine basis

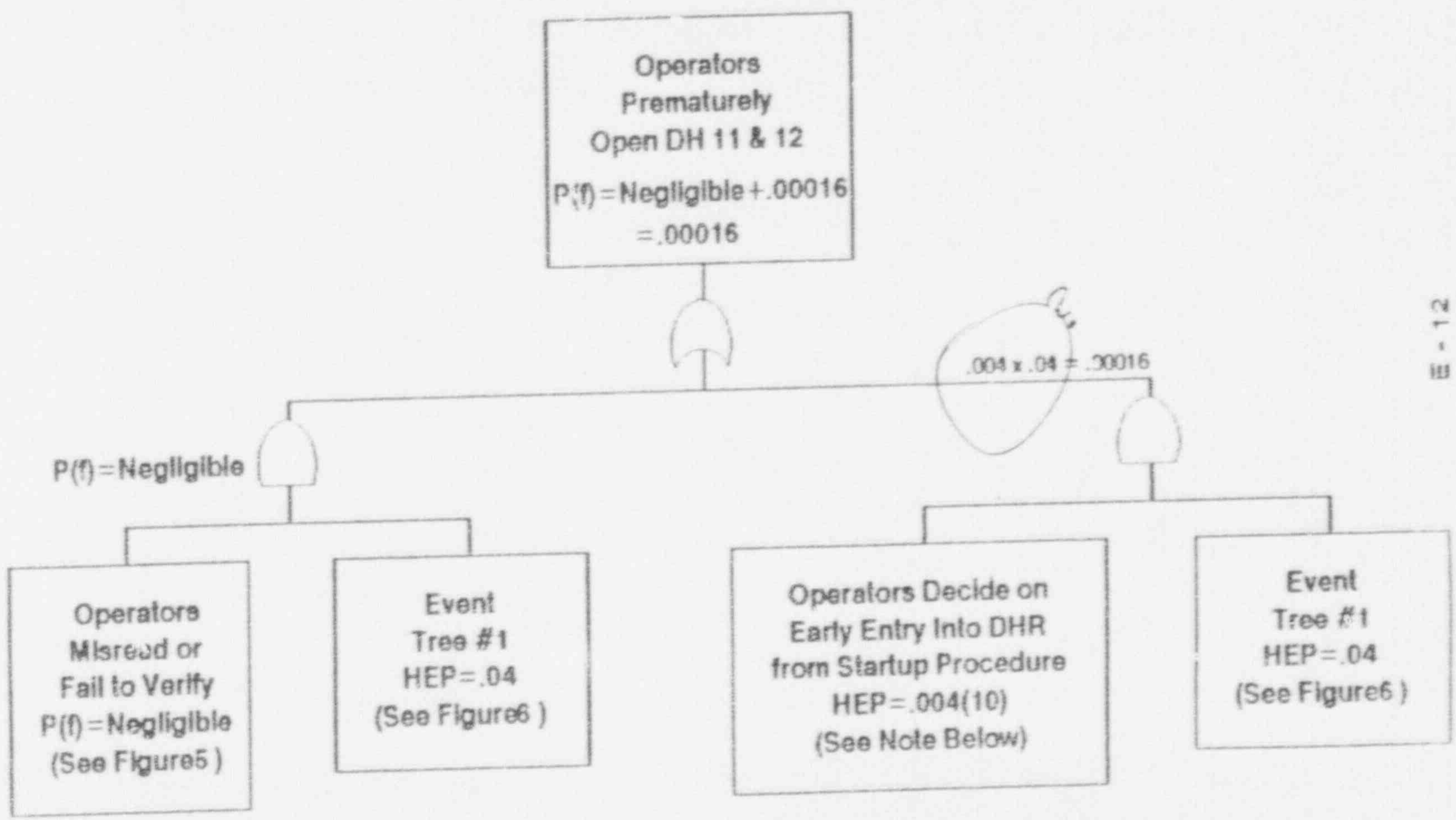
  - Procedures contains no warnings or cautions

- Operators are primed for error by the combination of these factors



ISLOCA E.T. for Davis-Besse DHR Letdown (Shutdown) DB-DHR-D.TRE 2/23/90

Figure 4: Fault Tree for ISLOCA DM1-SD
Operators Prematurely Open DH 11 & 12



NOTE: This was judged to be a cognitive action. So, the HEP was derived through the use of Expert Estimation & a Delphi Technique.

Figure 5: Fault Tree for ISLOCA DM1-SD Part 2
Operators Prematurely Open DH 11 & 12

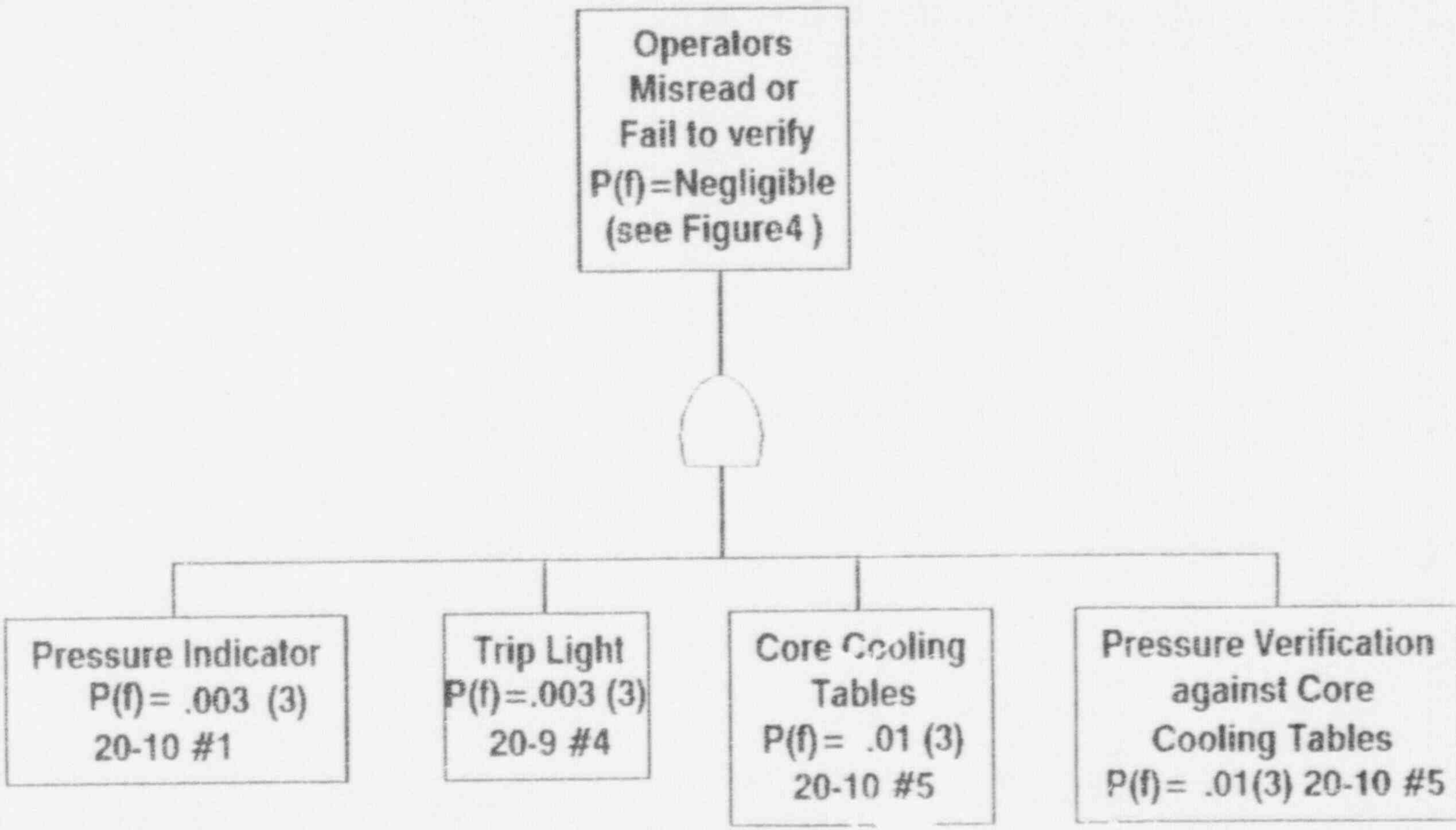
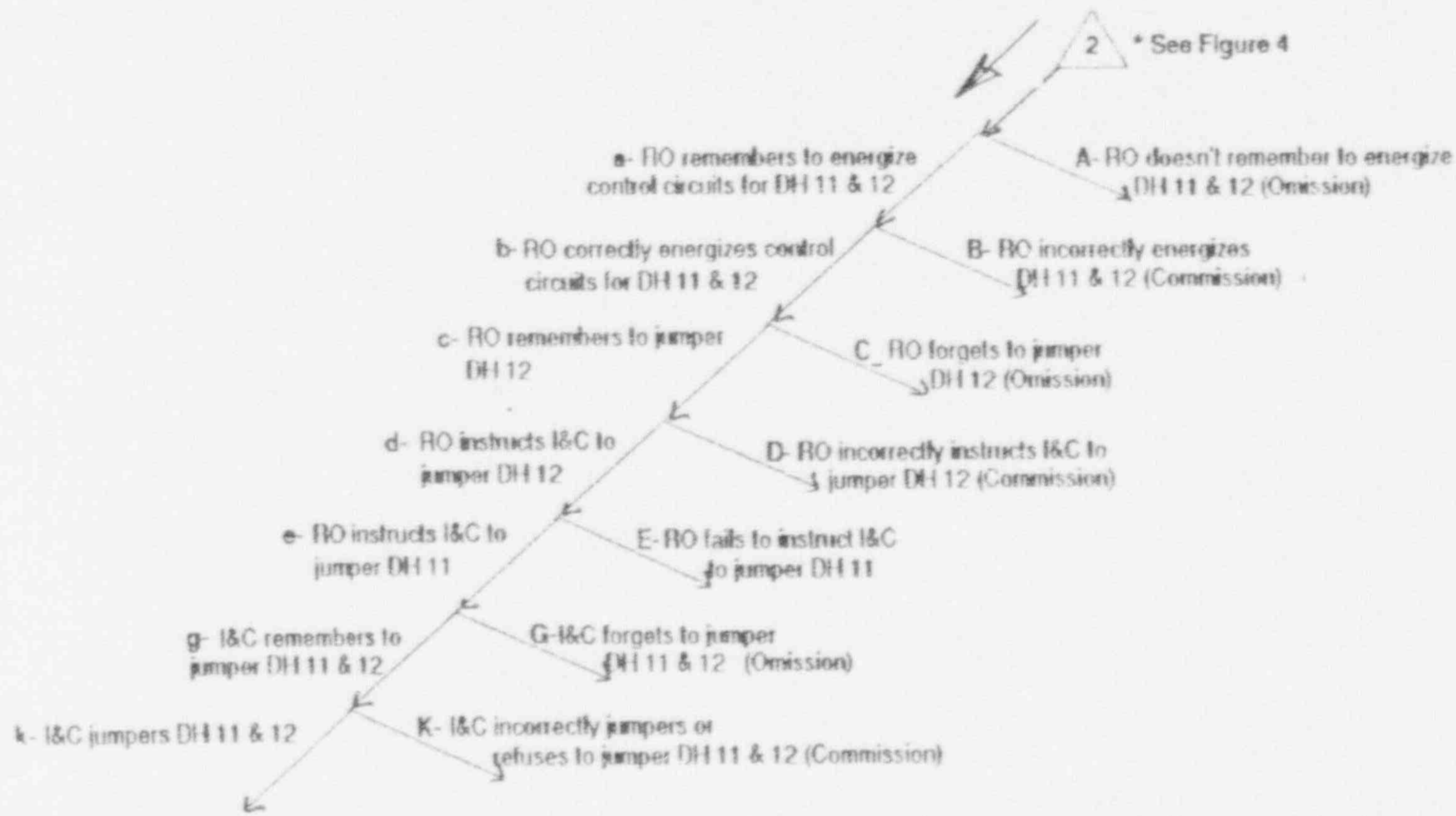


Figure 6: HRA Event Tree for ISLOCA DM1-SD Premature Opening of DH-11 & DH-12

Note: The total HEP for this tree =.04



Mechanisms "at work"

WHAT ARE THE COGNITIVE COROLLARIES

- BOUNDED RATIONALITY - OVERSIMPLIFICATION OF THE PROBLEM (REASON, 1983)
- IMPERFECT RATIONALITY - PREVIOUS SOLUTIONS PERCEIVED AS APPROPRIATE (REASON 1983)
- SALIENT CUES SUGGEST A SOLUTION WHICH IS INAPPROPRIATE (MORRIS AND TLOUSE, 1988)
- PASSIVE (LATENT) FAILURE IN DESIGN PROCESS WHICH SETS THE STAGE (REASON, 1989)

FINDINGS AND CONCLUSIONS

- An Integrated Process is required in order to perform a meaningful HRA
- HRA Models must be carefully selected for Application
- Latent Errors should be considered for ISLOCA Risk Analysis
- Modeling errors of commission require a thorough plant specific analysis

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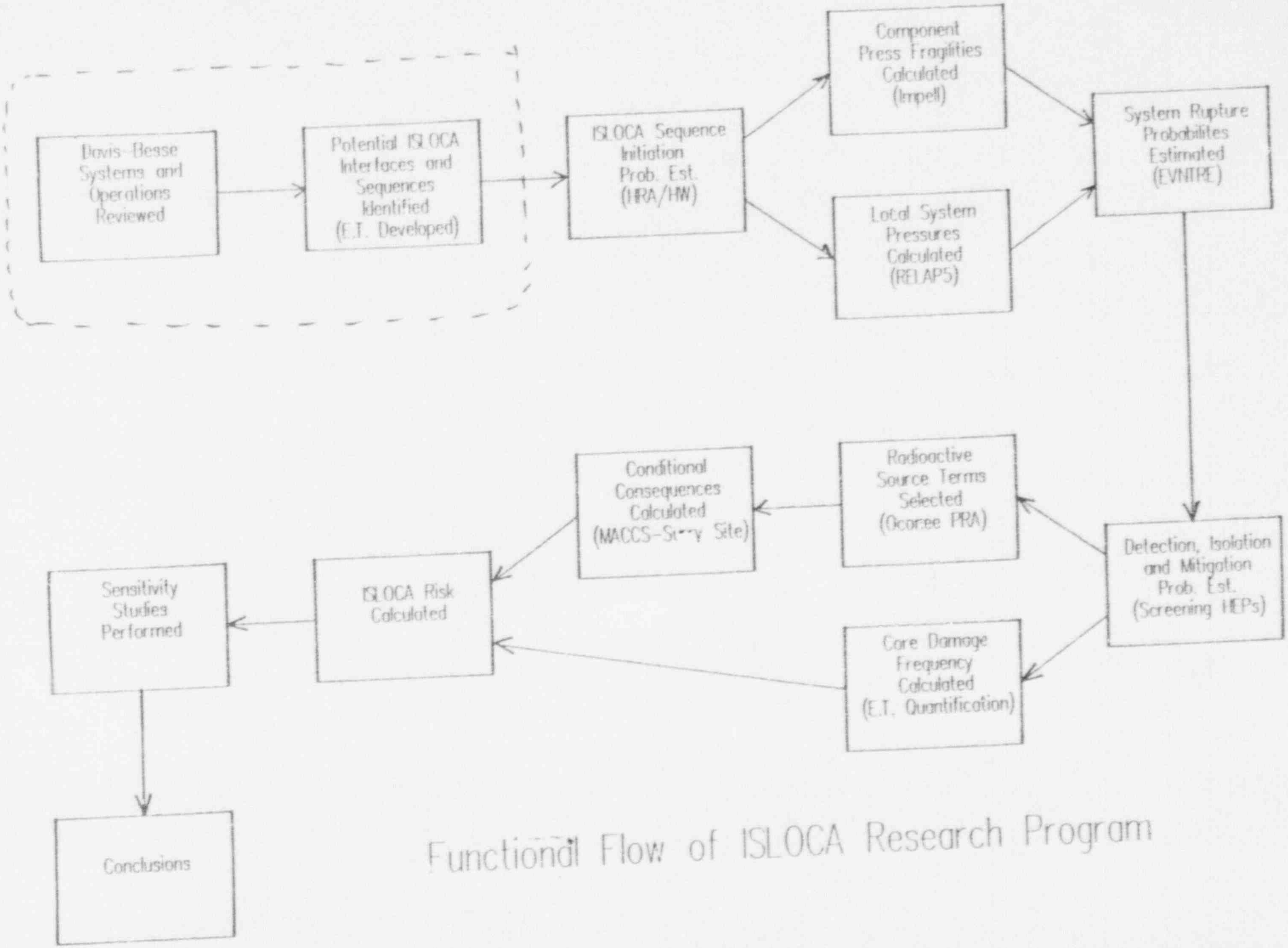
Identification of Davis-Besse ISLOCA Sequences

February 26/27, 1990

W. J. Galyean

EG&G Idaho, Inc.



Functional Flow of ISLOCA Research Program

ISLOCA SEQUENCES INITIATED BY MULTIPLE HUMAN ERRORS OR COMBINATIONS OF HUMAN ERRORS AND HARDWARE FAULTS

o HISTORICAL EXPERIENCE INDICATES:

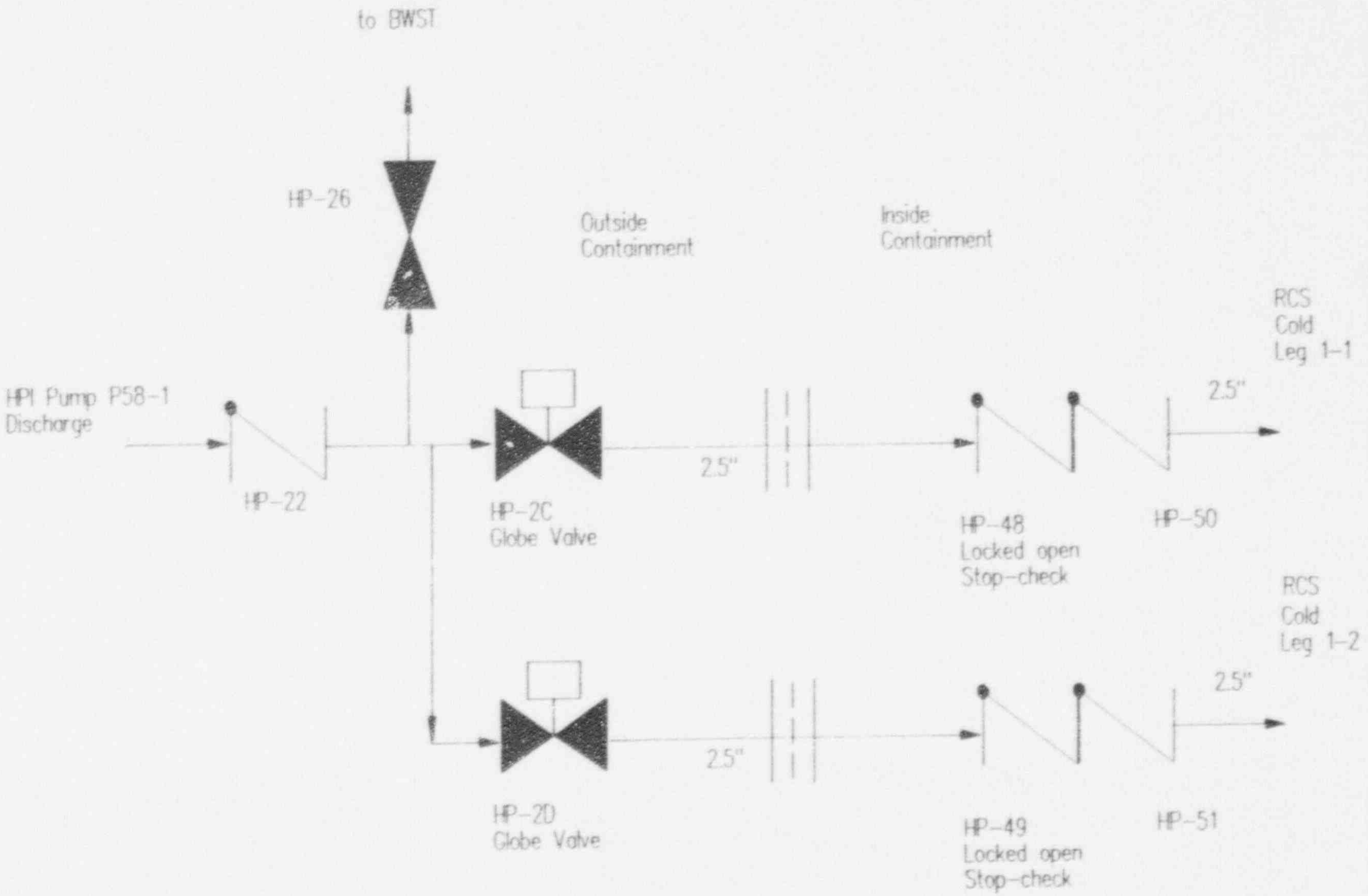
- IMPROPER VALVE LINEUP AND OPERATOR ERRORS IN MISPOSITIONING VALVES - RELATIVELY LIKELY.

- RANDOM AND CATASTROPHIC FAILURES OF REDUNDANT VALVES IN STANDBY - NOT SUPPORTED.

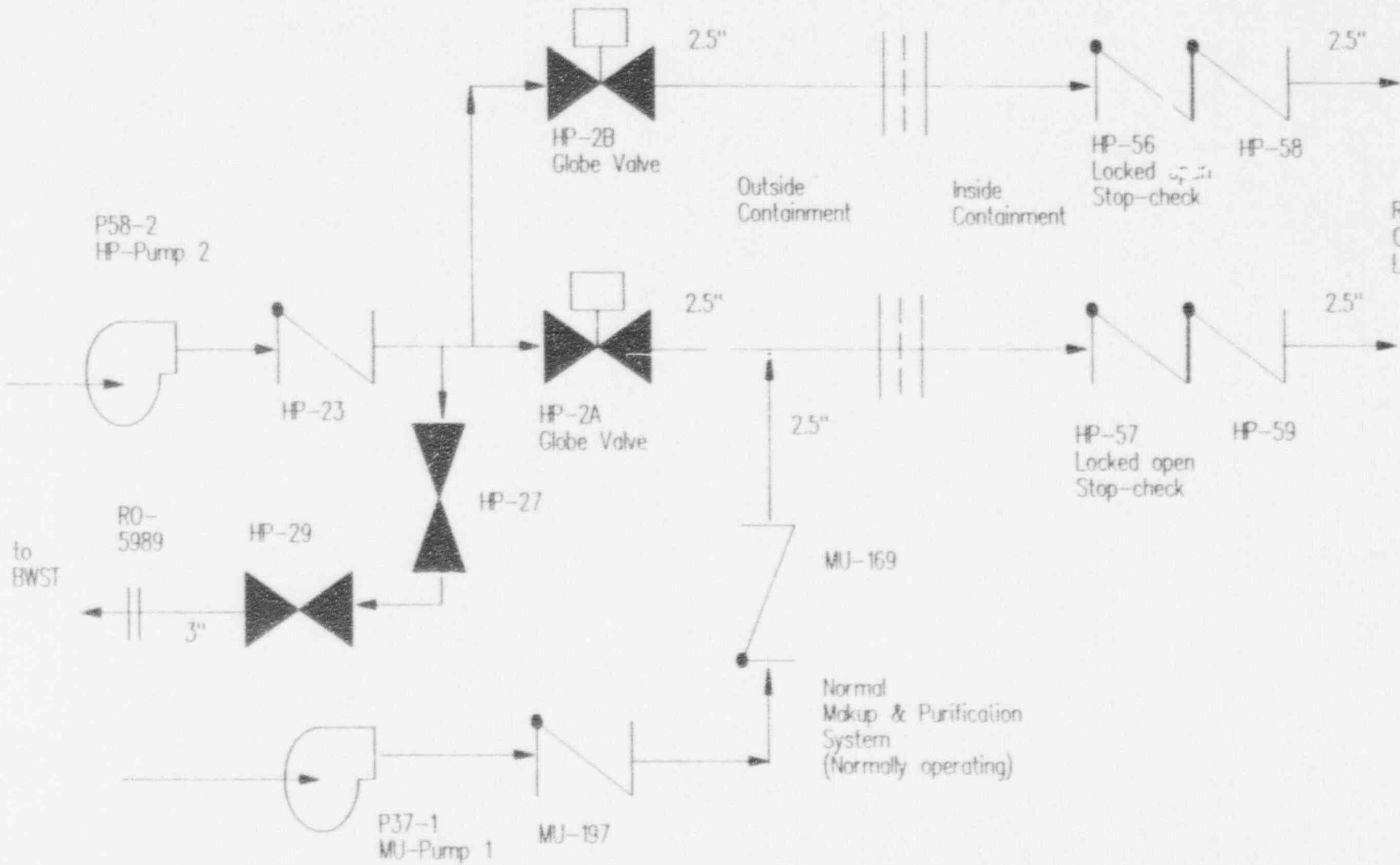
REVIEW OF D-B SYSTEMS AND OPERATIONS LEADS TO IDENTIFICATION OF ISLOCA INTERFACES AND SEQUENCES

- 1-INCH AND SMALLER LINES, AND <200 GPM DEEMED RISK INSIGNIFICANT.
- THREE ISLOCA INTERFACES IDENTIFIED: HPI, LPI, AND DHR LETDOWN.
- FIVE POSSIBLE ISLOCA SEQUENCES IDENTIFIED:
  - HPI
  - MU&P/HPI
  - LPI
  - DHR-STARTUP
  - DHR-SHUTDOWN

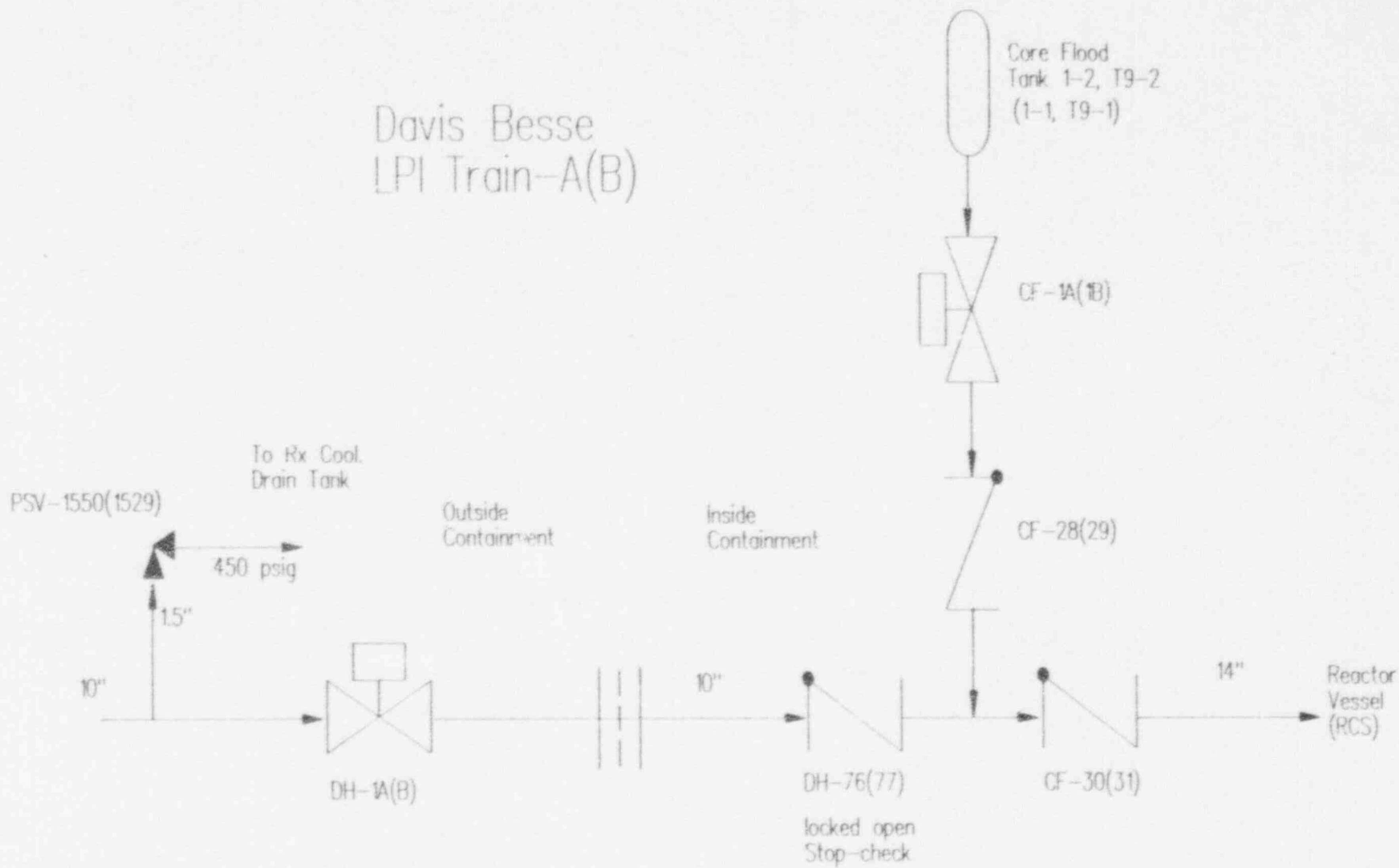
Davis Besse
HPI Legs C & D



Davis Besse
MU&P/HPI Leg-A/B

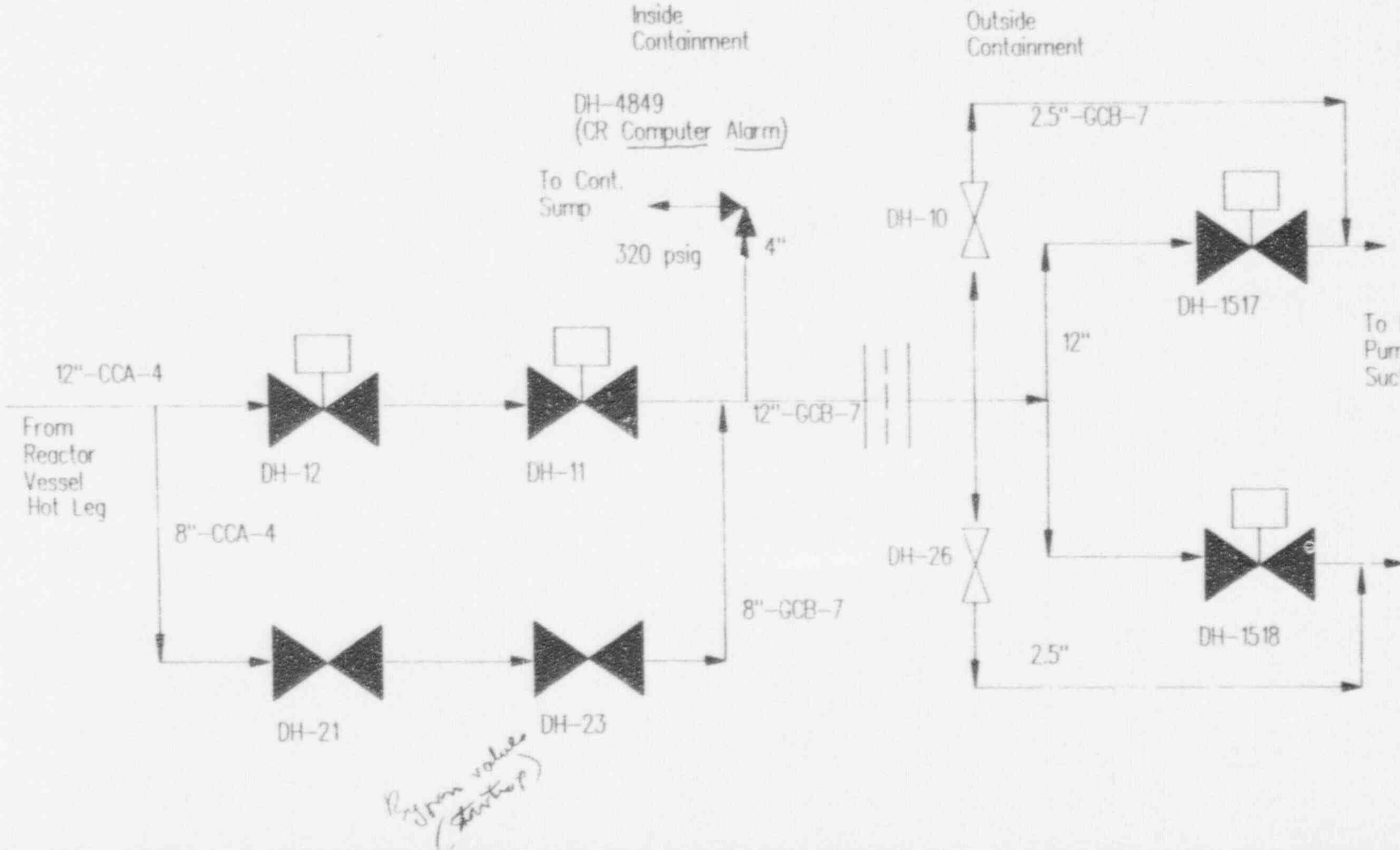


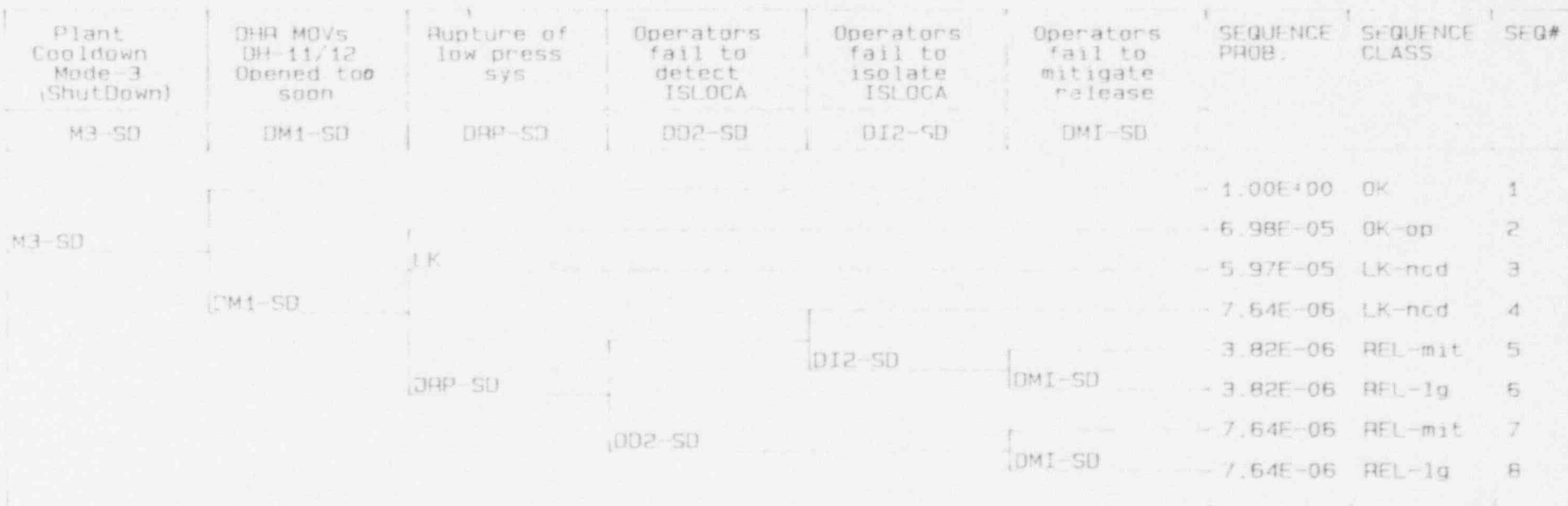
Davis Besse
LPI Train-A(B)



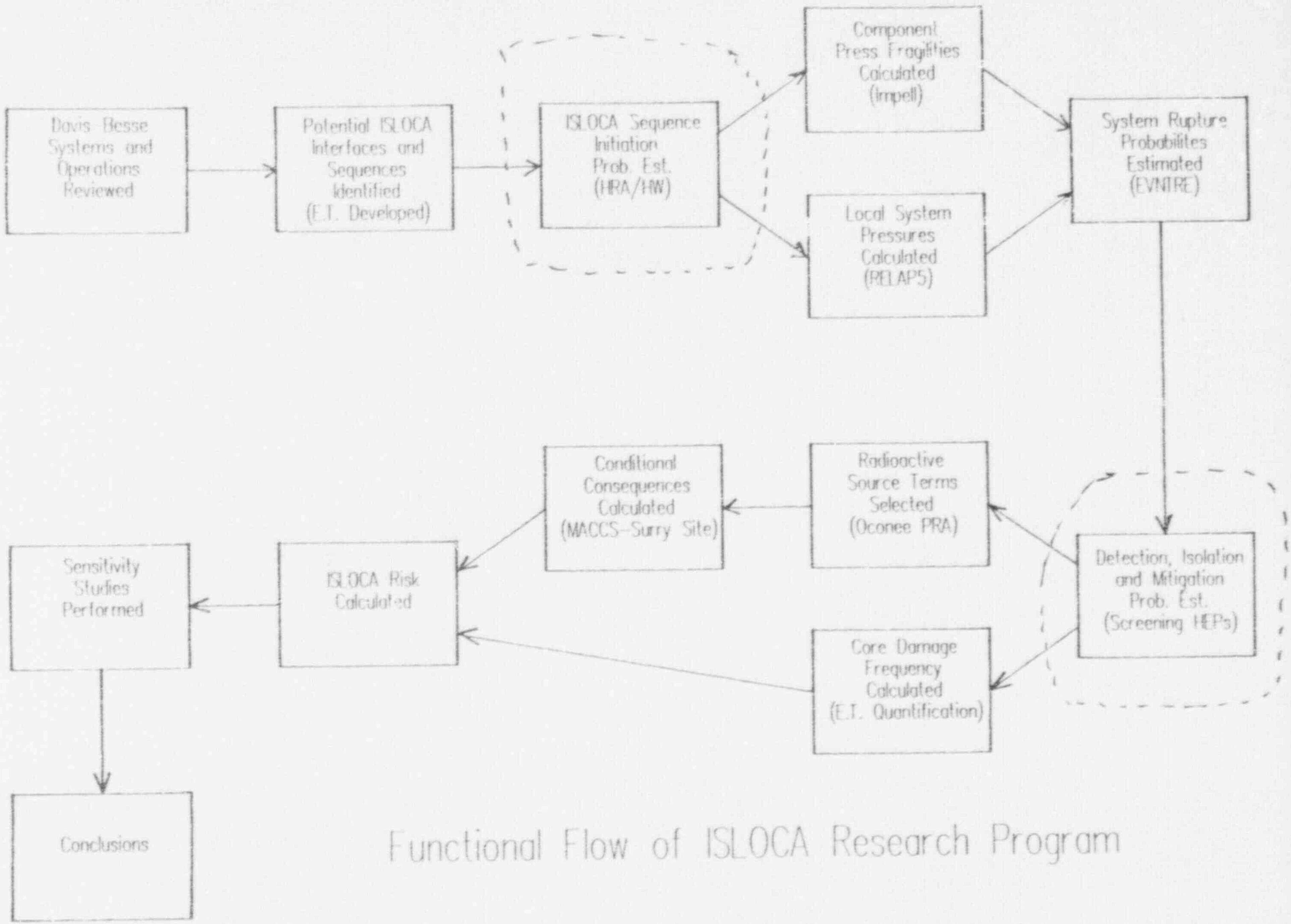
Davis Besse
DHR Letdown

DHR system in operation when:
RCS temp. < 280 F and
RCS press. < 266 psig
(Mode-4)





ISLOCA E.T. for Davis-Besse DHR Letdown (Shutdown) DB-DHR-D.TRE 2/23/90



Functional Flow of ISLOCA Research Program

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ASSESSMENT OF THE POTENTIAL FOR

ISLOCA AT DAVIS-BESSE

INTRODUCTION

DUANE J. HANSON

FEBRUARY 27, 1990

EG&G Idaho, Inc.

AGENDA FOR DISCUSSION OF ISLOCA PROGRAM RESULTS

8:40 - 9:00	Introduction - D. J. Hanson
9:00 - 9:30	Identification of Davis-Besse ISLOCA Sequences - W. J. Galyean
9:30 - 10:00	HRA Methods in Support of ISLOCA - D. I. Gertman
10:00 - 10:40	Event Tree Analysis Support - W. J. Galyean
10:40 - 11:00	Quantification and Hardware Sensitivity Study Results - W. J. Galyean
11:00 - 11:30	HRA Sensitivity Study Results - H. S. Blackman
11:30 - 12:00	Conclusions - W. J. Galyean

ISLOCA PROGRAM OBJECTIVES

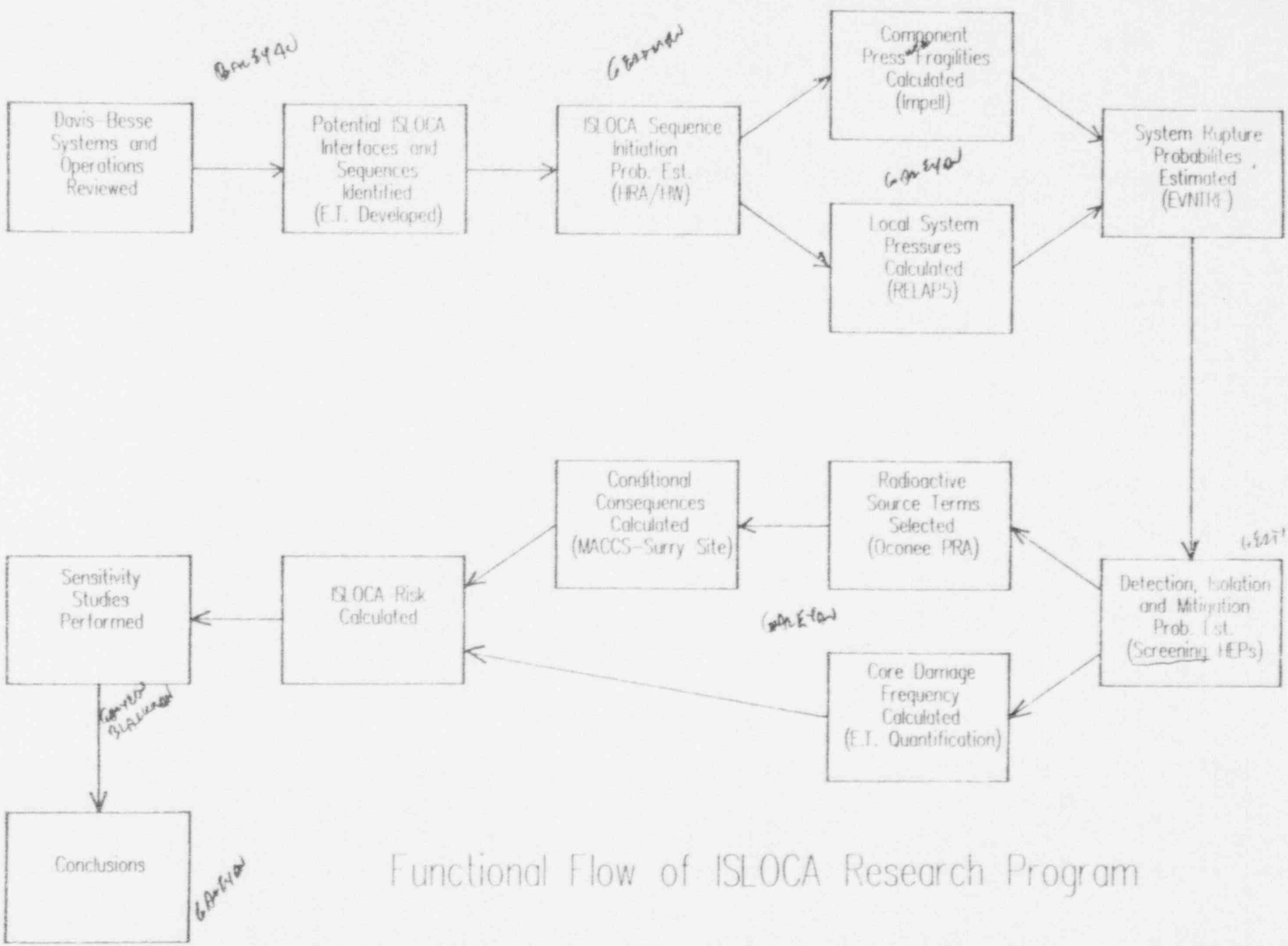
- o DEVELOP A FRAMEWORK FOR EVALUATING ISLOCA RISK
  - HARDWARE
  - HUMAN ERROR
  - POTENTIAL FOR RISK REDUCTION

- o DETERMINE THE EFFECTS OF HUMAN ERROR AND THEIR CONTRIBUTORS

- o EVALUATE THE FRAGILITY OF LOW PRESSURE SYSTEMS
  - FAILURE LOCATIONS
  - FAILURE PROBABILITIES

ISLOCA PROGRAM OBJECTIVES (CONT.)

- o IDENTIFY AND DESCRIBE POTENTIAL ISLOCA SEQUENCES
  - TIMING
  - ACCIDENT MANAGEMENT STRATEGIES
  - EFFECT ON OTHER EQUIPMENT
- o ESTIMATE CONSEQUENCES OF POTENTIAL ISLOCA SEQUENCES
  - CORE MELT FREQUENCY
  - OFFSITE CONSEQUENCES
  - RECOMMEND CONSEQUENCE REDUCTION ACTIONS



Functional Flow of ISLOCA Research Program

IMPORTANT INTERFACING SYSTEMS FOR DAVIS-BESSE

- High Pressure Injection (HPI) discharge lines
- Low Pressure Injection (LPI) discharge lines
- Decay Heat Removal (DHR) letdown lines

IMPORTANT CONSIDERATIONS FOR FOLLOWING PRESENTATIONS

- EFFECT OF HUMAN ACTIONS AS INITIATORS FOR AN ISLOCA (commissions)

- RELATIVE EFFECT OF HUMAN ERRORS AND HARDWARE FAILURES AS CONTRIBUTORS TO ISLOCA CORE MELT FREQUENCY AND RISK

- COMPONENTS THAT WOULD FAIL WHEN EXPOSED TO OVERPRESSURE

- INFLUENCE OF PROCEDURES, TRAINING, AND INSTRUMENTATION ON THE CAPABILITIES OF PLANT PERSONNEL