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ISLOCA Research Program

Final Report

Prepared by
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EG&G Idaho, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This report contains a compilation of information generated during the ISLOCA research program. Presented is a screening analysis and a procedures guide for performing an ISLOCA evaluation. This methodology has been distilled from past analyses performed for the U.S. Nuclear Regulatory Commission and documented in a series of NUREG/CR reports. The methodology comprises five distinct steps: (a) containment penetration screening, (b) interfaces for ISLOCA analysis, (c) mechanisms for failing the pressure boundary, (d) construction of event trees and estimation of rupture probabilities, and (e) quantification of the event trees. Included in the methodology are steps required for a detailed human reliability analysis. In addition, this report presents a BWR ISLOCA evaluation, a survey of PWR auxiliary building designs and identification of one design deemed most disadvantageous with respect to ISLOCA risk, and a PWR ISLOCA cost/benefit analysis.

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EXECUTIVE SUMMARY

This report contains a compilation of efforts completed as part of the ISLOCA research program. Included here are: a screening method and procedure for performing an ISLOCA analysis; an ISLOCA analysis of a BWR; an analysis of the effects of a PWR auxiliary building judged as representing a bounding design with respect to ISLOCA risk; a collection of PWR insights gained during the course of the program.

The screening technique utilizes aspects of the pressure isolation interface that have been found to influence the potential for an ISLOCA. Specifically, the number of redundant pressure isolation valves (PIVs), the hardware and administrative controls, and the pressure capacity of the low pressure system are all considered in the screening, which is evaluated in a subjective quantitative manner.

The major methodological elements of the ISLOCA procedure are summarized in Table ES-1. This table outlines the steps and calculations required to assess ISLOCA risk for either boiling or pressurized water reactors. This assessment comprises five distinct steps: (a) ISLOCA interface screening, (b) identification of ISLOCA interfaces requiring further analysis, (c) evaluation of mechanisms for failing the interface pressure boundary, (d) construction of event trees and estimation of rupture probabilities, and (e) quantification of the event trees.

Presented is a simplified modeling technique to identify potentially vulnerable configurations. This technique involves a review and analysis of the interfacing systems, procedures, and operations involving the pressure isolation valves and includes ISLOCA screening criteria. These screening criteria eliminate from further analysis interfaces that are too small or that are unlikely to be sources of risk-significant ISLOCAs. Recommendations and guidance to incorporate operator detection, diagnosis, isolation, and mitigation into the ISLOCA event tree are also provided.

Table ES-1. ISLOCA Evaluation Guide.

Step 1—Identify and Screen Interfaces
• Remove 1-in. and smaller lines
• Remove lines with three or more pressure isolation valves
• Remove lines rated for reactor coolant system pressure.
Step 2—Interfaces for Analysis
• List pressure isolation valves and types
• Identify major operations affecting the pressure isolation valves.
Step 3—Identify Mechanisms for Failing the Pressure Isolation Boundaries
• Hardware failures
• Human errors.
Step 4—Construct Event Trees and Estimate Rupture Probability
• Perform bounding analysis on rupture probability:
- Estimate local pressure in interfacing system.
- Estimate local pressure fragility.
- Calculate rupture probability.
• Estimate Recovery Actions:
- Is recovery possible?
- How much water is available?
- How much time is available?
Step 5—Calculate Core Damage Frequency
• Place event tree end states into plant damage groups.

The BWR ISLOCA analysis only includes a screening assessment and bounding calculations. Based on these examinations, no interfaces warranted further study, and therefore no detailed calculations were performed. For the most part, the design margins available in the BWR systems preclude the possibility of an overpressure induced rupture. In those instances where an interfacing system component is susceptible to rupturing, the redundancy of PIVs and the interlocks and controls in place render the possibility sufficiently low as to be negligible.

One of the ISLOCA concerns identified during the research program is the effect of an ISLOCA induced severe environment in the auxiliary building. To further examine this issue a survey was performed on 42 of the 76 PWR auxiliary buildings in the U.S. Out of these 42 units (29 W, 8 CE, and 5 B&W), one specific design was chosen that was judged to represent a worst-case design with respect to ISLOCA (a W unit). This auxiliary building was evaluated for the ISLOCA sequences postulated for the B&W plant analysis. The calculations show flooding to be an important issue in the accident progression, limiting the amount of time operators might have to recover the situation. The larger sized ruptures will result in both RHR pumps failing within a few minutes of the rupture, and all ECCS failing within 20 to 30 minutes. For the smaller ruptures, RHR still fails rapidly; however, HPI and charging will survive several hours. The situation is termed artificial because none of the B&W plants examined were found to have an auxiliary building deemed to be an unfavorable design with respect to ISLOCA. Although the situation is artificial, the calculations show the potential effects of auxiliary building design on ISLOCAs.

ACKNOWLEDGMENTS

The authors wish to express their appreciation to Dr. Gary R. Burdick, NRC Generic Issue 105 Program Manager, for guidance and support.

ACRONYMS

ADS	Automatic depressurization system
ASEP	Accident sequence evaluation program
ATOG	Abnormal transient operating guidelines
BWR	Boiling water reactor
BWST	Borated water storage tank
CRD	Control rod drive
CS	Core spray
CST	Condensate storage tank
CVCS	Chemical and volume control system
DFs	Decontamination factors
DHR	Decay heat removal
ECA	Emergency contingency action guidelines
ECCS	Emergency core cooling system
ECI	Emergency coolant injection
EOPs	Emergency operating procedures
EPGs	Emergency procedures guidelines
ERGs	Emergency response guidelines
HCR	Human cognitive reliability
HELB	High-energy line break
HEPs	Human error probabilities
HPCI	High pressure coolant injection
HPI	High pressure injection
HRA	Human reliability analysis
ISLOCA	Inter-system loss-of-coolant accident
ISO	Isolation
LDS	Leakage detection system
LERs	Licensee event reports
LPCI	Low pressure coolant injection
LPCS	Low pressure core spray
LPI	Low pressure injection
MIT	Mitigation
MOVs	Motor-operated valves
MU&P	Makeup and purification
MSIV	Main steam isolation valve
NRP	Nonresponse probability
ORG	Optimal recovery guidelines
P&IDs	Piping and instrumentation diagrams
PIB	Pressure isolation boundary
PIVs	Pressure isolation valves
PRA	Probabilistic risk assessment
PSFs	Performance shaping factors
PWR	Pressurized water reactor
RCIC	Reactor core isolation cooling
RCS	Reactor coolant system
RHR	Residual heat removal
RWCU	Reactor water cleanup
SP	Suppression pool

ISLOCA Research Program Final Report

1. INTRODUCTION

1.1 Purpose

The objective of the Inter-System Loss-of-Coolant Accident (ISLOCA) Research Program is to provide the U.S. Nuclear Regulatory Commission (NRC) with qualitative and quantitative information on the hardware, human factors, and accident consequence issues that dominate the nuclear power plant risk associated with an Inter-System Loss-of-Coolant Accident. To accomplish this objective, a methodology has been developed based on probabilistic risk assessment (PRA), human factors, and human reliability analysis (HRA) techniques. This methodology can be used to:

- Identify the risk contribution from both hardware failures and human errors issues and to develop recommendations for risk reduction

- Identify the effects on ISLOCA risk of specific types of human errors and their root causes

- Evaluate the fragility of low-pressure systems exposed to high-pressure, high-temperature reactor coolant. These evaluations include identification of likely failure locations and estimates of probabilities of failure

- Identify and describe potential ISLOCA sequences including sequence timing, possible accident management strategies, and the effects of possible ISLOCAs on other equipment and systems

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

1.2 Scope

This report represents the concluding efforts of the NRC's ISLOCA Research Program. Previous reports[1,2,3] have presented both the methods that were developed and their use on three PWR plants. This report presents a refinement of that method, a procedure for screening high/low pressure interfaces with respect to ISLOCA risk and, an examination of a BWR plant using bounding techniques.

1.3 Background

The Reactor Safety Study, WASH-1400,[4] 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. A subset of these ISLOCAs were called V-sequences or event V. These sequences were characterized by the failure of motor-operated valves and/or check valves, and the rupture of low-pressure piping outside of the containment building. Some event V ISLOCAs were shown to be significant contributors to risk because the rupture caused core damage and fission products bypassed the containment and were discharged directly to the environment. Subsequent PRAs, including Draft NUREG-1150[5] results for Surry and Sequoyah, have identified ISLOCAs as relatively significant 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 initiation frequency.[6,7]

Recent events at several operating reactors have been identified as precursors to an ISLOCA. These events have raised questions about the previously assumed frequency of occurrence, potential initiators, and means of identifying and mitigating this potential accident; suggesting that the risk associated with an ISLOCA may be larger than previously estimated and that additional measures may be needed to prevent and/or control these accidents. In response to these questions, 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 Mr. Eric S. Beckjord. The ISLOCA Research Program described in this report was initiated as a result of this memorandum.

1.4 Report Organization

This report contains seven sections, each of which is basically a stand-alone document. Following this introduction (Section 1) is the ISLOCA screening procedure. This procedure utilizes the experience and insights gained during the ISLOCA research program to evaluate the various ISLOCA-related aspects of a particular high/low pressure interface. This determination, although made in a rudimentary way, quantifies the potential susceptibility to an ISLOCA type of event. The ISLOCA evaluation procedure is described in Section 3. This provides a more detailed method for probabilisticly quantifying the likelihood of an ISLOCA. A BWR ISLOCA analysis follows in Section 4. This analysis comprises a screening and bounding analysis. This analysis yielded the conclusion that ISLOCA is not of concern for this particular plant. Section 5 revisits the PWR analysis performed previously in the program to examine the effect of what is judged to be a "worst-case" auxiliary building design (with respect to ISLOCA concerns) on ISLOCA risk. A qualitative survey was first performed during which the subject for the detailed analysis was chosen. This subject auxiliary building was then mated to the B&W plant analysis performed previously and the auxiliary building environmental effects studied. This information was then used to estimate the time available for the operators to recover from the ISLOCA. Section 6 contains a compilation of insights gained during the course of the ISLOCA research program.

2. ISLOCA SCREENING

Lower pressure rated systems connected to the high pressure RCS are typically isolated by a series of normally closed valves. This pressure isolation boundary (PIB) usually comprises one or more check valves and/or one or more motor operated valves (MOV). Initial screening of the interfaces can be accomplished based on three criteria: (a) a minimum pipe size of 1-in., (b) systems designed for high pressure (i.e., at least 67% of RCS pressure), and (c) systems isolated by redundant normally closed and locked manual valves that are independently verified to be closed and locked before plant startup.

The 1-in. criteria is based on thermal-hydraulic code calculations[a] that predict an expected flow rate of about 200 gpm out of a 1-in. pipe.[b] The 200-gpm limit is used because that is typically close to the capacity of both the auxiliary building sump pumps and of the borated water storage tank (BWST) makeup. Screening at 67% of RCS pressure is justified by the common practice of hydrotesting fluid systems to 150% of their design pressure. Lower values may be justified on other bases. Interfaces that use locked closed manual valves and independent verification of the locks are screened out based on the lack of credible failure mechanisms that can result in overpressurizing the low pressure system. Locking the valve guards against the inappropriate opening and independent verification ensures against the valves being left open. These measures produce very low probabilities of PIB failures.

Each PIB that survives the initial screening should be evaluated using the following criteria. The following procedure utilizes a tally system for recording the effects of the various ISLOCA-related design features. Values recorded by the tally can be viewed as a gross quantitative estimate for the probability of an ISLOCA by representing the negative exponent to the power of 10 (i.e., a tally of 5 would roughly correspond to an ISLOCA frequency of 1E–5/yr). The tally involves a number of integer variables and one real variable. There is one variable for each phase of the potential ISLOCA sequence, namely: initiating event (IE), rupture (RUPT), diagnosis (DD), isolation (ISO), and mitigation (MIT). Each of these is discussed in the following sections.

a. Calculations are given in Volume 2, Appendix F, Section F.5 of Reference 1.

b. Based on a simple RELAP model of a 50-ft long, 1-in. schedule 160 pipe with 2200 psi and 550°F on one end and atmospheric conditions on the discharge end.

2.1 Sequence Initiation Potential

ISLOCA initiation comprises those events that result in breaching the pressure isolation boundary. This includes hardware faults, human errors, test and maintenance procedures, or combinations of these. The purpose of the items listed below is to produce a crude measure of the susceptibility of the PIB to being inappropriately opened. A variable name of IE is used to record the ISLOCA initiation potential for a specific interface.

2.1.1 Does the interface contain at least two normally closed pressure isolation valves (PIVs)?
If no, need more PIVs.
If yes, IE = 3.

2.1.2 Does the interface contain at least three normally closed PIVs?
If no, add zero to IE.
If yes, add one to IE.

2.1.3 Is each check valve PIV leak tested after every cold shutdown, and is each motor-operated PIV position indicated in the control room and verified to be closed and locked (or disabled) before every startup? (True independence requires a previously uninvolved person reporting on the *status* of a valve or interlock and not a "check to see if the valve is closed" type of instruction.)

If no, add zero to IE.
If yes, add two to IE.

2.1.4 If ECCS injection valves (PIVs) are stroke-tested while the plant is at power, are at least two redundant PIVs verified to be closed before the test? (Note: non-testable check valves position typically cannot be verified without leak testing.)
If valves are not stroke-tested at power, subtract zero from IE.
If no, subtract one from IE.
If yes, subtract zero from IE.

2.1.5 Are the motor-operated PIVs interlocked against being opened at high pressure?
If no, add zero to IE.
If yes, are the interlocks ever defeated while the plant is operating (including startup and shutdown)?
If no, add one to IE.
If yes, add zero to IE.

2.1.6 Do plant procedures (both operating and test and maintenance procedures) that cover actions involving PIVs contain warnings and cautions about the importance of maintaining the primary system pressure isolation boundary?
If no, subtract one from IE.
If yes, subtract zero from IE.

2.2 Rupture Potential

Estimating the rupture potential (RUPT) of the low-pressure interfacing system uses the information contained in NUREG/CR-5603[8] and NUREG/CR-5862.[9] For the screening analysis, the local internal pressure is conservatively assumed to be equal to the reactor coolant system pressure. Typically, significant overpressure protection (using relief valves) is only achieved in those configurations in which a choke plane exists upstream of a relief valve. A choke plane is created when the flow area is increased significantly such that water flowing through suddenly flashes to steam because of the sudden decrease in pressure caused by the volume expansion immediately downstream of the flow restriction. The steam volume creates enough backpressure to inhibit flow through the choke plane and together with the relief valve, prevents the local pressure from reaching that upstream of the choke plane. A number of factors determine the pressure generated in the interfacing system. In addition to relief valves and flow restrictions, these factors include flow losses through pipe and other system components, and the amount of flow through the system (without ruptures). In the extreme case, a completely closed system (i.e., no flow out of the system) that is normally filled (which is typical) will, upon opening of the PIB, experience a virtually instantaneous pressurization to full RCS pressure.[c]

A rupture probability for each component in the system can be estimated assuming the local internal pressure is equal to the RCS pressure. The equation that estimates the median failure probability for each component is given as

$$P_i = phi\{[\ln(LP/F_i)]/beta_i\}$$

where

P_i	=	individual failure probability of component i
phi	=	standard normal (Gaussian) function
ln	=	natural logarithm
LP	=	local internal pressure
F_i	=	median pressure fragility of component i (from NUREG/CR-5862)
$beta_i$	=	uncertainty associated with F_i (from NUREG/CR-5862).

The individual failure probabilities can be combined utilizing the following formula:

$$Total\ probability\ of\ rupture = RUPT$$

$$= 1 - [(1 - P_1)(1 - P_2)(1 - P_3)...] \ .$$

c. This result is described in Reference 1 (Volume 2, Appendix F).

2.3 Diagnostic Potential

Rupture detection and identification (diagnosis) refers to the process performed by the control room crew in understanding details of a specific ISLOCA scenario, should one occur. The primary source of guidance for the crew should be the emergency operating procedures. However, appropriate training in recognizing and addressing these situations is also important. Lastly, the plant design needs to be such that adequate time is available to the control room crew for achieving the correct understanding.

2.3.1 Do plant emergency operation procedures (EOPs) instruct the operators to review the likely indicators resulting from a potential LOCA outside containment? For example, radiation alarms in the auxiliary building, auxiliary building sump pumps starting, actuation of fire alarms and sprinkler systems, etc.

If no, DD = zero.
If yes, DD = one.

2.3.2 Do plant EOPs address the detection, identification (diagnosis), and isolation of possible ISLOCA ruptures or LOCAs outside the containment in a timely manner? If, for example, a primary system leak were to occur via a rupture in the RHR heat exchanger, would the operators reach that point in the EOPs where such a leak would be addressed within 30 min?

If no, subtract one from DD.
If yes, subtract zero from DD.

2.3.3 Are redundant trains of ECCS spatially separated such that severe flooding will not affect all primary system makeup? This would ideally take the form of separating redundant trains of ECCS both horizontally and vertically (i.e., on different floors or elevations but not directly above and below each other such that flooding in the upper equipment room does not drain or flow down into the lower room). However, watertight flood barriers would also be acceptable. Further, if they are isolated from any flooding, primary system charging (CVCS, MU&P, etc.) could also be relied upon in conjunction with the auxiliary feedwater system to provide primary system cooling and makeup (provided the break can be isolated). When considering equipment separation with respect to flooding, drain lines in the auxiliary building rooms need to be considered as well. Specifically, if the drain lines empty into a common sump inside the auxiliary building and these drains do not have check valves to prevent water from backflowing, water from an overflowing sump can backflow into additional rooms.

If ECCS trains are not separate, add zero to DD.
If ECCS trains are separate, add one to DD.

2.3.4 Does the operator training program include large ruptures outside containment [e.g. a rupture of the RHR (DHR) letdown line just outside the containment wall]? Does the training program (for both reactor operators and equipment operators) sensitize the staff to possible ISLOCA events?

If no, add zero to DD.
If yes, add one to DD.

2.4 Rupture Isolation Potential

The procedure of this section addresses the ability to isolate a rupture that has occurred as a result of breaching the PIB and assigns a measure (ISO) to that ability. Because the local environment will be very severe (i.e., about 100% humidity and 212°F), personnel will be precluded from entering the area.[d] In addition, electrical equipment that is not adequately environmentally qualified will likely not survive. Therefore, postulating the availability of remotely operable valves relies upon the survivability of these valves in a severe environment. Typically, the

d. This is documented in Reference 1 (Volume 3, Appendix M).

mechanical portion of the valve is very robust and only the electrical portion is of concern. However, it is common for the valve operators to be able to survive 100% humidity and 212°F (in a few cases valve operators have been qualified for complete submergence). Of more concern than the operators are the control, actuation, and power circuits. These are usually very vulnerable to severe environmentally induced failures and for a valve to be relied upon these components need to be environmentally qualified for the expected conditions or physically removed and isolated from the expected area of concern.

Are remotely operable valves available to isolate the most likely ruptures?

If no, stop screening process; ISO = zero.

If yes, are valve operators sufficiently sized to operate the valve in a maximum delta-P environment?

If no, stop screening process; ISO = zero.

If yes, ISO = one for each valve in series (maximum = three).

2.5 Mitigation Potential

This issue deals with the potential consequences of possible ISLOCA sequences. Although this aspect is unique in that it addresses events after core damage begins, it is included here for completeness. Also, it allows for estimating potential consequences relative to whether or not the radioactive release is mitigated or unmitigated. If core damage occurs and a radioactive release were to occur, options and design features are potentially available for reducing the severity of the release. There are many uncertainties associated with these features. Foremost is the timing of a sequence. The quantity of material released depends partly on the effective decontamination factor produced by the thermohydraulics of the sequence and the design of the auxiliary building. The progress of the accident determines the conditions present in the auxiliary building at the time of the release. For example, the release point might be flooded; however, if the water is saturated at the time the release occurs, very little decontamination will take place.

2.5.1 Are fire suppression sprinkler systems available in the likely rupture locations?
If no, MIT = zero.
If yes, MIT = one.

2.5.2 Will the likely rupture locations be flooded in the event of an ISLOCA?
If no, add zero to MIT.
If yes, add one to MIT.

2.6 Screening Results

Once the individual segments of the screening have been evaluated, a total screening frequency can be calculated. This provides a gross measure of the ISLOCA susceptibility of a particular interface.

ISLOCA screening frequency

$$= (10^{-IE}) * RUPT * (10^{-DD} + 10^{-ISO}) \ .$$

If the screening frequency is estimated at 1E–8 per year or lower, the interface can been screened out and not considered further. This screening value is arrived at by postulating a maximum of 10 interfaces per plant and if each interface was screened out at 1E–8, the total ISLOCA frequency would be bounded by 1E–7, which would still be small compared to typical nuclear power plant results, on the order of 2E–6 per yr.[1,2,3]

3. ISLOCA EVALUATION PROCEDURE

3.1 Introduction

3.1.1 Purpose. Consistent with the objectives described in Section 1.1 of this report, a detailed ISLOCA Evaluation Procedure was developed. This procedure is based on the experience gained in performing the three PWR and one BWR ISLOCA analyses comprised by the ISLOCA research program. The aim of this procedure is to provide a proven framework for determining the risk posed by ISLOCAs.

3.1.2 Background. The first step in the development of this methodology was a review of historical plant operating information publicly available in the United States. This review included an identification and evaluation of all of the Licensee Event Reports (LERs) that involved valve failures resulting from either hardware or human causes or indicated an ISLOCA had occurred. The results of this review 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 to identify systems to be reviewed during the development of the event trees, and for quantification of the failure rates of some interfacing system valves.[1] Appendix A to Reference 1 provides a brief summary of the historical experience related to ISLOCA events.

3.1.3 Scope. The procedure for performing an ISLOCA analysis comprises a number of steps and requires capabilities in a variety of disciplines. These steps are outlined in Figure 1, which depicts the flow of the major procedural tasks.

The first step in quantifying ISLOCA risk is to gather and evaluate information on the plant's interfacing systems. As a result, all systems that interface with the reactor coolant system (RCS) should be identified. Plant-specific information is collected on systems, hardware, operations, and procedures. The system information must include data on hardware maintenance and testing, logic circuits, and power supplies. The required information includes the following:

- Plant procedures:
  - Maintenance
  - Testing
  - Operations

- Piping and instrumentation diagrams (P&IDs)

- Isometric drawings

- Training material

- Hardware design details required to determine failure probability:
  - Valves
  - Flanges
  - Piping sizes
  - Pumps
  - Tanks and heat exchangers.

A plant walkdown is recommended to completely review, develop, and assess the interfacing systems data. The types of information that are obtained during a plant walkdown include the following:

- Detailed information on the hardware that could be involved in an ISLOCA (e.g., control valves, relief valves, piping, flanges, pumps, heat exchangers)

- Detailed information on the procedures and guidelines imposed on plant personnel during startup, normal power operation, and shutdown of the plant

- Detailed information on maintenance and in-service test practices

- Detailed information on the factors that could influence plant personnel performance specific to initiation, detection, prevention, and mitigation of an ISLOCA.

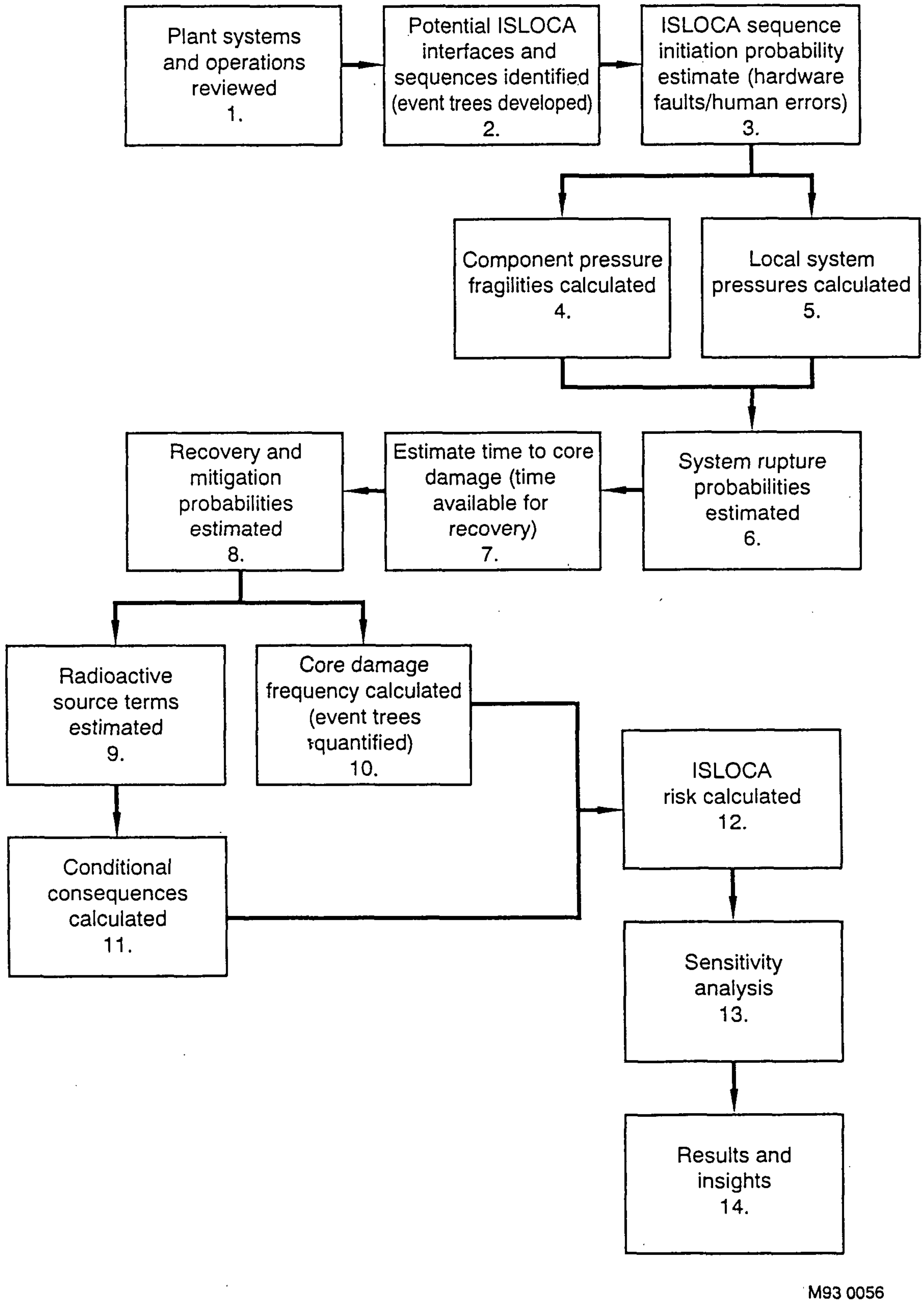

Figure 1. Procedure for performing a plant-specific ISLOCA analysis.

3.1.3.1 Event Tree Development. The first stage in developing the event trees is the application of a screening process to the interfacing systems. The interfacing systems are screened and categorized for further analysis of the potential for core damage if they are overpressurized. The system screening criteria are based on pipe larger than a specified size, potential for containment bypass, and potential for rupture if exposed to reactor coolant system operating pressure. The systems that meet the screening criteria are taken into a second stage of analysis to identify potential ISLOCA initiators and using event trees to develop accident sequences. The second-stage ISLOCA event trees comprise three phases:

- **Initiating events:** Combinations of hardware and human failures that result in a breach of the pressure isolation boundary and allow high-pressure RCS water to enter the lower pressure interfacing system

- **Rupture events:** Events associated with the size and location of a rupture in the interfacing system

- **Postrupture or recovery events:** Events that model the actions and estimate the likelihood that the control room operators will be able to recover from an ISLOCA or mitigate its consequences.

The event trees are developed at the component level and combine the hardware faults and human errors that make up each sequence.

3.1.3.2 Initiating Events: Hardware Failures. Some ISLOCA initiators have contributions from hardware faults. These failures involve the pressure isolation valves (PIVs). Typical hardware failures that can lead to an ISLOCA include failure of the PIV interlocks and check valve and motor-operated valve failures. Hardware failures involving the check valves include internal leakage, internal rupture, failure to reseat, and failure to hold on demand. Motor-operated valves have the same hardware failure modes as the check valves but also can be mispositioned due to component failure and can open spuriously because of electrical circuit failures.

3.1.3.3 Initiating Events: Human Error. Some ISLOCA initiators also have contributions from human errors. These errors are associated with the proper functioning of the motor-operated or testable check valve PIVs. Typical human error-induced events include mispositioning the PIVs, bypassing interlocks, and failure to close a valve when required. To be complete, development of ISLOCA initiating events should include consideration of PIV maintenance and testing procedures. As implied by this list of human error-induced events, a human reliability assessment is an important component of the ISLOCA methodology.

3.1.3.4 Estimate Interfacing System Failure Potential. The conditional probability that the interfacing system will fail upon overpressurization is required to quantify the ISLOCA event tree. The failure probability of each component in the interfacing system is described by a lognormal distribution with a specified median failure pressure and standard deviation. The components modeled include valves, flanges, pipes, heat exchangers, and tanks.

The performance of plant components designed for low-pressure conditions but exposed to the high pressures associated with an ISLOCA must be assessed to estimate failure probabilities. The methodology for performing this assessment is as follows:

- A model of each system is built that will compare the estimated failure pressure with the expected local system pressure for each important component.

- The pressure distribution in the affected system is estimated based on the expected initiating event, the initial primary system conditions, and the performance of relief valves.

- The probability of component failure is calculated based on the calculated pressure distribution, the median failure pressure, and the failure pressure variance of the affected components.

3.1.3.5 Operator Actions and Human Factors. Human reliability analysis (HRA) is sometimes an indispensable component of the ISLOCA risk model. Given that human errors are postulated to be an important influence on risk, HRA can be used to estimate the error probabilities associated with maintenance and testing procedures, operations, and operator actions required to recover from the accident. The operator recovery actions are those of detection and diagnosis, system isolation, and mitigation of the release.

A series of HRA steps were identified and incorporated into the ISLOCA methodology. These steps are necessary if one is to apply HRA to the actions of operations and maintenance staff. These steps are as follows:

- Identify specific human actions which are expected to be significant contributors to ISLOCA risk.
- Develop a description of important human actions and associated key factors. The key factors include human failure modes, whether the failures involve active or latent errors, identification of errors of intention/execution, and review of performance shaping factors.
- Quantify the probabilities for the various human actions, determine sensitivities, and establish uncertainty ranges.

Review the HRA results for completeness and relevance.

3.2 ISLOCA Sequence Identification

3.2.1 Review of Plant Systems and Operations. The initial step in the ISLOCA evaluation is a preliminary, qualitative assessment of the potential for an ISLOCA. Hardware and operating information on a wide range of low- and high-pressure interfacing systems should be collected, as this information will form the foundation for the HRA. For example, the systems analysts will identify barriers to an ISLOCA and the human reliability analysts will decide upon means by which these barriers could be circumvented. The required information includes plant procedures, piping and instrumentation diagrams (P&IDs), isometric drawings, training manuals, etc. This information should be reviewed by the team of PRA and HRA specialists to familiarize them with the systems and operations that have the potential to initiate, prevent, or mitigate an ISLOCA. All systems that interface with the reactor coolant system (RCS) should be identified in this preliminary evaluation.

3.2.1.1 Information Needs. Information is required on all systems that interface with the primary system, such that exposure to full primary operating conditions would exceed the design pressure or temperature rating of the system. For a BWR these systems include, but might not be limited to

1. Core Spray (LPCS) system
2. High pressure coolant injection (HPCI) system
3. Reactor core isolation cooling (RCIC) system
4. Residual heat removal (RHR) system
5. Reactor water cleanup (RWCU) system
6. Control rod drive (CRD) system.

For a PWR, this list might include

1. Low pressure coolant injection
2. High pressure coolant injection
3. Component cooling water systems
4. Letdown and makeup systems
5. Decay heat removal systems
6. Chemical and volume control system (CVCS).

The information needed on all of these systems would include

- P&IDs
- Isometrics
- Plant layout showing the relevant equipment rooms
- Emergency Operating Procedures
- Piping specification including information on flanges and other components, such as types of bolts used and their torque settings for flanged connections (two possible sources of information are the original design specifications for the systems and the vendor packages for components of interest, particularly the RHR heat exchangers)
- System descriptions and training manuals
- Relief valve capacities and ratings
- Test and surveillance procedures on PIVs
- System test and surveillance procedures
- Fire Hazards Analysis Report.

3.2.2 Interface Review and Screening. The maximum interfacing system break size that will not result in core damage must be determined to aid where possible in screening out interfaces from a more detailed analysis. This determination should consider the potential leak rate, normal RCS makeup capacity, capacity of the makeup water source (e.g., refueling water storage tank), and the ability of the auxiliary building or secondary containment (into which the postulated rupture discharges) to accommodate the volume of water released. The interfacing systems are then screened and categorized according to these criteria. The systems that survive the screening are analyzed further to identify potential ISLOCA initiators and sequences. The identified sequences are developed in detail sufficient to guide a team of PRA and HRA specialists in obtaining detailed information during an extended plant visit.

3.2.2.1 ISLOCA Sequence Screening. The formulation of the ISLOCA sequences begins with an assessment of the initiating events for the plant under investigation. The number of possible ISLOCA sequences that can be developed from these initiating events is quite large. The ISLOCA sequences for some systems can and should be eliminated before developing the event trees. This simplifies the event trees and allows for the elimination of some small-diameter interfacing lines from consideration. A screening process is developed in this section to eliminate sequences which do not contribute significantly to core damage.

The system interfaces are screened and categorized in terms of break size, rupture probability, and the potential for containment bypass. Specifically, a particular interface can be eliminated from further consideration based on pipe size smaller than a specified minimum (hence limiting the size of the potential rupture), those systems that do not bypass the containment, and the probability of rupture being less than a specified value (i.e., the interfacing system is rated for a relatively high pressure). The systems that are not screened out are analyzed further to identify potential ISLOCA initiators and sequences. This system-level event tree screening process is described in detail in the following sections.

3.2.2.1.1 Interfacing System Data Collection—The first activity in screening the interfacing systems is to assimilate information on all systems interfacing with the RCS that are not rated, in all or in part, to withstand full RCS operating pressure. The interfacing system data is collated in terms of system, hardware, operations, and procedures. The system information must include data on the maintenance and testing of hardware, logic circuits, and power supplies. The required information was specified in Section 2.1.1.

The system components are further categorized in terms of likely break size and the potential for containment bypass. This categorization is followed by a walkdown of the interfacing systems, including interviews with cognizant personnel, tabletop exercises with procedures, and observations of control room practices and simulator training sessions. During a plant walkdown detailed information is collected on

- The hardware that could be involved in an ISLOCA (i.e., control valves, relief valves, piping, flanges, pumps, heat exchangers)
- The procedures and guidelines imposed on plant personnel during startup, normal power operation, and shutdown
- Maintenance and in-service test practices
- The factors that could influence plant personnel performance as it relates to initiation, detection, prevention, and mitigation of an ISLOCA.

The information obtained from the plant walkdown is required to verify, support, and refine the ISLOCA system model.

3.2.2.1.2 Break Size Screening Criterion—The break size screening criterion has two elements. The first element compares the flow rate out of the break to the design flow rate of the normal charging pumps. A sequence is included in the event tree if the break flow rate is larger than the normal charging pump flow rate. The second element is based on the makeup to the ECCS reservoir. A sequence is included if the makeup flow is less than the break flow.

3.2.2.1.3 Probability Screening Criteria—The interfacing systems can be screened using two probability-based criteria. The first criterion is the number of PIVs that separate the high- and low-pressure lines. The second criterion is related to the potential for failure of the low-pressure side of the interfacing system when it is exposed to RCS operating conditions.

Screening on the number of PIVs is based on the premise that as the number of PIVs increases (for a single interfacing line) the probability of overpressurizing the interfacing system decreases. For the most part this is a valid assumption. Possible exceptions occur when considering common cause or common mode failures (both hardware faults and human actions), and when considering periodic normal operations that open any of the PIVs.

The interfacing systems are also screened based on the potential for the system to rupture given RCS operating pressure as the driving force. The procedure outlined in Section 4 of this report is used to determine the failure probability for the system components. One possible criterion would be to eliminate a system from further consideration if its rupture probability is less than 10^{-3} at the nominal ISLOCA sequence pressure. This probability limit is a product of the methodology used to generate the pressure fragility estimates for pipes (see References 8 and 9). In that work, a 10^{-3} probability is assumed for the presence of a flaw in the pipe such that the pipe ruptures at its yield pressure. Calculating a rupture probability less than 10^{-3} means the pipe is postulated to fail without yield stresses being produced in the material, a questionable assertion. In addition, this value combined with a generic ISLOCA initiator bounding frequency of 10^{-3} per reactor year and a generic recovery bounding probability of 0.1, produces an ISLOCA bounding frequency of 10^{-7} per reactor year.

The interfacing systems are designed with a large factor of safety. Therefore, they may not fail even if pressure is elevated to that of the RCS. This robustness has been observed in several reactor incidents in which RCS pressure was inadvertently applied to an interfacing system. Based on the potential for the interfacing system to survive the overpressurization, a screening criterion is useful for eliminating these systems from further consideration.

3.3 ISLOCA Sequence Event Tree Development

3.3.1 Introduction. The ISLOCA event trees comprise three phases:

- **Initiating events** are those combinations of hardware and human failures that result in a breach of the pressure isolation boundary and allow high-pressure RCS water to enter the lower pressure interfacing system.

- **The rupture event** models the conditional probability of interfacing system rupture and includes both size and location.

- **The postrupture events** identify the actions and estimate the likelihood that control room operators can successfully recover from an ISLOCA or mitigate its consequences.

The ISLOCA sequences are modeled using component-level event trees comprising the hardware faults and human errors that make up each sequence. In some cases, fault trees are also developed to support the quantification of the event tree top events. The fault trees include both hardware and human error contributions to top event failure probability. The event tree is quantified after all ISLOCA event trees have been developed and potential sequences have been identified.

This section of the report describes the formulation of the component-level event trees and the supporting fault trees. The methodology employed in this document makes use of simplified event tree and fault trees models as the mechanism for quantifying ISLOCA risk.

3.3.2 ISLOCA Event Tree Formulation. The ISLOCA event tree formulation is based on previous analyses performed for three PWR plants.[1,2,3] The simplified event tree is shown in Figure 2. This event tree has three top-event groups listed as ISLOCA initiation, interfacing system rupture, and recovery events. The selection of these top events results in a simplified template that can be used to assess ISLOCA risk. As shown in Figure 2 the event tree template has only eight end states. The event tree is structured based on the up branch representing the success event and the down branch modeling the failure event (i.e., the "up-is-good-and-down-is-bad" system).

The event tree end states are defined by the combinations of the top events. Four of the end states do not result in core damage. End state 1 is a sequence in which the PIVs do not fail and the interfacing system is not pressurized. The second end state represents a sequence in which one of two PIVs fails while the other does not; hence, the

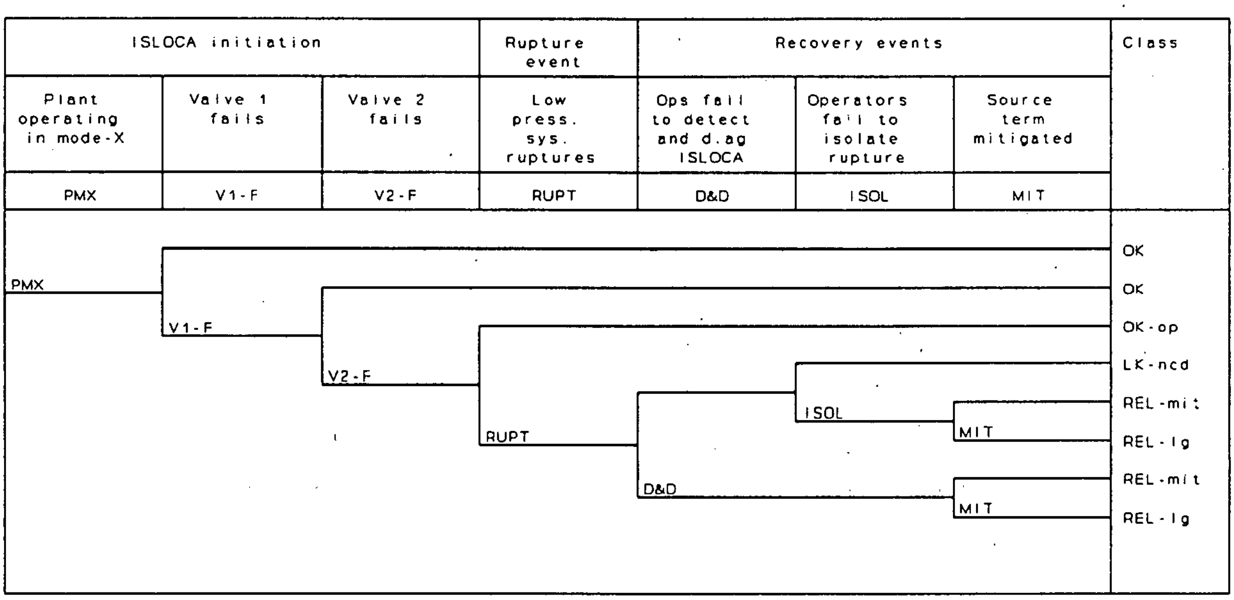

Figure 2. Generic ISLOCA event tree.

interfacing system is not pressurized. In end state 3, both PIVs fail and the interfacing system is pressurized but does not rupture. End state 4 portrays a situation where the PIVs fail and the interfacing system ruptures, but the rupture is identified and isolated before core damage can occur. The last four end states are core damage sequences comprising a rupture of the interfacing system and a failure to isolate the rupture. In two of these the resulting radioactive release is mitigated either through some type of accident management strategy or as a consequence of events occurring in the auxiliary building or secondary containment. For example, the rupture might be flooded at the time of the release or it might be possible to create a filter in the release path (e.g., actuate fire suppression sprays).

3.3.3 ISLOCA Initiation. All relevant aspects of plant operations should be examined in the process of identifying possible ISLOCA initiation scenarios. Particularly, activities that involve the PIVs should be reviewed carefully to assess the potential for reducing the reliability of the pressure isolation boundary. These activities include plant mode changes, PIV stroke-testing, testing actuation logic, and valve maintenance. Various combinations of hardware faults and human errors by themselves and in conjunction with the above activities, including administrative work controls, should be reviewed with respect to being possible initiators of ISLOCA sequences.

3.3.3.1 PIV Failure Mechanisms. The first failure examined in the ISLOCA event tree is the inboard PIV. The type of valve being used (check valve, motor-operated valve, etc.) determines the possible failure mechanisms that are included. Motor-operated valves are susceptible to spurious operation as a result of either hardware failure or human error, while check valves are vulnerable to hardware faults, design errors, and maintenance errors. The dominant issue in estimating check valve failure likelihood is operational history. For example, is the valve normally closed, normally free, and is it leak-tested or inspected and how often?

Actions that should be included in estimating motor-operated valve failure rates include logic- and stroke-testing, operator actions, and valve interlocks and controls. HRA is useful for estimating the potential for these activities to cause PIV failure. The details of the HRA are described in Section 3.5 of this report. The HRA should consider the following in estimating the probability of human-caused PIV failure:

- Operations crew bypassing the PIV interlocks
- Operator failure to follow procedures to close PIVs
- Inadvertent opening of PIVs by the operations crew
- Mispositioning of the valves due to human error.

Hardware failure analysis of the PIVs should consider the following events:

- **Internal Rupture/Leakage.** This item is included only if the leakage rate is larger than the normal makeup capacity or the makeup to the ECCS reservoir.
- **Failure to Hold.** This failure occurs when the valve is suddenly pressurized. The pressurization is caused by the sudden rupture of the upstream valve.
- **Mispositioned.** This failure is caused by mechanical failure of the valve position indicators.
- **Spurious Failure.** This failure is caused by failure of the valve mechanical and electrical control circuits.
- **Failure to Reseat.** The valve is stuck open.

The data and information required to estimate these probabilities can be found in various PRA data bases. However, when possible, plant-specific data should be used. The effects of maintenance on failure probabilities can be estimated from plant-specific maintenance procedures.

3.4 Interfacing System Rupture Probability

3.4.1 Introduction. The probability that the interfacing system will fail upon overpressurization is required to quantify the ISLOCA event tree. This failure probability is a function of component design. Components requiring a failure analysis include valves, flanges, pipes, heat exchangers, and tanks.

The methodology for performing the hardware component failure assessment consists of three steps. These three steps are as follows:

- A model of each system is built that will compare the estimated failure pressure with the expected local system pressure for each important component.
- The pressure distribution in the affected system is estimated based on (a) the expected initiating event, (b) the initial primary system conditions, and (c) the expected performance of relief valves.
- The probability of component failure is calculated based on the calculated pressure distribution, the median failure pressure and the failure pressure variance of the affected components.

Estimation of the interfacing system pressure distribution should be considered carefully. The primary system pressure and temperature can be used as a basis for calculating a bounding estimate of the failure probability of the interfacing system components. If more accurate calculations are desired, the pressure reduction effects of the interfacing system relief valves can also be included. The relief valves reduce local system pressure for components downstream of the relief valves.

Detailed calculations of the response of the interfacing system to a sudden pressurization indicate that the system pressure rapidly approaches that of the primary system.[1] It is thus recommended that the primary system pressure and temperature be used as simplified boundary conditions in estimating the rupture probability of the interfacing system.

Two parameters are required to estimate the failure probability of the pressure component: the median failure pressure and the logarithmic standard deviation of this pressure. The following sections provide recommended guidelines for selecting these two parameters.

There are several ways in which the failure probability of the interfacing system can be estimated. The approach utilized in the more detailed assessments makes use of Monte-Carlo techniques.[1] In this method the probability of system failure is determined by randomly selecting a failure pressure from the failure pressure distribution of the appropriate component and comparing the selected component failure pressure with a randomly selected system pressure. The system pressure is 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 is assumed to have failed. Otherwise no failure is assumed. Each component in the low-pressure system is evaluated in this manner until all components have been examined. This process is repeated in a Monte Carlo simulation until the variance is acceptable. Once the simulation is completed, the output is binned and the relative frequency of various equipment failures can be estimated.

3.4.2 Rupture Probability Estimation. The component pressure fragility is modeled with a lognormal distribution, which is a valid description of the variation in material properties.[8]

A component failure probability can be estimated as a function of the applied system pressure. The calculation is analogous to that performed for seismic failure of structural components. The component failure probability as a function of the internal pressure is

$$Prob(P_f \le P_i) = \Phi\{[\ln(P_i) - \ln(P_f')]/\beta\} \quad (1)$$

where

P_f	=	failure pressure
P_i	=	local internal pressure
P_f'	=	estimated median failure pressure
β	=	logarithmic standard deviation of P_f [s.d. of the corresponding normal distribution of $\ln(P_f)$]
$\Phi()$	=	standard cumulative normal (Gaussian) function
ln()	=	natural logarithm.

3.4.3 Gasketed-Flange Connections. The systems interfacing with the RCS may contain a number of gasketed-flange connections. Knowledge of how these connections can fail is important to understanding the response of interfacing systems to overpressurization. The failure modes allow RCS water to either spray or jet from the flange. Depending on the leak size, these sprays and jets can inhibit operator actions in the vicinity of the leak and flood the compartment in which the leak is located. It is recommended that the gasketed-flange connections be evaluated on a plant-specific basis. Such an evaluation is required because the flanges are installed using a wide variety of different types of bolts and studs. These bolts and studs have a correspondingly wide range of prestresses and applied torques.

Gasketed-flange connections are used with flow-restricting elements, flow-measuring devices, and some major equipment, including motor-, air-, and manually operated valves and relief valves. The valve bodies also have flanged connections as part of the opening mechanism. One additional and potentially important application of a flanged connection is the tube sheet in a heat exchanger. Typically these are very large flanges (e.g., 44 in. in diameter) and have been found to be among the weakest points in the RHR heat exchangers evaluated in previous analyses.[1,2,3] The failure mechanisms and failure areas of the gasketed-flange connections are important in evaluating the progression of an ISLOCA because they influence the time to core uncovery, the operator actions required to recover from or mitigate the accident, and other human factor aspects of the accident.

3.4.3.1 Variables Affecting Flanged-Joint Leakage Behavior. There are several factors that influence the pressure at which flange leakage begins. And leakage behavior tends to be complex. This complex behavior results from a combination of variations in pressure and temperature and the flange's previous loading history. Predicting the leakage behavior of an overpressurized gasketed flange requires information on the following variables:

- Bolt/stud preload
- Bolt/stud temperature
- Bolt/stud material (yield strength)
- Bolt/stud stress-strain relationship
- Bolt relaxation
- Flange flexibility
- Initial gasket stress
- Gasket loading stiffness
- Gasket unloading/reloading stiffness
- Gasket creep and relaxation
- Pipe bending moments.

3.4.3.2 Flanged Joint Behavior. The behavior of the flanged joint under pressure loading is a function of the service history of the joint, as well as its design. Service history refers to joint assembly, total applied internal pressure, bolt torque, etc. The behavior of gasketed-flange connections subjected to the increasing pressure load associated with an ISLOCA is determined by the following attributes.

Preaccident flanged-joint conditions:

- During installation, the flange bolts are torqued and retightened to develop an initial gasket stress. This results in a lockup condition between the flange and the gasket.

- The flange gaskets experience cyclic creep and relaxation over the course of normal plant operation. The relaxation reduces the gasket stress with an accompanying increase in the lockup stress but a negligible change in bolt stress.

- The bolts that are prestressed beyond the material yield stress relax during operation. This reduces the bolt stress with a corresponding reduction in the lockup stress and a possible reduction in the gasket stress if the lockup stress was small.

ISLOCA flanged-joint conditions:

- During ISLOCA initiation, the rising pressure must overcome any lockup loads before leaks can develop. This does not require any reduction in the gasket stress or increase in bolt stress.

- As pressure increases to the gross leak pressure (see Section 3.4.3.3 below for definition of gross leak pressure), the pressure increase is shared by the gasket and bolts, decreasing the gasket stress and increasing bolt stress. If the bolt yield stress is reached at a pressure less than the gross leak pressure, 97% of the pressure load above the bolt yield pressure contributes to a reduction of the gasket stress, while the remaining 3% contributes to an increase in the bolt stress.

- Increases in pressure beyond the gross leak pressure increase the bolt stress, with accompanying increases in bolt length, in accordance with the bolt stress-strain diagram, up to the bolt failure strain.

3.4.3.3 Estimating Leak Rate and Leak Area. This section discusses how to estimate leak rates from overpressurized gasketed-flange connections. Gross leakage from these connections begins when the gasket stress is equal to the pressure being retained by the gasket. This point is defined as the *gross leak pressure*. This definition was derived from experiments in which O-rings and flat-face gaskets suffered blowout. Spiral-wound gaskets are not likely to be on the verge of catastrophic failure when gasket stress is reduced to the pressure being retained, but the potential for such failure does exist.

The mass leak rate at pressures less than the gross leak pressure is estimated from correlations of the results of gasket leakage tests using water. This low-pressure gasket leakage is caused by seams and crevices in the flange/seal joint. The total water mass leak rate case is computed as

$$W_g = 2.87 \times 10^{-3} \frac{D_o + D_i}{t_g \; T_p} \tag{2}$$

with

$$T_p = \exp\left\{6.19 - 6.24[9.21 - \ln(SG_o)] - 0.79 \ln\left(\frac{SG_o}{SG}\right)\right\} \tag{3}$$

where

T_p	=	tightness parameter
D_o	=	gasket outside diameter in inches
D_i	=	gasket inside diameter in inches
t_g	=	gasket thickness in inches
SG_o	=	initial gasket stress in psi
SG	=	current gasket stress in psi
W_g	=	leak rate in mg/s.

A correction factor must be introduced to account for variations in the effective gasket width. The calculation of gasket width and gasket area should not include the outer 1/8 in. of the gasket as this outer region is not effective in the sealing process.

A gasket leak rate of a few milligrams per second should not affect the ability of operators to perform actions in the vicinity of the leak. However, the ability of an operations crew to isolate leaks in the interfacing systems can be impaired by larger leaks. The types of problems that could be encountered can be better understood if a drop of water is idealized as a 1/8-in.-diameter solid sphere. A leak rate of 1 mg/s would correspond to 3.5 drops per minute or about one drop per 17 s. If the leak rate increased to 20 mg/s, the joint leakage would be about 1 drop per second, still not enough to impair operator response. However, if the leak rate increased to 200 to 500 mg/s, a spray of water would form, which could inhibit some operator actions in the vicinity of the leak.

Gross gasket leakage is caused by separation of the flange from the gasket at internal pressures beyond the gross leak pressure. The leak area is calculated as the mean gasket perimeter times the separation distance. The separation distance is affected by bolt extension, gasket recovery, and flange flexibility. Of these parameters bolt extension is the dominant contributor to the leak area. Thus, for simplicity, the calculation of separation distance includes only the effect of bolt extension. Note that excluding the effect of gasket recovery from the leak area is conservative and leads to a slight overestimation of leak area.

The leak area at pressures beyond the gross leak pressure is estimated using Equations (4) and (5). These two equations represent cases where (a) the bolt stress is less than or equal to the bolt material yield strength and (b) the bolt stress exceeds the material yield strength. For bolt stresses less than the yield strength

$$A_l = \frac{\pi}{2}(D_o + D_i)\frac{L_b(P - P_g)}{N_b\ E_b}\frac{A_p}{A_b} \quad . (4)$$

For bolt stresses greater than the yield strength

$$A_l = \frac{\pi}{2}(D_o + D_i)\ L_b\left[\frac{(S_{by} - S_{ba})}{E_b} + \frac{(S_{bo} - S_{by}) + \dfrac{(P - P_g)A_p}{N_b\ A_b}}{E_y}\right] \quad (5)$$

where

P	=	local pressure in psi
L_b	=	bolt/stud length in inches
P_g	=	gross leak pressure in psi
A_p	=	area based on gasket inside diameter
N_b	=	number of flange bolts
A_b	=	bolt tensile stress area per bolt
E_b	=	bolt material elastic modulus
S_{by}	=	bolt material yield stress
S_{ba}	=	actual bolt stress (psi)
	=	(1–JR/100)S_{bo} for no lockup case
	=	S_{bo} for lockup case
JR	=	joint relaxation (JR) expressed in percent of SG_o
S_{bo}	=	initial bolt stress
E_y	=	inelastic modulus for the bolt's stress-strain diagram.

To be more accurate, these equations should contain a term describing the recoverable gasket deflection at pressures beyond the gross leak pressure. However, to be simple yet conservative, this term has been neglected. Omission of this term is reasonable because there are problems associated with obtaining an accurate estimate of the term. Also, bolt extension length exceeds gasket recovery length.

3.4.3.4 Pressure-Induced Leakage and Failure Probabilities. The leakage potential of gasketed-flange connections should be evaluated on a plant-specific basis. This requirement is a result of the diversity of bolts and studs used to secure the flanges. For example, the flanges may be secured with bolts or studs of different materials with different bolt preloads. The bolts or studs may be classified in two fairly broad categories: low-strength and high-strength. Low-strength or "soft" bolts are much more likely to yield at relatively low internal pressures than are high-strength or "hard" bolts. The plant-specific flange evaluation should include an assessment of the bolt material and installation torque range.

Tabular values of the gross leak pressures, leak rates, and leak areas for various flange ratings and pipe sizes have been developed.[9] These tables can be used in assessing the fragility of gasketed-flange connections. Two sets of tables corresponding to low- and high-strength bolts are presented in Reference 9. The recommended procedure requires a determination of the expected initial bolt prestress and joint relaxation for a given pipe diameter and flange rating. To account for relaxation and cyclic creep in those cases where lockup between the flange and the compression ring does not occur (normally all 150-psi flanges and 300- and 600-psi flanges with low bolt stress and high gasket loading stiffness) a median joint relaxation of 25% is recommended, with the range from 0 to 25% chosen to represent a 95% (2σ) variation. Once the required parameters have been determined, the gross leak pressure as well as the leak rate or leak area can be obtained from the tables for discrete values over the range of 0.25 to 2.0 times the gross leak pressure. The results for low-strength bolts are given in Tables 4-5 to 4-29 and for high-strength bolts in Tables 4-30 to 4-54 of Reference 9.

The variabilities expected for bolted-flange leak rates are shown in Table 4-55 of Reference 9. A leak area variability of 0.12 is recommended for all pipe diameters and flange ratings.

3.4.4 Piping. Only stainless steel piping has been employed in all of the interfacing systems examined in the course of the ISLOCA Research Program. However, carbon steel piping is included here for completeness. Piping is considered to be a generic component. This is in contrast to pressure vessels, which are often designed specifically for a given plant. Tabulated failure pressures are presented in Reference 9 for use in estimating the probability of failure of the interfacing system piping. These tables were developed for both Type 304 stainless and SA 106 Grade B carbon steel piping.

These tables list median failure pressures tabulated for different pipe sizes and schedules, and for a temperature range of 70 to 800°F. The tables also list three values for corrosion allowance. However, these tables do not consider the effects of erosion, which can normally be neglected because most interfacing systems are not exposed to continuous flow. Linear interpolation may be used with the above tables to establish a pressure capacity for any desired intermediate corrosion allowance.

Type 316 stainless steel is stronger than Type 304 at elevated temperatures. For Type 316 stainless pipe, the values for Type 304 stainless in the tables of Reference 9 should be increased by 10% for temperatures above 400°F. Since failure pressure is dependent on both failure strain and stress, care should be taken in using the tables with other materials.

The pipe failure probability calculations must include consideration for (a) scatter in the strength and uniaxial elongation material properties, (b) the variation in material properties with temperature, (c) the biaxial strain and gauge-to-length ratio effects on failure strain, (d) thermal bending strain, (e) branch connections, (f) flanges, and (g) the possible existence of partial through-wall cracks. These effects can be expressed in terms of a logarithmic standard deviation for both stainless and carbon steel piping. The values recommended for ISLOCA assessments are shown in the tables of Reference 9, which include the median failure stresses used in estimating the tabulated median failure pressures.

3.4.5 Pumps. Based on the experience gained in performing three plant-specific ISLOCA analyses,[1,2,3] pumps represent the most difficult component for which to draw generalizations. This results from the wide range of pump types supplied by different vendors that are being used in commercial nuclear plant systems. These different pumps comprise a wide variety of specific design aspects including cases, bolted flange connections, mechanical shaft seals, and occasionally seal-water cooling tubes and other similar pump-specific features.

The vendor-supplied hydrostatic test pressures for the pump casing may be used as a lower bound estimate of the case failure pressure. However, the hydrostatic test often does not include the pump seal and shaft assemblies, with a dummy plate being used to replace the seal assembly in those hydrostatic tests in which the seal assembly is not tested. Separate vendor test information is sometimes available on the seals. The seal face loading springs and elements are often amenable to analysis using conventional strength-of-materials methods. The leak areas based on calculated seal distortions may be estimated to determine the leak flow rates.

3.4.6 Tanks and Heat Exchangers. Two approaches are presented in this section to estimate the failure probability of tanks and heat exchangers in the interfacing system.[8] The first approach takes into consideration the factor of safety, the design pressure, and the variability in the failure pressure. This approach reflects the range of results obtained from previous ISLOCA evaluations.[1,2,3] The second approach applies simple strength-of-materials analysis to the cylinders and heads of the tanks and heat exchangers.

Either approach can be used for both stainless and carbon steel vessels. The recommended factors of safety are based on the vessel design pressures. The components considered in developing the methodology were either mounted vertically or saddle-mounted and were fabricated from Type 304 or Type 316 stainless steel or from carbon steel with properties up to SA 516 Grade 70.

The two approaches are presented for both cylindrical hoop failure and failure of the dished heads. Cylindrical hoop failure governs the probability of failure at lower metal temperatures, while buckling of semi-ellipsoidal or torispherical heads dominates at higher temperatures.

3.4.6.1 Factor of Safety on Design Pressure. The structural considerations that control the failure probability include hoop failure in the vessel cylinder and plastic collapse of the head due to high internal pressure. As mentioned above, cylindrical hoop failure is expected to control the failure probability at low metal temperatures. The median factors of safety over design pressure for hoop failure at room temperature can be expected to range from about 4 to over 15. A median factor of safety of about 6.5, together with a logarithmic standard deviation of 0.45, is recommended in screening for cylindrical hoop failure at room temperature. At increased temperature, a decrease in the median factor of safety of about 20% for stainless steel tanks, together with an increase in the logarithmic standard deviation to about 0.66 at 600°F, is recommended. The increase in the logarithmic standard deviation reflects the increasing uncertainty in material properties, as well as the uncertainty regarding increased thermal nozzle loads in piping systems designed for cold service. No reduction in the median factor of safety for carbon steel tanks is required for temperatures up to 600°F. The same logarithmic standard deviation is recommended for carbon steel as for stainless.

The plastic collapse of the head controls the failure of typical tank designs at 400°F or higher. However, buckling of the head does not necessarily lead to formation of a crack. Both asymmetric and plastic collapse failure modes exist for dished heads subjected to high internal pressures. For steel heads, plastic collapse will govern. A conditional probability of crack formation of 0.20 is recommended, given buckling of the head. It is also recommended that a median factor of safety of 1.6 (compared to design pressure at room temperature), with a logarithmic standard deviation of 0.50, be used for dished head buckling at 600°F. A summary of expected factors of safety above design pressure for steel tanks is shown in the tables of Reference 9.

The recommended logarithmic standard deviations for the median failure pressures are large. These standard deviations are intended to provide a conservative estimate of the tank or vessel failure probability, which can be obtained without a detailed analysis. The estimate can be made based on the design pressure and the knowledge that the design temperature is 300°F or less. As an example, a probability of failure of less than 0.001 would be associated with a room temperature hydrostatic test at 150% of the design pressure. This probability of failure may be too conservative, but it allows for corrosion and mechanical or thermal fatigue cracks formed since the original hydrostatic test.

3.4.6.2 Tank Analysis. Analytical methods are available that can provide a more accurate estimate of the failure probability. This section presents the recommended formulae. Also included are recommended material properties and logarithmic standard deviations for failure pressures. The material properties are considered to be median-centered. The logarithmic standard deviation is associated with failure pressure, tank modeling uncertainty, the possibility of partial through-wall cracks, and nozzle loads from thermal strains in the attached piping.

3.4.6.2.1 Cylindrical Hoop Failure—The median failure pressure in a cylindrical vessel subjected to hoop stress can be calculated as

$$P_f = \frac{\sigma_f \, t}{r(1 + \varepsilon_f)} \tag{6}$$

where

P_f = median failure pressure

σ_f = median failure stress

t = nominal wall thickness

r = initial inside radius

ε_f = median hoop strain at failure.

The thickness t may include some provision for corrosion if applicable. Values of σ_f and ε_f for two representative tank materials at discrete temperatures from room temperature to 800°F can be found in Reference 9.

3.4.6.2.2 Buckling of Dished Heads—Buckling of dished heads subjected to high internal pressure can be calculated based on the analysis conducted by Galletly and his co-workers,[10,11] which used the BOSOR-5 computer code.[12] The methods recommended here use the results of Galletly, modified to provide appropriate median failure pressures, together with estimates of the logarithmic standard deviation, based on limited test results. The median plastic collapse capacity (P_o) for dished heads may be calculated from

$$P_o = 1.78 \, \frac{\sigma_y(1 + 50\varepsilon_y)}{r} \tag{7}$$

for 2:1 semi-ellipsoidal steel heads, and

$$P_o = \frac{22.4\sigma_y(1 + 240\varepsilon_y)(r_t/2r)^{1.04}}{(2r/t)^{1.09} \, (R_s/2r)^{0.79}} \tag{8}$$

for torispherical heads,

where

σ_y = yield stress

t = head thickness

r = radius of the attached cylinder

ε_y = yield point strain

r_t = torroidal radius

R_s = radius of the spherical portion.

These expressions are valid for most tanks and vessels with dished heads found in nuclear power plants. The validity of the above two equations covers the parameter ranges listed below:

Semi-ellipsoidal Heads

200 < r/t < 750

30 ksi < σ_y *< 60 ksi*

with a strain hardening slope of 0.5 and 10%, and

Torispherical Heads

250 < r/t < 750

1.5 < R_s/r *< 3*

$0.12 < r_t/r < 0.36$

$20\ ksi < \sigma_y < 75\ ksi$

with a strain hardening slope of 0.5 and 5%. Again, a conditional probability of crack formation of about 0.2 is assumed, given head collapse. Median material properties for typical nuclear power plant applications are shown in Reference 9. Plastic collapse has been shown to dominate in steel heads. Should another material be encountered, asymmetrical buckling should be evaluated also.

The variabilities associated with both cylinder hoop failure and dished head buckling are shown in the tables of Reference 9. Separate variabilities for stainless steel and carbon steel are given. These values are considered representative for most tanks, vessels, and heat exchangers. However, for unusual configurations, such as thin-wall vessels with large thickness tolerances, larger variabilities may be needed. Except for extremely short cylindrical vessels or other unique configurations, leak areas associated with either cylindrical hoop failure or dished head failure should be considered as large, uncontrolled leaks in the context of the ISLOCA.

The calculations of this section are not applicable to flat-bottomed tanks designed for atmospheric pressure. Any such tanks identified as susceptible to ISLOCA should be evaluated separately. Potential failure modes evaluated for such tanks should include hoop failure in the cylinder, membrane hinging in the cylinder at the base or dome ring girder, dome membrane failure, failure of the anchor system, and unseating of manways.

Tube sheets in heat exchangers may sometimes be evaluated using failure pressures listed in Reference 9 for bolted flanges with flexitallic or equivalent gaskets. However, tube sheets often involve nonstandard flange designs with dimensions and number of bolts, sizes, and bolt installation torques outside the range of applicability of the gasketed-flange tables. Such cases should be evaluated on an individual basis using the analytical techniques presented in Section 3.4.3 on gasketed flange connections.

Bolted flange connections, such as manways or other access hatches involving elastomeric seals, should also be evaluated using other techniques. Silicone rubber or ethylene propylene O-rings can usually be relied upon to provide a seal at temperatures up to 700°F in a steam environment, even though the elastomer may be severely damaged. This is true provided flange separation does not occur. If flange separation occurs, extrusion of the elastomer followed by leakage may occur at low bolt stresses. Also, if the elastomer is exposed to accident temperatures for a significant period of time, compression set may occur such that much of the rebound of the seal is lost. Very small increases in pressure may then cause leakage in cases where relatively long high-temperature exposure times occur, even if the seal is not extruded. Evaluation of elastomeric seals involves consideration of the bolt clamping force, bolt length, type of elastomer, and time and temperature at normal operating conditions and accident conditions. Such evaluations should be based on the specific configuration under consideration.

3.4.7 Valves. Valves used in the interfacing systems of the RCS typically have three failure modes. These are: failure of the valve body, failure of the valve stem packing, and failure of the bolted bonnet. Two of these failure modes can be eliminated by simple arguments based on the relative thickness of the valve body and the type of valve stem packing. Valve body failure is eliminated because the adjacent piping should fail before the valve body, since the thickness of the valve body is typically greater than that of the adjacent piping in most interfacing systems. Elimination of the valve stem leakage failure mode is possible because the valve stem packing used in nuclear plants tends to compress under pressure, providing greater resistance to leakage. It is possible that the stem packing for some valves could deteriorate under normal service conditions. It was judged that the leak rate resulting from such packing deterioration would be quite small and would have a negligible effect on valve and system operation. Thus, the only credible pressure-induced failure mode for the interfacing system valves is failure of the bolted-bonnet seal.

Bolted-bonnet valves are typically sealed using Style R spiral-wound gaskets compressed between the bonnet and the valve body. The bonnet and valve body are generally machined in a tongue-and-groove configuration. Normally, the valves are fitted with high-strength bolts having tensile yield strengths of 100,000 psi or greater. The bonnet bolts are normally torqued to specified levels that prestress the bolt in the range of 35,000 to 45,000 psi. Preloading the bolt to the specified level produces a substantial lockup force between the bonnet and the valve body. Frequently a seal weld is applied to the lip at this junction, also. The bolted-bonnet valves should be analyzed in a manner identical to that used for the gasketed-flange connections, with no credit taken for the seal weld.

3.5 Operator Actions and Human Reliability Analysis

3.5.1 Introduction. This section provides recommendations for quantifying the human errors that contribute to ISLOCA risk. These recommendations are incorporated into a simplified HRA.

Human reliability has been shown to be an important aspect of the risk associated with an ISLOCA.[1,2,3] Human errors are associated with (a) maintenance and testing procedures, (b) operations, and (c) operator actions required to recover from the accident. The operator recovery actions are associated with (a) detection and diagnosis, (b) isolation, and (c) mitigation. HRA is used to estimate these error probabilities.

The ISLOCA HRA methodology consists of a series of steps. These steps are useful for understanding the actions of operations and maintenance staff. These steps are as follows:

- Identify specific human actions which are expected to be significant contributors to ISLOCA risk. This step is driven by the systems analysis, which identifies the pressure isolation barriers.
- Develop a description of important human actions and associated key factors. The key factors include (a) human failure modes, (b) whether the failures involve active or latent errors, (c) identification of errors of intention/execution, and (d) review of performance shaping factors.
- Estimate the error probabilities for the various human actions, determine sensitivities, and establish uncertainty ranges.
- Review the HRA results for completeness and relevance.

The HRA steps can be used in an iterative scheme to refine the analysis. This iteration can be continued until the HRA results converge to a specified tolerance.

The human error probabilities, or HEPs, for maintenance and testing should be compared with plant data on errors that have occurred during maintenance and testing. This comparison provides a means to validate the results of this portion of the HRA. Validation of the human error analysis against known information is an important aspect of the analysis. Human error models are subjective and require validation for each specific application in order to provide a reliable result. Development of a simplified HRA method for the ISLOCA analysis was a difficult task because HRA is not supported by widely accepted theories and empirical data bases.

A detailed HRA analysis is recommended only for those sequences that dominate ISLOCA risk. Thus, detailed HRA information is needed only for human actions (including actions that could initiate an ISLOCA) associated with *candidate* dominant sequences. All other human actions are assigned screening HEPs of 0.5 (1.0 in certain cases where an action obviously is impossible, e.g., isolating an interfacing system before it is overpressurized when the rate of pressurization is very high).

3.5.1.1 ISLOCA Recovery Actions. The detection, diagnosis, and isolation of an ISLOCA are guided by emergency procedures, which are in turn based upon Emergency Procedure Guidelines (EPGs) developed by the four nuclear steam supply system owners groups. (Note: Not all vendors refer to their guidelines as EPGs. This term has been selected for generic use in this report to avoid unnecessary complications.) These procedures rely on instruments and alarms to detect the ISLOCA. The instruments, alarms, and sensors include (a) system pressures and temperatures, (b) valve position indicators, (c) valve bypass alarms, (d) reactor water level, (e) reactor water clean-up high differential flow (BWR only), and (f) HPCI, RCIC, and main steam line flows (BWR only). The secondary containment conditions monitored include (a) temperatures, (b) water levels, and (c) radiation levels. The EPGs, with the exception of those developed by Westinghouse, do not address ISLOCA specifically. The BWR EPGs, developed by General Electric, instruct the operations crew to monitor not only the reactor vessel pressure and water level but also the conditions in the secondary containment. The symptom-based Westinghouse and General Electric EPGs provide guidelines for (a) isolation of pathways leading to leakage of primary system coolant to the secondary containment, (b) restoration of water level in the reactor vessel, (c) emergency depressurization (BWR only), and (d) reactor scram. These symptom-based procedures should allow detection, diagnosis, and isolation of ISLOCA breaks outside of the primary containment.

The B&W Owners Group[13] EPGs provide no specific guidance for ISLOCA sequences, particularly in regard to operator response. However, some individual B&W plants have developed emergency procedures that address ISLOCA sequences in varying degrees. For example, the Arkansas Nuclear One, Unit 1 leak detection procedures address loss of RCS inventory to either the containment or auxiliary building and identify interfacing systems that could be potential leak paths. However, no explicit instructions are given in the emergency procedures for dealing with ISLOCA sequences.

The Combustion Engineering EPGs provide actions to support long-term core cooling, some of which might be taken in responding to an ISLOCA. Examples are break isolation and use of safety injection systems. However, the EPGs do not make use of the symptoms present in the auxiliary building to diagnose an ISLOCA.

The emergency procedures provide the framework for estimating probabilities of detecting, diagnosing, and isolating an ISLOCA. Computation of these probabilities relies on a human reliability assessment of the procedures. The calculation of the probability of detecting, diagnosing, and isolating an ISLOCA must include an HRA and be coupled with an understanding of the environmental conditions present in the secondary containment and the time available before core damage occurs. The secondary containment environmental conditions must be known to estimate what equipment is operational in the secondary containment and to determine the ability of the operations crew to enter this environment and operate the equipment if required. The time available before core damage occurs is the time period the operators have to detect, diagnose, isolate, and mitigate the accident.

The environmental conditions in the secondary containment or auxiliary building influence the options the operators have in isolating an ISLOCA. The key parameters to consider are the temperature, relative humidity, and amount of flooding. Previous calculations of the effects of large and small breaks on PWR auxiliary buildings indicate that the temperature in the building can approach 100°C, and the relative humidity can approach 100%.[1] Since these conditions may be established within minutes of the break, operator actions requiring entry into the auxiliary building may not be possible. Also, remotely operated electrical equipment may fail within minutes of the break if not qualified for this type of environment. These maximum conditions should be used as the environmental conditions in assessing equipment performance unless plant-specific calculations are available. Also, any equipment located in the compartment in which the leak occurs should be assumed to fail (i.e., inoperable and irreparable). The compartments that are susceptible to flooding should be inventoried for safety-related equipment that is vulnerable to flooding damage. Flooding conditions should be assessed to determine whether this equipment will be rendered unavailable by flooding. The remaining equipment can then be used in the HRA to estimate the probability of isolating the ISLOCA. If the resulting recovery probabilities are unacceptably low, detailed thermal hydraulic analyses may be necessary to support relaxed environmental conditions. Some additional guidelines for performing the supporting calculations are provided in Section 6 of this report.

The time available before core damage occurs is difficult and complex to estimate. The recommended approach is to estimate the time required to exhaust the ECCS coolant supply using the run-out flow rate of the ECCS pumps under consideration. This time can be extended by the time required for boil off to uncover the core after injection fails. It may also be possible to show that the operators will likely throttle injection flow to prolong the time before the ECCS coolant supply is exhausted. However, in the absence of equipment failures caused by the environmental conditions, the time available for operator actions will be long enough that the additional time to core damage that can be demonstrated with more refined time-to-core-uncovery calculations will not change the HEP estimate significantly. Of more importance is the need to account for the loss of coolant injection that may occur very early in the scenario if the plant is susceptible to steam propagation and to the flooding that can result from the break. This is discussed in more detail in Section 6.

3.5.1.2 ISLOCA Mitigation Actions. The radionuclide transport and environmental effects of an ISLOCA are influenced by natural processes[14] and by operator actions. In this assessment an ISLOCA is considered mitigated if the break is rapidly flooded or if operators take action to reduce ventilation flow or actuate fire sprays in the auxiliary building. If the break is not flooded, a knowledge-based HEP for mitigation is recommended, provided viable options are available to the operators for mitigating the release.

The operations crew can take actions to increase fission product deposition in the auxiliary building. The success of these actions is a function of both the auxiliary building environmental conditions and the understanding the operations crew has of fission product transport and deposition.

Radionuclide deposition in the auxiliary building occurs by several natural processes. The dominant processes include sedimentation, inertial impaction, steam condensation, thermophoresis, pool scrubbing, and vapor sorption.[14] In several of the ISLOCA sequences the break location will flood. This flooding traps fission products in the surrounding water pool. This pool-scrubbing phenomenon is very effective in reducing the environmental radionuclide release.

There are two cases of interest with respect to mitigation of the environmental source term. The first case is associated with unflooded breaks. In this case reducing the inflow and outflow of air through the building is desirable. This flow reduction allows more time for fission product deposition by natural processes. A typical auxiliary building will retain more than 75% of the fission products without any operator actions.[e] Operator actions to increase deposition might include securing the building ventilation system, closing personnel and loading dock doors, and actuating the fire suppression system. These activities can increase the decontamination factor by a factor of 10. A human factors assessment is required to estimate the potential for the operations crew to mitigate these dry releases.

The second ISLOCA case of interest pertains to flooded breaks, which tend to trap some of the fission products in the surrounding water pool. No operator actions are required to mitigate the fission product release from a flooded break.

e. R. E. Henry and M. N. Hutcherson, Evaluation of the Consequences of Containment Bypass Scenarios—Volume 2: BWR Results, NP6586-L Volume 2, November 1989 (EPRI Proprietary).

Since no operator actions are required an HRA is not necessary. The probability of failure to mitigate a flooded break is then set to zero in the ISLOCA event tree.

The BWR and PWR auxiliary building decontamination factors (DFs) for early ISLOCA source term mitigation range from 1 for unmitigated releases to 10 for mitigated releases. These values are recommended for simplified assessments. These DFs are typical values that can be used for scoping calculations. More refined analyses should be performed if it is desirable to estimate time-dependent radionuclide source terms for the entire transient.

The auxiliary building DF acts in conjunction with fission product deposition in the interfacing system. DFs for the interfacing systems have been measured as large as 30 in some of the EPRI LACE experiments. A DF of 10 for interfacing systems and the RCS is recommended for this analysis. Using the combinations of ISLOCA DFs, a total DF of 30 for the unmitigated release and 300 for the mitigated (e.g. flooded) case are recommended for the simplified ISLOCA consequence analysis.

3.5.2 Development of Human Action Sequences. Development of the postulated human actions required by the sequence definitions is a multistep process which uses information from plant walkdowns and reviews of control room configuration and procedures. This information is used in conjunction with the PRA hardware analysis to perform an HRA task analysis. This task analysis is then used to estimate the HEPs for the ISLOCA sequences.

3.5.2.1 Collection of Human Reliability Information. Collection of the task analysis and HRA data involves several activities. The data collection activities recommended to complete the ISLOCA HRA consist of the following steps:

- Observations of and interviews with control room and training personnel to assess level of ISLOCA awareness
- A review of plant history documentation related to valve testing, maintenance, etc.
- An inspection of the control room from a human factors standpoint
- A review of the interfacing system PIVs, their locations (e.g., can they be operated locally), and tests performed to ensure integrity.

A requisite part of the ISLOCA HRA data collection activities is the system walkdown. These walkdowns are used to inspect the control room instrumentation and displays, to inspect likely break locations, to qualitatively estimate break flow paths and identify features (such as auxiliary building fire sprays and blow-out panels) that can affect the auxiliary building DF, and to collect human factors data vital to the estimation of error probabilities for detection, diagnosis, and isolation.

A data collection form should be developed to aid in the HRA data collection effort. Such data collection forms have been used successfully in previous ISLOCA evaluations.[1,2,3] In these past uses, the data forms were filled out by HRA personnel during NRC inspection team visits. In some cases the data collection forms were completed after the plant inspection by the HRA team from working notes, interview data from both plant personnel and inspection team members, comparison of procedures to P&IDs, and follow-up telephone calls to plant personnel.

3.5.2.2 Recommended Approach to Data Classification. The HRA requires that human actions be classified into the groups given in Table 1. For the ISLOCA evaluations, human actions should be decomposed into the following:

- **Dominant failure mode(s)**—omission, simple commission, complex commission (decision-based errors)
- **Time dimension**—active and latent
- **Place within the PRA event sequence**—preinitiator, initiator, detection,

diagnosis, or recovery action (either isolation or mitigation activities)

- **Place within the plant's activity cycle**—maintenance in-service test, calibration, normal operations, off-normal operations or emergency operations.

Human performance is also influenced by the performance shaping factors described in Table 2. The performance shaping factors provide a reference for describing the nature of human error in nuclear power plants. The inclusion of these factors in the ISLOCA HRA is described in a later subsection.

3.5.2.3 Combination with Hardware Failure Rate Data. It is recommended that cognitive errors be treated as a subset of errors of commission in the ISLOCA assessments. Cognitive errors

Table 1. Human error classification scheme (preliminary human error taxonomy).

FAILURE MODE	
Omission:	Failure to perform a task action or step in a procedure.
Commission:	
Simple	Failure to perform a task properly.
Complex	Decision-making based.
ERROR ACTIVITY DIMENSION	
Latent:	Refers to an error with no immediate impact. For example, a valve lineup is improperly performed, yet the consequence of that action fails to impact the plant until a surveillance procedure is implemented. Latent errors may be present in procedures as well. For example, an error in an emergency procedure may go unnoticed until the crew calls upon that procedure.
Active:	Actions with immediate impact. For example, an instrumentation and control technician causing a spurious safety injection or an inadvertent reactor trip, or control room personnel misreading their instrumentation and manually tripping the reactor.
EVENT SEQUENCE ANALYSIS	
Preinitiator:	Latent errors occurring prior to the beginning of an event and that have a consequence at a later time.
Initiator:	A human action taken that causes an event to occur.
Postinitiator:	A series of events occurring after the initiating event.
Detection:	Observing that some abnormal condition has occurred.
Diagnostic:	Proper recognition by the crew of the event that has just occurred and some basic understanding about its occurrence.
Isolation:	Those actions taken to contain the source of damage to the plant systems.
Mitigation:	Those actions taken to lessen the impact of an abnormal or emergency event.

Table 2. Common performance shaping factors (PSFs).

Performance Shaping Factor	Definition
Crew experience	Characterizes the experience of the operating crew.
Time to perform	Time required to perform the task.
Time available	How much time is available to perform the task before it no longer matters if the task is performed or not.
Stress	Characterizes the amount of stress the task performer is under.
Quality of plant interface	Characterizes the quality of the controls and instrumentation. Do they meet basic standards, and do they provide the necessary information?
Type of instrument/control	Describes the type of instrument or control.
Feedback to operator actions	What type of feedback does the operator receive after a control action?
Procedure required	Is a procedure available for use by the operator(s)?
Action covered by procedure	Does the content of the procedure address the actions required to perform the task(s)?
Procedure well-written	Does the procedure conform to acceptable procedure-writing standards?
Procedure practiced	Is the procedure practiced by the operations staff?
Procedure understood	Is the procedure understood by the operator?
Cognitive level of behavior	Is the behavior or action taken by the operator skill-based, rule-based, or knowledge-based?
Recovery actions	Are any actions possible that would aid the operator recovering from an error?
Tasks dynamic or step-by-step	Is the task performed concurrently with other tasks or is it performed step-by-step?
Task dependency	Is the correct performance of this task dependent on the performance of another task?
Tagging	Is tagging involved in the performance of the tasks?
Local versus remote control	Is the task performed in the control room or locally at a valve, switchgear room, etc.?

Table 2. (continued).

Performance Shaping Factor	Definition
Clothing and tools required	What special tools or equipment, such as anticontamination clothing, are required to complete the task? Usually this evaluation also entails assessing whether clothing is available and ready for use.
Environment	What are the lighting, temperature, radiation, and noise levels during task performance under conditions specified by the event sequence? For example, lighting may be poor during station blackout but temperature may be acceptable. RHR pump servicing may not be performed in warm conditions but lighting may be perfectly adequate. The environment needs to be specified in detail.

are those errors that are thought- or knowledge-intensive. They are reflected as imprecise knowledge regarding system functions or boundaries, or as poor tradeoffs between system performance and safety goals. The inability to recognize an event's signature and take the required actions is also deemed a cognitive error. All of these human errors should be treated as complex errors of commission in the ISLOCA analysis.

Human actions should be evaluated for each of the three error types (sometimes referred to as human failure modes) to make the HRA modeling process complete. The analysis should consider errors of omission and commission, both simple and complex. The completion of the HRA requires data on potential operator actions and associated errors. It is recommended that this data be obtained through onsite observations to the greatest extent possible.

There are several generic data sources available to supplement plant-specific HRA data. The generic data sources recommended for the ISLOCA analysis are

- THERP[15]
- INTENT[16]
- NUCLARR[17]

There are also several models available for use in the ISLOCA HRA. The models recommended for use include the following:

- THERP HRA event trees[15]
- HRA fault trees
- THERP special applications: error of commission event trees.[18]

3.5.3 Selection and Application of HRA Modeling Techniques. THERP-type HRA event trees are recommended for modeling most of the human actions. In some instances, however, human actions are better represented by HRA fault trees. There is no generic guidance that can be given to determine which method is better suited for a particular application.

HRA is a combination of tools and methodologies that has evolved to model different classes of human errors. Human errors are classified into categories of error for purposes of description and modeling.

There are seven categories of human error currently in use. These seven categories are described by the following terms: (a) preinitiator, (b) initiator, (c) postinitiator, (d) detection, (e) diagnosis, (f) isolation, and (g) mitigation.

Human errors are described also in terms of error type or failure mode. There are three types of human errors. Omission errors are the first type. Errors of omission are related to failure to perform a task. Errors of commission are the second type. Errors of commission are those errors associated with performing a task poorly. The last type of human error is labeled complex commission. These human errors are associated with knowledge-based decisions. Human errors are also classified either as active or latent.[19] Active errors are those whose impact is immediate. Conversely, latent errors are those whose impact is delayed. Latent errors typically are associated with maintenance and in-service testing activities.

The types and categories of human error are further subdivided in the HRA. This further classification is necessary to facilitate tracking the cause of the error. As a result, human errors are also classified in term of maintenance, in-service testing, calibration, normal operation, and off-normal operations or emergency operations.

Human errors and their associated probabilities are also influenced by situational factors. These situational factors include items such as the task, environmental factors, and personnel factors. These situational factors are often referred to as performance shaping factors (PSFs).

A classification scheme that encompasses the human error factors mentioned above was presented in Table 1. This scheme is referred to as the preliminary human error taxonomy. Table 2 presented the PSFs commonly cited in HRA and their associated definitions.

A human error may fit into more than one of the classification categories. For example, an active recovery-based isolation error could occur because of a misleading indication or lack of a procedure that would provide a means for response. The human error could also be an omission error, commission error, or involve a poor decision. The human errors in mitigating abnormal events could be due to lack of appropriate training, procedures, and equipment, or to personnel factors such as diminished physical and intellectual capacity resulting from stress and task demands.

Certain approaches to error evaluation estimate HEPs on the basis of time. One of the more prominent of these techniques is human cognitive reliability or HCR.[20] HCR allows the user to estimate a nonresponse probability or NRP. The NRP is similar to an error of omission during emergencies in which the crew fails to respond in the time available before severe plant consequences occur. Error rates calculated by this method tend to be higher for instances in which the time allowed for crew response is so brief that there is little opportunity for diagnosis and recovery. For the most part, the HCR model is more concerned with omission than with commission errors. The rationale for this emphasis is that it is difficult to postulate all the conceivable ways in which people in a crew might fail.

The range of human error evidenced in contemporary PRA often depends upon whether the HRA analysts have used nominal (i.e., best estimate) values or screening values for HEPs. The former can, in the case of THERP, be quite detailed in the information required by the analyst.[1,2,3,15] This information includes the nature of the interface, the degree of experience, the extent of tagging (e.g., the degree to which it is apparent that equipment is either in or out of service), the assessment of the potential for recovery from an error, and the time between annunciated events.

Also modeled in THERP are dependencies between people or activities, whether personnel are subject to low, medium, or high stress, whether actions to be taken are dynamic or step-by-step, whether personnel have to perform and record arithmetic calculations, and how frequently equipment such as analog meters and chart recorders is scanned. For example, people are poor in performing arithmetic calculations; estimates for error are in the range of 0.02. Therefore, procedures that call for personnel to perform such calculations will be associated with a higher probability of error. The ease of reaching equipment, such as locally operated valves often found in remote locations, is also accounted for when using THERP.

To estimate error rates with the HCR model, analysts must determine the time available for response, the average time required to respond, the stress level, crew experience, and the quality of the human-machine interface. HEP values are then obtained from a probability distribution that has been fit to theoretical and empirical performance data.

Techniques such as the success likelihood indexing method and multiattribute utility theory require input from knowledgeable experts.[21] These experts assess the most important PSFs affecting human actions for a specific situation. The experts are asked to determine which PSFs are the most important in terms of influencing human behavior. The importance rating of a PSF is multiplied by a favorableness rating, producing an HEP that is within the calibration values supplied by the HRA analyst.

In contrast to these detailed approaches, human reliability screening estimates take into account only a limited number of factors. This is in line with their intended purpose, which is to highlight efficiently those errors that are important from a risk standpoint, and thus are deserving of a more detailed analysis. Typical factors considered in a screening analysis include information as to whether actions are skill-, rule-, or knowledge-based. This approach is used in the systematic human action reliability procedure or SHARP.[22] Other screening techniques take into account whether these errors are strongly associated with the time allowed for crew diagnosis and whether the use of written procedures is well-defined for the situation. The NRC has sponsored a refined screening process for THERP as part of the Accident Sequence Evaluation Program or ASEP.[23] The ASEP-HRA analysis makes use of recovery factors in assigning screening values. Other possibilities for using screening values include using the upper bound failure rates for events.

3.5.3.1 Techniques for Identifying Human Errors. Several different analysis techniques are available for *identifying* potential human errors. The method recommended for the ISLOCA analysis is described by Hahn et al.[24] The method of Reference 24 is adept at uncovering errors that have a basis in system design, faulty maintenance or testing, and poor management (i.e., administrative control).

The method of Reference 24 identifies human errors caused by inadvertent design features as opposed to simple component failures. These types of errors can occur when there are inadequate procedural barriers, physical barriers, or operating practices. The method also addresses errors that occur when

- There are ambiguous or incorrect information displays.
- Unexpected actions occur.
- There is unintended suppression of input or output information.
- Labeling is inaccurate (e.g., when equipment that has been tagged out has been placed back in service unknowingly).

Reference 24 also can be applied in finding human error pathways that can exist if a barrier is insufficient to prevent operators from taking inappropriate actions or actions that are less than optimal for a given scenario. This type of situation is complex because operators may misread control room indications and therefore mislead themselves. Errors in timing related to performing a well-rehearsed action during the wrong plant evolution are also addressed by Reference 24.

3.5.3.2 Errors of Commission and Decision-Based Errors. HRA event trees (see Reference 15) are recommended for modeling the unique situation where successful execution of a procedure may contribute to the initiation or progression of an ISLOCA. This application of the HRA event tree is called a commission event tree or COMET.[18] The COMET graphically portrays the activities that contribute to the continuation of that particular error.

Construction of a COMET follows identification of an error of *intention* (i.e., a decision-based error). An error of intention is the execution of a task that is thought by personnel to be the correct action although it may be recognized as incorrect when all information is obtained and understood. An example of an error of intention would be the proper execution of a task performed during the wrong plant evolution or at the wrong temperature such that technical specification requirements are violated. In this example, failing to execute a subtask may actually prevent an ISLOCA from occurring. Likewise, omitting a procedural step may constitute an effective recovery factor.

The major objectives of the methodology outlined in References 16 and 18 are to provide a method for modeling decision-based errors not included in contemporary PRAs and to provide a preliminary set of data based on expert opinion that may be used to bound the estimates of such failure probabilities. The methodology provides upper and lower confidence bounds for 20 generic decision-based errors. The INTENT methodology allows analysts to refine these bounding limits by rating 11 commonly used PSFs and then multiplying these ratings by a series of weights. The resultant value is used to determine a HEP value lying between the two extremes.

3.5.4 Quantifying the HEPs. The human actions that make up the significant ISLOCA sequences will have been identified after completion of the above analyses, reviews, and evaluations. It is recommended that a list of human actions to be quantified be developed. This list provides the basis for the required human error data. It is recommended that the data sources used to derive the HEPs come from three sources. The recommended data sources are

- THERP Tables from Chapter 20 of Reference 15
- Values from Table 1 of Reference 16
- Values selected from Reference 17.

3.6 Auxiliary Building Evaluations

Auxiliary building thermal hydraulic calculations may be needed to judge what equipment will be available to operators attempting to control the plant and to estimate the time available for recovery actions before core integrity is threatened. At one extreme, the operators may have no more than a few minutes to isolate the break before the high temperatures and humidity from the break disable the equipment needed to keep the core covered with water. At the other extreme, the required equipment may be qualified to the expected temperature and humidity levels, and the operators may be able to throttle coolant injection such that many hours will be available in which to isolate the break. This section provides some guidance for how detailed an analysis is needed to realistically evaluate the time available for recovery.

3.6.1 Temperature and Humidity Effects. Temperature effects are the easiest to evaluate since some generalizations are possible. For the most likely break sizes and for most auxiliary building designs the temperature in the auxiliary building will reach 100°C within minutes after the break. Exceptions to this occur in highly compartmentalized auxiliary buildings in which the separate trains of ECCS are relatively isolated from one another. Review of a number of PWR auxiliary building designs as part of the ISLOCA Research Program has shown that most designs are relatively open with respect to steam flow paths. There are generally sufficient stairwells, doorways, and pipe chases to allow unobstructed flow of steam through the auxiliary building. Even plants with watertight doors isolating the different ECCS pump rooms were sometimes found to have these doors ajar. Since the most likely break locations have been shown to be in the large pipes and heat exchangers of the low-pressure injection systems, the time required to flush air from the building can be as little as 2 or 3 min. The live steam environment that remains should be assumed to disable any ECCS components not qualified to this environment. Temperatures higher than 100°C may be possible in rare cases in which there is either significant pressurization of the auxiliary building or superheat in the break discharge. Of these two considerations, superheat in the break discharge is the more realistic possibility. This would require an RCS configuration that allows steam to be pulled into the interfacing system piping with very little accompanying liquid, a condition most likely just before core damage, when temperature stratification in either the hot leg or cold leg (depending on the location of the interface) could uncover the entrance to the interfacing system piping.

From the above considerations, it is apparent that the first step in estimating if the available recovery time is more than several minutes is to determine whether the ECCS equipment has been qualified to 100°C and 100% relative humidity. This may be the case if high-energy lines pass through the ECCS areas of the auxiliary building. A simple review of any existing high-energy line break analyses may be sufficient to dismiss temperature and humidity as limiting factors for the available time to recovery. If the equipment qualification is not compatible with the expected environment, it may still be possible to demonstrate longer equipment survival times if the building has strong isolation between the equipment rooms. This type of argument should be supported by thermal hydraulic calculations as discussed in Section 3.6.3.

3.6.2 Flooding. Even if the ECCS equipment is qualified to the expected effects of temperature and humidity, the break discharge is still likely to flood some or all of the ECCS systems. However, what equipment is flooded is very dependent on the specific auxiliary building design. The review of PWR auxiliary building designs performed as part of the ISLOCA Research Program found a large variation in vulnerability to flooding. The most flood-resistant buildings had large floor areas to spread the water and limit the rate of water rise, and had the ECCS equipment located on different levels of the building. Plants with this type of configuration may argue that, for the range of break sizes expected, the accumulated break discharge will never affect all available ECCS equipment. However, for the majority of designs, the threat from flooding will be hard to evaluate without detailed calculations (discussed in Section 3.6.3).

If detailed calculations are required to estimate the time available before essential equipment is submerged, the following should be included in the analysis:

- Drainage paths among compartments. The obvious paths are those provided by doorways and pipe chases, but floor drain systems can transport water to ECCS compartments from spaces that are otherwise well-isolated by doorways or flood walls.

- The presence of sprinkler systems for fire protection. Live steam and hot water fogs will likely actuate any sprinkler systems installed within reach of the expanding steam. The discharge from sprinkler systems may limit the available recovery time even when break sizes are sufficiently small that flooding via unflashed break discharge and condensate is not significant.

- The possibility of flooding directly from the ECCS water storage tank. Breaks in the LPI systems may occur on the suction side of the pumps, where discharge from the break may originate from both the RCS and from the water storage tank.

3.6.3 Detailed Code Calculations. The thermal hydraulic models required to do a credible job of evaluating steam and flood water propagation through the auxiliary building are found in most containment analysis codes. Codes such as CONTEMPT, HECTR, CONTAIN, MAAP, and MELCOR are all capable of providing adequate auxiliary building simulations. The first three will require separate calculation of the break discharge with a LOCA code unless the required blowdown information is available in the plant's Final Safety Analysis Report (FSAR). The MAAP and MELCOR codes provide integrated RCS and auxiliary building simulations. Extending blowdown codes such as RELAP5 to include auxiliary building volumes is not recommended because RELAP5 as it currently exists does not conserve energy adequately at the break plane.

Based on the ISLOCA Research Program experience, the models for the auxiliary building do not need to be complex. Compartments distant from the ECCS equipment can be lumped together, unless they contain fire protection systems that may contribute to flooding. Most auxiliary building designs could be simulated adequately with as few as five or six volumes. Larger models are unlikely to yield more information of value, since the important transient behavior is occurring within the compartments close to the break, and the compartments further away have relatively little influence on the phenomena occurring closer in. It is sufficient to get the auxiliary building volume correct and to correctly model the flow paths between the break compartment and the remaining ECCS compartments (or compartments connected to them by floor drains).

Modeling of heat structures such as concrete walls, ceilings, and floors is recommended, although experience has shown these will have little affect when the break size corresponds to an LPI/RHR system failure. However, smaller breaks may be affected strongly by these modeling components. The modeling of piping and other metal masses could also affect the smaller break calculations; however, it is probably not worthwhile to attempt to make a reasonable estimate of the available metal mass and condensing mass in each compartment.

Most of these codes have all the features required to model flood propagation. The most important modeling parameter here is the floor area in each affected compartment. The major contributors to flooding will be the unflashed break discharge, which may be as large as the runout flow of two LPI pumps, and the discharge from fire suppression systems. Condensate will be a much smaller contributor. Given the magnitude of these sources, floor sump pumps will do little to mitigate flooding.

The above discussion is not intended to provide an absolute set of guidelines for auxiliary building environmental analysis. Instead, it is intended to focus attention on the most important aspects of auxiliary building modeling. It is hoped that this information will allow the analyst to construct adequate models without an excessive investment in modeling detail.

4. BWR ISLOCA ANALYSIS

4.1 Introduction

The primary purpose of this analysis is to assess the ISLOCA risk for a BWR plant. Previous reports[1,2,3] have documented the results of ISLOCA evaluations of three PWR plants, and to complete the picture a BWR plant was examined. One objective of the ISLOCA Research Program is the identification of generic insights. Toward this end, a BWR that, as much as possible, is representative of a large population of U.S. BWR plants was chosen as the basis for this analysis.

The reference BWR plant used as the subject of this analysis is one of the BWR/4 design series with a Mark-I containment. Power is rated at 3293 MWt and 1150 MWe. The plant uses a high pressure coolant injection (HPCI) system, a single mode reactor core isolation cooling (RCIC) system, a low pressure core spray (CS) system, and a multimode residual heat removal (RHR) system with no steam condensing capabilities (original capability has been removed). Plants in the U.S. that are of the BWR/4 design include:

- Vermont Yankee
- Peach Bottom Units 2 and 3
- Hatch Units 1 and 2
- Cooper
- Fermi Unit 2
- FitzPatrick
- Brunswick Units 1 and 2
- Browns Ferry Units 1, 2 and 3
- Duane Arnold
- Hope Creek
- Limerick Units 1 and 2 (with a Mark-II containment)
- Susquehanna Units 1 and 2 (with a Mark-II containment).

The reactor coolant system (RCS) normally operates at a pressure of 1020 psia and contains water and steam volumes of 13,161 ft^3 and 8,873 ft^3, respectively.

This document describes an evaluation performed on the reference BWR from the perspective of estimating or bounding the potential risk associated with ISLOCAs. Toward this end, those sequences that are judged to be not risk-significant are neglected. This includes operational type of sequences in which an interfacing system is overpressurized but either does not rupture or does not result in a reactor coolant flow path from the reactor vessel to the outside of the primary containment. Note that this criteria screens out feedwater flow diversion sequences.

Screening criteria were also used to remove from further consideration scenarios that were judged to be too improbable. A two-step screening criteria was used based (a) on the conditional probability of rupturing the interfacing system and (b) on a bounding calculation on the probability of producing an ISLOCA sequence. A lower bound conditional rupture probability of 1E–3 is used for screening out fluid system components from the analysis. This lower bound on the rupture probability is based on the work of Wesley et al.[8,9] In that work, the median pipe rupture probability was estimated using actual material properties rather than code or design specifications. The uncertainty associated with the median value was estimated assuming a 1E–3 probability of the presence of a very large flaw in the pipe that results in the failure of the pipe at yield. The consequence of this premise is that failure probabilities less than 1E–3 indicate the system pressure is not high enough to produce yield stresses in the pipe. A value of 1E–8 per year is used as the cutoff for further considering ISLOCA sequences. This limit includes all events that compose an ISLOCA sequence and include bounding estimates on valve failures, human errors, and the conditional rupture probability identified above.

4.2 Interfacing Systems

A survey of all containment penetrations was performed to identify possible situations in which an inter-system LOCA could occur. The approach taken began with an inventory of these penetrations to compile a list of interfacing systems. Once the list was complete, the design information for each system was reviewed to determine the potential for a rupture given that an overpressure had occurred. This list included the following systems:

1. Reactor core isolation cooling system
2. High pressure coolant injection system
3. Core spray system
4. Residual heat removal system
5. Reactor water cleanup system
6. Control rod drive hydraulic system.

4.2.1 Reactor Core Isolation Cooling (RCIC). RCIC is a low flow, high pressure, open loop water supply system with a turbine driven pump. RCIC automatically starts when a low reactor pressure vessel (RPV) water level (Level-2) is detected. After initiation, RCIC provides the required flow (600 gpm) of makeup water to the reactor vessel, at normal operating reactor pressure, in less than 50 s after receipt of the initiation signal. Figure 3 provides a simplified schematic of the RCIC system.

The two main flow paths associated with RCIC are a steam flowpath and a water flowpath. When RCIC is initiated these flowpaths are automatically aligned. The steam flowpath begins at Main Steam Line B, upstream of the inboard Main Steam Isolation Valve (MSIV). Steam passes from Main Steam Line B through the inboard and outboard RCIC isolation valves, the turbine supply valve isolation valve, the trip-throttle valve, and the control valve. The steam then enters the RCIC turbine, where it expands to provide the motive force for the RCIC pump. The turbine exhausts to the suppression pool.

The major water flowpath is from the Condensate Storage Tank (CST) through the RCIC pump and to feedwater line-B through a tee connection outside the primary containment. RCIC water is distributed through the feedwater sparger. The CST is the preferred water source for RCIC. Makeup water is initially supplied from the CST, but can alternatively be supplied from the suppression pool if the water in the CST falls below a minimum level. In this situation, RCIC pump suction is automatically transferred to the suppression pool.

Additional flowpaths include steam leak-off and drains directed to the barometric condenser, cooling water from the RCIC pump discharge to the lube oil cooler and the barometric condenser, barometric condenser condensate pump discharge to the suction of the RCIC pump, and the vacuum pump discharge to the suppression pool.

Table 3 contains a list of the RCIC design specifications and the pressure capacities for various portions of the system.

4.2.1.1 RCIC Leak Detection System. The RCIC system is a high pressure core coolant makeup and cooling system. The system shares a corner room with one train of the core spray system (Division I). The RCIC system consists of both low and high temperature flows, and is designed for high pressure coolant injection. The high temperature flow consists of steam bleed from the RCS to drive the RCIC turbine. The low temperature system is the piping containing the RCS injection flow. The RCS injection flow is taken from either the condensate storage tank or the suppression pool.

Several measurements are required to detect leaks in the RCIC system. The leak detection system consists of measurements of steam flow, compartment temperature, and sump level. The RCIC system automatically isolates from the reactor's coolant system upon the detection of

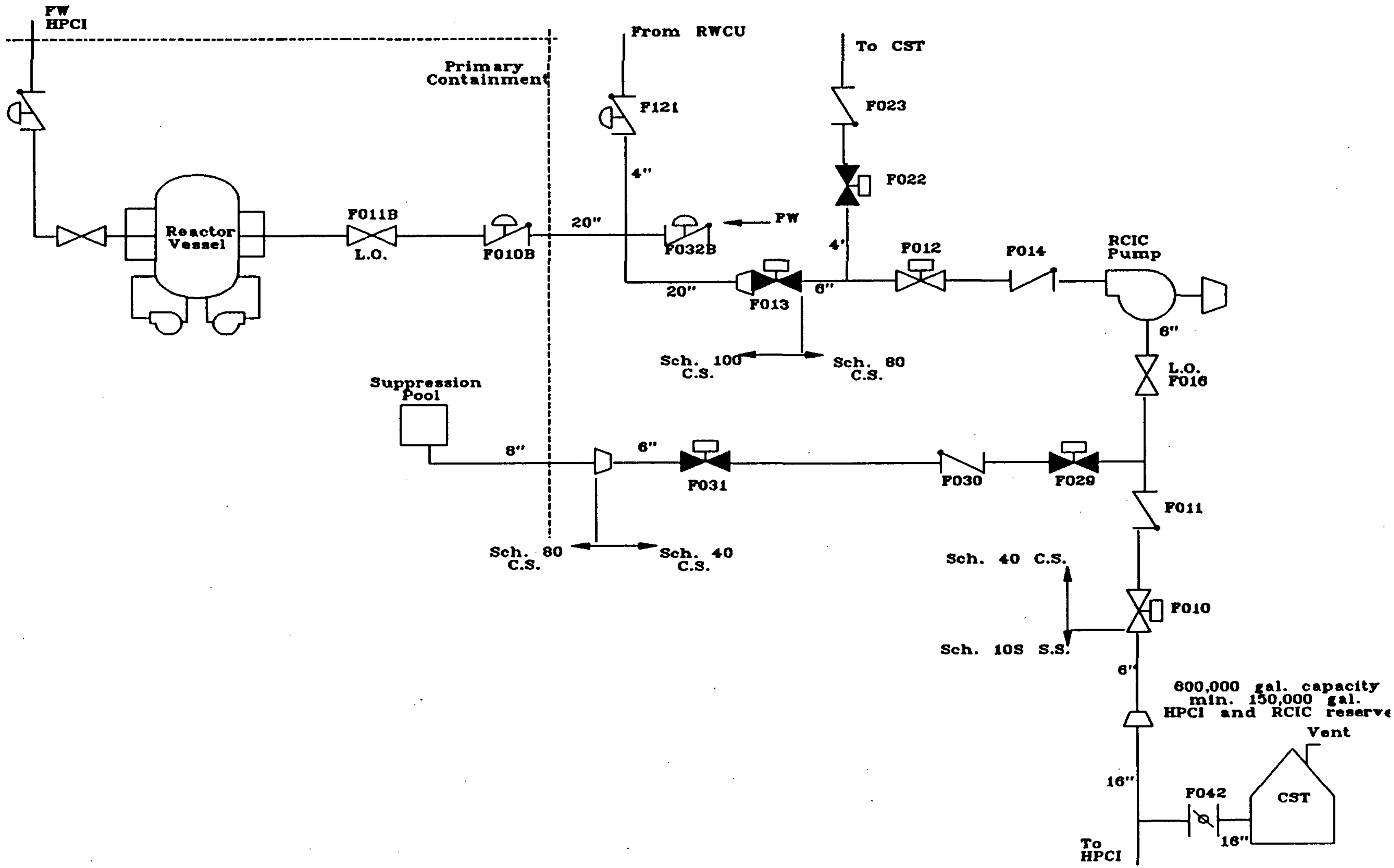

Figure 3. RCIC simplified diagram.

Table 3. RCIC system design specifications.

Component	Design pressure (psig)	Design temperature (°F)	Material	Dimension	Median pressure fragility[a] (psi)	Rupture probability[b]
Class 1 piping, Injection line	1275	450	Carbon steel	6 in. Sch. 20, 20 in. Sch. 100	11,034 8,031[c]	Negligible Negligible
Class 2 piping, pump discharge	1280	170	Carbon steel	6 in. Sch. 80	8,009	Negligible
RCIC pump	1500	212	Carbon steel	625 gpm, 2800 ft discharge head	>>1,000	Negligible
Class 2 piping, pump suction: - From CST - From SP and RHR HX	18 125	120 190	Stainless steel, carbon steel	6 in. Sch. 10S 16 in. Sch. 10S 6 in. Sch. 40S	1,585 951 4,801	0.11[d] —[d] Negligible

a. Median pressure fragility data for carbon steel pipe assumed a corrosion allowance of 0.02 in. and a temperature of 600°F.

b. Rupture probability calculated using an internal pressure of 1,020 psi.

c. Interpolated between the fragilities for 20 in. Sch. 80 and 20 in. Sch. 120.

d. CST is vented to the atmosphere, and therefore the 16-in. portion of the suction pipe (closest to the CST) will not pressurize to as high a pressure as the 6-in. portion will.

leaks as a result of either high steam flow or high RCIC compartment temperature.

Temperature sensors are used to detect RCS leaks into the RCIC and core spray compartment. These temperature sensors are located in the inlet and outlet of the ventilation ducts and in the inlet of the emergency area cooler of the RCIC room. The temperature sensors are used to measure the temperature rise in the RCIC room and the room's ambient temperature. A high ambient temperature and high temperature rise are annunciated in the main control room. A high area temperature results in the automatic isolation of the RCIC system from the reactor coolant system.

Steam line ruptures are detected by differential pressure transmitters in the RCIC turbine feed lines. Steam leaks in the RCIC system are detected by a set of two differential pressure transmitters. These pressure transducers sense the differential pressure drop produced by steam flow across an orifice plate. Flow in excess of a specified limit isolates the RCIC system and activates an alarm in the main control room.

The steam driven turbine's exhaust vent lines are also monitored for leaks. The monitoring system consists of four pressure transducers. A high turbine exhaust pressure results in the isolation of the RCIC and activates an alarm in the main control room.

Since the RCIC system shares a room with core spray, leaks in the RCIC injection suction lines from the torus are detected by the core spray sump level alarm. This alarm notifies the operations crew that a flooding condition may exist in the RCIC/CS room. This signal does not result in the automatic isolation of the RCIC system from the RCS. The signal allows the operator to terminate the leak. These actions prevent the loss of suction to other ECCS suction lines. The plant

EOPs address these sump level alarms and direct the operations crew to isolate these RCIC leaks.

An intersystem LOCA through the RCIC injection line could result in the rupture of the pump suction line. The RCIC system suction is normally aligned to the CST. If a suction line rupture occurred in the RCIC room, the high temperature coolant would result in an increase in the ambient temperature. The room's high ambient temperature would result in the automatic isolation of the RCIC system from the RCS.

RCIC Flooding: The RCIC system is located in one of the four corner rooms of the reactor building subbasement. An unisolated RCS rupture in the RCIC or core spray system pressure boundary in this corner room can result in the flooding of the corner room. This flooding, if the leak is not isolated, can propagate to the remainder of the reactor building, the auxiliary building, and the turbine hall. The facilities leak detection system will automatically isolate the RCIC system from the reactor coolant system. Automatic isolation of RCIC should occur due to high ambient temperature in the corner room. The isolation should terminate the flood propagation.

Emergency Operating Procedures are in place to detect, diagnose, and isolate leaks that occur in the RCIC/CS corner room. These procedures enhance the ability of the operations crew to prevent flooding conditions from spreading to other portions of the reactor building. The operations crew are instructed to line up the RHR service water bypass to provide long-term core cooling in the advent of loss of torus and condensate storage water. This coolant supply provides a long term supply of coolant water to the reactor.

The flood from an unisolated break will propagate up from the RCIC corner room (subbasement level) into the basement. The flood will then enter the auxiliary building through openings around a 3-h fire door. The flood will propagate throughout the auxiliary building and will enter the turbine room and the control rod drive pump room. The control rod drive pump room is located away from the RCIC/CS room on the other side of the auxiliary building. The flood will then drain into the HPCI room from the control rod drive pump room. After flood of the HPCI room, the flood will propagate into the other core spray room through an opening between the CRD pump room and the basement level of the core spray corner room. If the water level is sufficient to bring the flood into the first floor level of the reactor building, the flood will then propagate into the RHR A&B corner rooms and the RHR A&B heat exchangers. The flood will not enter the environment at this level due to the use of watertight doors between the reactor building and the grade level.

4.2.1.2 RCIC Suction Line Analysis. The rupture probability screening shown on Table 3 identifies a single pipe as being susceptible to rupture caused by overpressurization of the RCIC system, namely the RCIC pump suction pipe from the CST (between valve F010 and the CST). However, in order to pressurize this pipe, three valves must fail or be inadvertently opened. These are MOV F013 (normally closed), and check valves F014 and F011. Furthermore, these three valve failures will only result in the diversion of feedwater. For a loss of coolant from the reactor to occur, feedwater check valve F010B must also fail. Therefore, the RCIC system is judged to be not susceptible to ISLOCA ruptures caused by overpressure conditions.

4.2.2 High Pressure Coolant Injection (HPCI). HPCI is an integral part of the Emergency Core Cooling System (ECCS). The HPCI system provides makeup water to the reactor pressure vessel (RPV) in the event of a small loss of coolant accident that does not rapidly depressurize the RPV. HPCI is an open loop water supply system with a turbine driven centrifugal pump assembly. It automatically starts when a low RPV water level and/or a high drywell pressure condition is detected. After initiation, HPCI provides the required flow (5,000 gpm) of cooling water to the reactor vessel, at normal operating reactor pressure, in less than 30 s after receipt of an initiation signal.

Figure 4 is a simplified schematic diagram of the HPIC system. The two main flowpaths

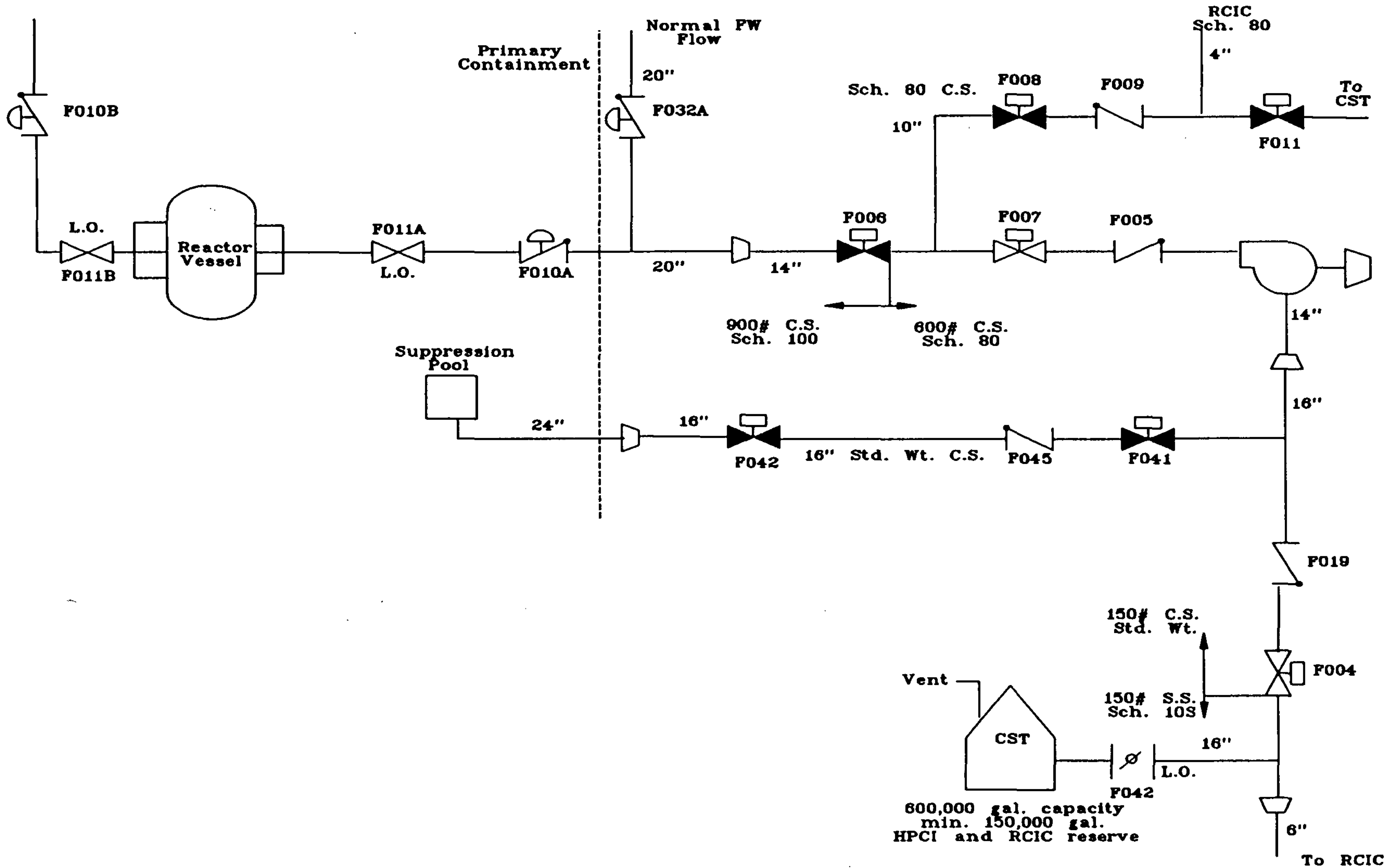

Figure 4. HPCI simplified diagram.

associated with HPCI are a steam flowpath (not shown in Figure 4) and a water flowpath. When HPCI is initiated these flowpaths are automatically aligned. The steam flowpath begins at a main steam line upstream of the inboard main steam isolation valve. Steam passes from the main steam line through the inboard and outboard HPCI isolation valves, the turbine supply isolation valve, the turbine stop valve, and the turbine control valve. The steam then enters the HPCI turbine, where it expands to provide motive force for the HPCI pump assembly. The turbine exhausts to the suppression pool.

The major water flowpath is from the condensate storage tank (CST) through the booster pump, through the main pump and to a feedwater line through a tee connection outside the primary containment. HPCI water is distributed through the feedwater sparger. The HPCI pump assembly consists of the booster pump, a reduction gearset, and the main pump mounted on a common baseplate. The booster pump assures adequate net positive suction head is available to the main pump to prevent cavitation. The CST is the preferred water source for HPCI. Makeup water is initially supplied from the CST, but can alternatively be supplied from the suppression pool if the water in the CST falls below a minimum level or suppression pool water level is high. In these situations, HPCI pump suction is automatically transferred to the suppression pool.

Additional flowpaths include steam leak-off and drains directed to the barometric condenser, cooling water from the booster pump discharge to the lube oil cooler and the barometric condenser, barometric condenser condensate pump discharge to the suction of the booster pump, and the vacuum pump discharge to the standby gas treatment system.

Table 4 lists the design specifications of the HPCI system as well as the pressure capacity information for various portions of the system.

4.2.2.1 HPCI Leak Detection. The HPCI system is a high pressure emergency core cooling system. The system consists of both low and high temperature coolant flows. The high temperature flow consists of steam from the RCS, which is used to drive the HPCI turbine. The low temperature system is the piping containing the RCS injection flow from either the condensate storage tank or the suppression pool. The most probable pressure induced failure locations of the HPCI system during an ISLOCA are the suction lines from the Condensate Storage Tank (CST) and the torus.

To detect leaks in this system, several measurements are required. The leak detection system consists of measurements of steam flow, compartment temperature and sump level. The HPCI system automatically isolates from the reactor coolant system upon the detection of leaks as a result of either high steam flow or high HPCI compartment temperature.

Temperature sensors are used to detect RCS leaks into the HPCI compartment. These temperature sensors are located in the inlet and outlet of the ventilation ducts and in the inlet of the emergency area cooler of the HPCI room. The temperature sensors are used to measure the temperature rise in the HPCI room and the room ambient temperature. High temperature and high temperature rise are annunciated in the main control room. A high area temperature results in the automatic isolation of the HPCI system from the reactor coolant system.

RCS steam leaks are detected by differential pressure transmitters in the turbine feed lines. Steam leaks in the HPCI system are detected by a set of two differential pressure transmitters. These pressure transducers sense the differential pressure drop produced by steam flow across an orifice plate. Flow in excess of a specified limit isolates the HPCI system and activates an alarm in the main control room.

The HPCI turbine exhaust vent lines are also monitored for leaks. The monitoring system consists of four pressure transducers. A high pressure on the turbine exhaust results in the isolation of the HPCI and activates an alarm in the main control room.

Table 4. HPCI system design specifications.

Component	Design pressure (psig)	Design temperature (°F)	Material	Dimension	Median pressure fragility[a] (psi)	Rupture probability[b]
Class 1 piping, pump discharge to feedwater line	1275	450	Carbon steel	14 in. and in. Sch. 100	>6,000	Negligible
Class 2 piping, pump discharge to feedwater line	1330	170	Carbon steel	14 in. Sch. 80	6,540	Negligible
Class 2 piping, pump recirculation line to CST	1330	170	Carbon steel	10 in. Sch. 80	6,710	Negligible
HPCI main pump	1500	40 to 212	Cast steel	5000 gpm at 2800 ft discharge head	>>1,000	Negligible
HPCI booster pump	450	40 to 212	Cast steel	5100 gpm	>>1,000	Negligible
Pump suction piping:	125					
- From CST		140	Stainless steel	16 in. Sch. 10S	1,204	0.32
- From SP		170	Carbon steel	16 in. × 0.375 in. wall thickness	2,607	0.0044

a. Median pressure fragility data for carbon steel pipe assumes a corrosion allowance of 0.02 in. and a temperature of 600°F.

b. Rupture probability is calculated using an internal pressure of 1,020 psi.

Suction line leaks that result in flooding of the HPCI room are detected by a sump level alarm. This alarm notifies the operations crew that a flooding condition may exist. This signal (i.e., the sump level alarm without the temperature alarm) does not result in the automatic isolation of the HPCI system. However, the plant EOPs address these sump level alarms and direct the operations crew to isolate these leaks.

An inter-system LOCA through the HPCI injection line could result in the rupture of the pump suction line to the CST. The HPCI system is normally aligned to the CST. An RCS leak through a suction line rupture in the HPCI compartment would result in an increase in the ambient temperature of the HPCI compartment. This high ambient temperature results in the automatic isolation of the HPCI system from the RCS.

4.2.2.2 HPCI Suction Line Analysis. The rupture probability screening shown on Table 4 identifies two sections of pipe as being susceptible to rupture caused by overpressurizing the HPCI system: namely, those sections of pipe to the HPCI pump suction from the CST and from the suppression pool. In particular, the pipe bounded by the HPCI pump suction, MOV F041, and check valve F019 appears most vulnerable to being overpressured and possibly ruptured. The sequence of events required to produce a rupture in this pipe begins with the inadvertent opening of MOV F006, followed by the failure of check valve F005 to prevent backflow. This scenario will result in the overpressure and possible rupture of the HPCI suction pipe. However, these failures only result in feedwater being diverted. For coolant to be lost from the reactor vessel, the feedwater check valve F010A also must fail to close. In addition the rupture probability (given the pipe is pressurized to 1020 psi) is 4.4E–3. An additional check valve (F019) failure is required to pressurize the 16-in. stainless steel schedule 10S pipe, and therefore that portion is neglected. A bounding calculation for this scenario is estimated at less than 1E–9 [using the data presented in Appendix A, MOV-NCFO[f] F006 = 5E–3, CV-NCFO F005 = 4E–3 (5E–7/hr * 8760hr), CV-NOFO F010A = 1E–3, and rupture of pipe = 4.4E–3]. Therefore, the HPCI system is judged to not be an important consideration with respect to ISLOCA risk.

4.2.3 Core Spray (CS). CS forms a part of the emergency core cooling systems (ECCS) divisional network. The ECCS network consists of

1. High Pressure Coolant Injection (HPCI) system
2. Automatic Depressurization System (ADS)
3. Core Spray (CS) system
4. Low Pressure Coolant Injection (LPCI).

HPCI and ADS provide high pressure ECCS capability while LPCI and CS satisfy low pressure requirements. LPCI is a mode of the Residual Heat Removal (RHR) system.

The purpose of the CS system is to provide inventory makeup and spray cooling during a large break loss of coolant accident (LOCA) in the nuclear system. Under emergency conditions coolant is automatically pumped from the suppression pool (or condensate storage tank if the SP is drained) to remove decay heat from the core. The CS system also provides protection during a small break LOCA if the feedwater pumps, control rod drive pumps, RCIC system, and the HPCI system are all unable to maintain RPV water level. Under these conditions, the ADS operates to lower RPV pressure so that CS can function to provide core cooling.

f. NCFO—Normally Closed, Fails Open.

The CS system is a closed loop system designed to reflood the reactor pressure vessel (RPV) by spraying water through spargers over the core following a LOCA. This action removes decay heat and is designed to prevent the peak fuel cladding temperatures from exceeding 2,200°F. Each of the two independent loops in the CS system consists of two 50% capacity pumps. These pumps take suction from the suppression pool and discharge through piping and spargers over the top of the fuel. The CS system automatically initiates on High Drywell Pressure or RPV Level-1. The CS pumps start and circulate water in a minimum flow path. Coolant injection to the RPV begins when reactor pressure decreases to allow injection valves to open and reactor pressure is below the pumps shutoff head. A simplified diagram of the core spray system is shown on Figure 5.

Table 5 lists the design specifications and fragility information for various portions of the core spray system.

4.2.3.1 Core Spray Leak Detection. The core spray system consist of two independent low pressure injection trains. One of these trains shares a subbasement corner room with the RCIC system. The other train occupies another subbasement corner room of the reactor building, which is adjacent to the HPCI room. The reactor building's leak detection system for the core spray system consists of temperature dependent sensors, sump level alarms and system overpressure alarms.

Temperature sensors are used to detect RCS leaks into the core spray corner rooms. The temperature sensors are located in the inlet and outlet of the ventilation ducts and in the inlet of the emergency area cooler. The sensors are mounted such that they are primarily sensitive to air temperature. The sensors are used to measure the temperature rise in the room and the room's ambient temperature. A high ambient room temperature and high temperature rise are annunciated in the main control room.

Core spray system leaks into the corner room are also detected by a sump level alarm. This

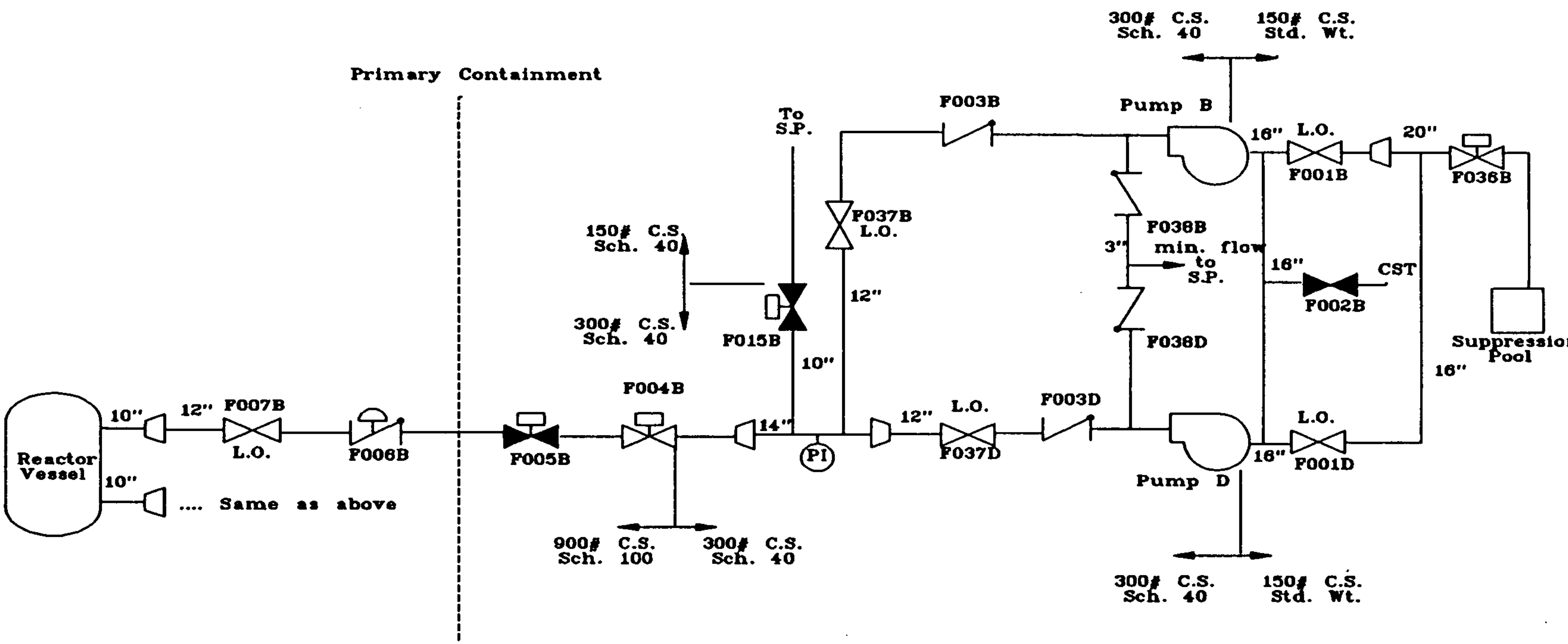

Figure 5. Core spray simplified diagram.

Table 5. CS system design specifications.

Component	Design pressure (psig)	Design temperature (°F)	Material	Dimension	Median pressure fragility[a] (psi)	Rupture probability[b]
Class 1 piping, pump discharge	1250	575	Carbon steel	12 in. Sch. 100	>6,500	Negligible
Class 2 piping, pump discharge	500	212	Carbon steel	12 in. and 14 in. Sch. 40	3,621 3,562	2E-4 3E-4
CS pump	500	40 to 212	Cast carbon steel	3175 gpm, 668 ft discharge head	>>1,000	Negligible
Class 2 piping, pump suction	125	212	Carbon steel	16 in. and 20 in. × 0.375 in. wall thickness	2,607 2,065	0.0044 0.023

a. Median pressure fragility data for carbon steel pipe assumed a corrosion allowance of 0.02 in. and a temperature of 600°F.

b. Rupture probability calculated using an internal pressure of 1,020 psi.

alarm notifies the operations crew that a significant leak exists. The signal allows the operator to terminate the leak to prevent further loss of reactor coolant. The plant EOPs address these sump level alarms and direct the operations crew to isolate these leaks.

The core spray system also contains an alarm to warn the operators of high pressure in the RCS injection line. This alarm is set to annunciate when the pressure on the discharge side of the pump exceeds 450 psig. This alarm is designed to warn of pressure isolation valve leakage into the core spray system.

An intersystem LOCA through the core spray injection line could result in the rupture of the pump suction line. The core spray suction line is aligned to the torus. If a suction line rupture occurred in the core spray corner room, the high temperature reactor coolant would result in an annunciation of three leak detection alarms. These alarms are (a) high ambient temperature, (b) high sump level, and (c) high injection line pressure.

The core spray system is divided into two corner rooms on the subbasement level of the reactor building. Division I of core spray occupies the same corner room as RCIC. Thus the flooding propagation analysis provided for the RCIC system is also applicable to Division I of core spray. The Division II core spray flooding propagation assessment is described in the following sections.

Emergency Operating Procedures are in place to detect, diagnose and isolate leaks that occur in the core spray corner rooms. These procedures enhance the ability of the operations crew to prevent flooding conditions from spreading to other portions of the reactor building. The plant staff are also instructed to line up the RHR B service water bypass to provide long term core cooling. These actions are taken in the advent of loss of wetwell and condensate storage water. This coolant supply provides a long term supply of coolant water to the reactor.

If the core spray room flooding is not terminated the flood will propagate up from the corner room subbasement into the basement. The flood will then enter the control rod drive pump room. From the control rod drive pump room, the flood will drain into the HPCI room through an open equipment hatch and stairwell. After filling the HPCI room, the flood will propagate into the auxiliary building and will enter the turbine room. From the auxiliary building, the flood will enter the RCIC and core spray corner room on the other side of the reactor building. If the water level is sufficient to bring the flood into the first floor level of the reactor building, the flood will propagate into the RHR A&B corner rooms and the RHR A&B heat exchanger compartments. The flood will not enter the environment from the first floor of the reactor building because of the use of watertight doors between the reactor building and the grade level.

4.2.3.2 CS Analysis. Two portions of the CS system were identified as potential concerns with respect to ISLOCAs. However, the CS pump discharge piping does not survive the 1E–3 rupture probability screening criteria and is not considered further. The CS pump suction pipe survived the first level screening and was subjected to a bounding estimate of the ISLOCA scenario probability (that is, the probability of failure of the CS piping = CS-ISLOCA). Because the 20-in. line is connected to the RCS by smaller diameter pipes (16-, 12-, 14-, and 10-in.) and empties to the suppression pool, it will not pressurize significantly. Consequently, only the 16-in. line is included in the bounding calculation. This calculation produced a bounding probability estimate of

$$\begin{aligned} CS\text{-}ISLOCA &= F006B * F005B * (F003B \\ &\quad + F003D) * RUPT \\ CS\text{-}ISLOCA &= TCV\text{-}NCFO * MOV\text{-}NCFO \\ &\quad * 2(CV\text{-}NCFO) * 0.0044 \\ CS\text{-}ISLOCA &= 9E\text{–}3 * 5E\text{–}3 * 2(5E\text{–}7 * 8760) \\ &\quad * 0.0044 \\ CS\text{-}ISLOCA &= 1.7E\text{–}9\,. \end{aligned}$$

Therefore, the core spray system is judged to not be susceptible to an ISLOCA caused by an overpressure induced rupture.

4.2.4 Residual Heat Removal (RHR). The RHR system is a closed loop system of piping, pumps, and heat exchangers divided into two loops. Each loop consists of two 33-1/3% capacity single stage centrifugal pumps and a 100% capacity shell and tube heat exchanger. The purpose of the system is to remove post power-operation thermal energy from the reactor under both normal and accident conditions. The system restores and maintains, if necessary, the water level in the reactor vessel after a loss of coolant so the core is sufficiently cooled to preclude fuel damage. The system also removes decay heat and sensible heat from the reactor primary system during shutdown so that the reactor can be refueled and serviced. In addition, when required, the system cools the suppression pool and fuel pool and provides for containment cooling spray. Figure 6 is a simplified diagram of the RHR system (Division II is shown, Division I is identical with the exception that there is only a single letdown line that is shared by the two trains). As can be seen from the figure, there are two interfaces between the RHR system and the RCS, the injection interface and the letdown line. The system comprises six subsystems or operating modes, which are briefly described below.

Low Pressure Coolant Injection (LPCI). The LPCI mode of RHR is part of the emergency core cooling system (ECCS). The LPCI mode uses the RHR pumps to inject water taken from the torus into the reactor during a loss of coolant accident (LOCA). Upon receipt of a LOCA signal, the LPCI pumps take suction from the water volume in the torus. The pump discharge travels through the RHR heat exchangers and into the reactor vessel via one of the reactor recirculation loops.

Suppression Pool (Torus) Cooling Subsystem. Under conditions where heat has been or is being added to the torus water (e.g., SRVs open) the RHR system can be aligned to remove the heat. In the torus cooling mode the RHR pumps take suction water from the torus and discharge through the RHR heat exchangers. The heat is transferred to the RHR service water and ultimately rejected via the mechanical draft cooling towers. The cooled water is returned to the torus through the RHR pump test line.

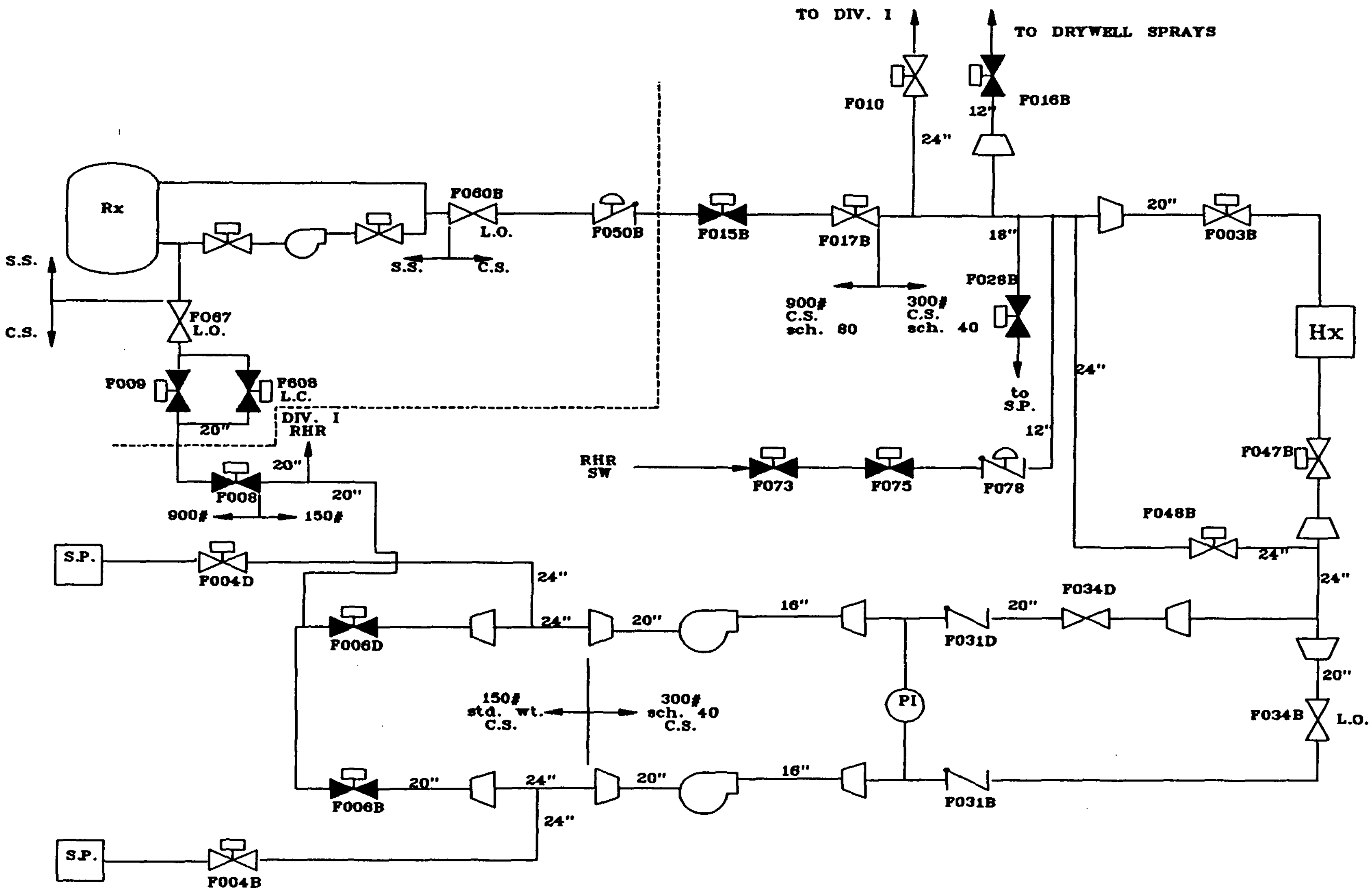

Figure 6. RHR simplified diagram for Division II; Division I (not shown) is essentially the same, but component identifiers have different alphabetical suffixes.

Torus Spray. The torus spray provides for condensation of steam and cooling of non-condensible gases that may be present in the torus air space. In this mode the RHR pumps draw water from the torus water volume and discharge through the RHR heat exchangers where it is cooled by the RHR service water. Flow returns to the torus via the torus spray line that contains a throttle valve. Flow then enters a spray sparger where it is sprayed into the air space above the torus water volume.

Drywell Spray Mode. Drywell spray is initiated manually under conditions where the drywell pressure and/or temperature has the potential of exceeding its design value. This provides containment cooling for postaccident conditions. In the drywell spray mode, the RHR pumps take water from the torus water volume and discharge through the RHR heat exchangers where it is cooled by the RHR service water. The train-A RHR drywell spray enters a sparger and sprays into the lower part of the drywell air space. Loop-B spray spargers direct water to the upper areas of the drywell. The sprayed water falls to the bottom of the drywell and eventually overflows back to the torus through the drywell downcomers.

Shutdown Cooling. The shutdown cooling mode is used to remove decay heat from the reactor core and the heat stored in the vessel internals so that the reactor can be refueled and serviced. This mode is initiated during normal shutdown when vessel pressure is less than 95 psig. The RHR system is designed to cool the reactor water to 125°F within 20 h of shutdown and maintain that temperature indefinitely. The RHR pumps draw water from the B recirculation loop suction pipe through two common isolation valves and each pump's own isolation. The pumps discharge through and/or around the RHR heat exchangers and return the water to the vessel through the LPCI discharge lines.

Fuel Pool Cooling. During periods of high fuel pool heat load (core off load), either loop of RHR can be aligned to assist the Fuel Pool Cooling and Cleanup System (FPCCU) in removing decay heat. Operation in this mode is accomplished manually. With the fuel pool gates closed, a manual isolation valve is opened in the line leading from the fuel pool skimmer surge tank to the RHR pump combined suction line for shutdown cooling. The selected RHR pump discharge is directed through the RHR heat exchanger. Water returns to the fuel pool through another normally closed manual isolation valve.

Emergency Core Flooding. If all high pressure and low pressure water systems are inoperative and no other system is available to maintain vessel water inventory, emergency core flooding can be used. This mode uses the RHR service water system cross-tie to the B RHR system on the outlet of the B loop RHR heat exchanger. Two normally closed motor operated valves are opened and service water enters the vessel through the LPCI injection valves, piping, and recirculation loop.

RHR System Test. The RHR system shall be tested during plant operation by recirculating suppression pool water through the test return line. This testing excludes the Class I portion of the system. (The Class I portion of the system is that part of the RHR system which is part of the primary system pressure boundary between the reactor pressure vessel and the outboard containment isolation valves.) Functional testing and flow measurements of the system requiring injection into the reactor shall be accomplished during periods of reactor shutdown.

4.2.4.1 RHR Leak Detection.

Low Pressure Coolant Injection (LPCI) Mode of RHR. The RHR system serves several core, drywell, and wetwell cooling functions. The failure of the RHR LPCI pressure isolation valves can result in reactor coolant entering the low pressure components and possibly the torus and reactor building. The RHR system has a leak detection system and a LPCI loop selection logic that can detect and isolate possible ruptures. The system is designed to function for the cases in which coolant is expelled into the reactor building or torus.

The RHR leak detection system consists of area temperature sensors, sump level alarms, and high pressure annunciators. The detection of leaks in the RHR systems results in the annunciation of alarms in the main control room. The RHR system does not automatically isolate on the detection of a leak in the system. Emergency Operating Procedures are in place to diagnose and isolate leaks in the RHR corner rooms.

Temperature sensors are used to detect RCS leaks into the RHR corner rooms. The temperature sensors are located in the inlet and outlet of the ventilation ducts and in the inlet of the emergency area cooler of the RHR room. The sensors are mounted such that they are sensitive to ambient air temperature. The sensors are used to measure the temperature rise in the room and the room ambient temperature. The high room temperature and temperature rise associated with the RCS leak are annunciated in the main control room.

RHR leaks into the corner room are also detected by a sump level alarm. This alarm notifies the operations crew that a flooding condition may exist. This signal does not result in the automatic isolation of the RHR system. However, the plant EOPs address the RHR sump level alarms and direct the operations crew to isolate these leaks.

The RHR system also contains an alarm to warn the operators of high pressure in the injection line. This RHR alarm is set to annunciate when the pressure on the discharge side of the pump exceeds 435 psig or suction side pressure exceeds 135 psig. This alarm is designed to warn of pressure isolation valve leakage into the RHR system. The alarm does not automatically isolate the RHR system.

An intersystem LOCA through the RHR LPCI injection line could result in the rupture of the pump suction line. The LPCI suction line is aligned to the torus. If a suction line rupture occurred in the RHR corner room, the high temperature reactor coolant would result in an annunciation of three leak detection alarms. These alarms are (a) high ambient temperature, (b) high sump level, and (c) high injection line pressure.

The RHR system LPCI injection logic can automatically isolate an ISLOCA rupture in the affected train of LPCI if the operators fail to isolate the affected train of LPCI. The LPCI logic automatically selects the unbroken recirculation line for low pressure injection. The LPCI train affected by the ISLOCA event would be automatically isolated from the RCS.

A severe ISLOCA event through a LPCI train could result in RPV depressurization and a reduction in the reactor vessel coolant level. A severe ISLOCA would result in the automatic injection of core spray and LPCI. The LPCI is arranged for automatic and remote-manual operation from the main control room. Manual operation allows the operator to act independently of the automatic controls in the event of a LOCA. The two automatic initiation functions provided for the LPCI systems are low reactor vessel water level and high dry well pressure. Either of these conditions will initiate the LPCI system. The operating sequence for the LPCI system upon receipt of an initiation signal is as follows:

1. All four system pumps start with no time delay. Suction is taken from the suppression pool. The pump suction valve is normally open so that no automatic action is required to line up the suction path.

2. Valves used for other RHR operating modes are automatically positioned so that the water pumped from the suppression chamber is routed correctly.

3. The LPCI system injection valves automatically open when the reactor pressure has dropped to a value at which the LPCI pumps are capable of injecting water into the vessel.

4. The LPCI loops then deliver water to the reactor pressure vessel until the vessel water level is adequate to provide core cooling. The LPCI operation cannot be terminated for five minutes.

The LPCI actuation includes a recirculation loop-selection logic designed to prevent LPCI flowing into a broken recirculation loop. Specifically, the logic system compares the pressure in the two recirculation loops and automatically injects into the one with the higher pressure. To ensure accurate differential pressure readings (particularly in the case when only one recirculation pump is operating), a time delay is incorporated into the loop selection logic. This time delay is as follows:

1. A 0.50-s delay to determine if either recirculation loop is shut down. If one loop is shutdown, the other loop is also shut down.

2. A 2.0-s delay to allow momentum effects to settle and system parameters to stabilize.

3. A 0.50-s delay while the loop selection logic is being cycled.

The RHR heat exchanger bypass valve receives an open signal and a block close permissive (blocks the valve from closing) from the LPCI initiation signal. This RHR configuration is entered so that maximum flow is available for injection. After 3 min, this signal is blocked and the operator can manually close, throttle, or leave the valve in the open position.

Once the specific recirculation loop is selected for injection and the reactor coolant pressure is below the RHR overpressure interlock setpoint, the RHR outboard and inboard valve circuits for that loop receive an open permissive and a block close signal. The signal to the outboard valve is locked in for 5 min. This time is considered sufficient for the system to reflood the core to a least two-thirds of its height. Expiration of the 5-min lock-in period does not initiate valve closure but does give the operator the facility to throttle the flow.

The recirculation loop not selected for LPCI injection receives a close signal for 10 min when the loop selection is made. If the LPCI initiation signal remains, there is no capability in the logic to manually bypass the 10- and 5-min delays in the loop selection logic. Once the loop is selected, the operator cannot change the loops for 10 min. These automatic actions would isolate the ISLOCA rupture from the RCS in the affected train of LPCI. As a result, the ISLOCA event would be terminated by the automatic LPCI loop injection selection.

Flooding: RHR A and B Trains. An ISLOCA coolant pressure boundary rupture in the RHR A and B corner rooms can result in the flooding of the affected corner room. Flooding can result if the LPCI injection logic fails and the operations crew is unable to isolate the rupture. The RHR corner rooms contain a leak detection system. The leak detection system warns the operations crew of a potential flooding condition. Emergency Operating Procedures are in place to detect, diagnose, and isolate leaks that occur in the RHR corner rooms. These procedures enhance the ability of the operations crew to prevent flooding conditions from spreading to other portions of the reactor building.

The operations crew is also instructed to line up the RHR service water bypass to provide long-term core cooling in the event of loss of torus and condensate storage water. This coolant supply provides a long term supply of coolant water to the reactor. With this coolant supply as the flooding source, the flood can continue to propagate throughout the reactor building, the auxiliary building, and the turbine hall.

If the leak is not isolated, the flood will propagate upwards in the RHR corner room. The flood can continue upward from the subbasement through the basement region until it reaches the first floor. The flood does not propagate to the basement region of the reactor building from the RHR corner rooms. From the RHR corner rooms, the flood will enter the rail car and truck loading area of the reactor building. The flood entrance path is the stairwell leading down to the RHR corner room's subbasement. The water then spreads laterally on the first floor. The flood will enter the RHR A and B heat exchanger rooms and also drain into the core spray and RCIC/CS corner rooms. As the flood level increases in the core spray room, the flood enters the CRD pump room at the basement level of the reactor building.

From the CRD pump room, the flood then enters the HPCI room through an open equipment hatch and stairwell. After the RCIC/CS corner room is flooded to the basement level the flood enters the auxiliary building through an open doorway. After the HPCI room is flooded, the flood will also propagate into the auxiliary building through an open door in the CRD pump room. From the auxiliary building the flood will enter the turbine building through a 3-h fire doorway.

The loading dock and personnel entrances from the grade level to the first floor are equipped with watertight doors. These doors were designed to prevent external flood waters from entering the reactor building and will prevent the internal flood from leaving the reactor building to the environment at grade level.

Shutdown Cooling Mode of RHR. The RHR system serves to remove decay heat and heat stored in the vessel components during shutdown conditions. The shutdown cooling mode of RHR is utilized to reduce the reactor temperature to 125°F within 20 h from reaching 95 psig.

The RHR system valves are manually aligned to produce the shutdown cooling mode. The system alignment takes suction from the B recirculation loop. The flow enters the RHR pumps, which then discharge the flow through the RHR heat exchangers and return the water to the vessel through the LPCI discharge lines.

The RHR system, when it is being aligned and utilized for shutdown cooling, can allow the vessel to drain into the torus if proper valve alignment is not maintained. There have been 10 events in which the vessel of a BWR has drained to the torus during shutdown.[25] In each event, rapid draining of the reactor vessel occurred. The events were terminated by proper operator actions before the fuel was uncovered and damaged. These events have been considered as precursor events to an interfacing system LOCA. The vessel draining event is not the typical ISLOCA event in which a pipe ruptures at high pressure. The draining condition can allow the fuel in the vessel and spent fuel pool to uncover by loss of coolant through the interfacing system. These events are protected by valve interlocks. It is possible that these interlocks can be bypassed during plant shutdown or refueling because they are not protected by technical specifications at these times. Also, the RHR valve interlocks do not extend across Division I and II boundaries. The applicable interlocks are briefly described below.

1. An interlock has been provided between the RHR pump shutdown cooling suction isolation valve (Valves F006A, B, C, and D) and the RHR pump suction valve (Valves F004A, B, C, and D). This interlock prevents opening the F006 valve until the F004 valve is shut. The interlock is extended to prevent reopening the torus suction valves once the RHR pump shutdown cooling suction isolation valves are open.

2. There is one interlock associated with the torus warm up valve F028A/B and the vessel level. This interlock allows vessel draining until the reactor vessel water level reaches Level I. Upon reaching Level I the interlock closes the torus warm up valve F028A/B.

3. An interlock between the RHR pump shutdown cooling isolation valves in one loop, F006A/C or F006B/D, and the torus spray outboard isolation/test valve in the same loop (F028A/B), prevents opening F006 unless the F028 valve in that loop is shut. *No interlock is provided between the valves in the opposite loop.* The interlock prevents reopening the F028 valve until the associated F006 valve is closed.

4.2.4.2 RHR Suction Line Analysis. From the table above, the only components of concern (i.e., >1E–3 probability of rupture) are the shutdown suction (letdown) line and the pipe connecting the pump suction to the suppression pool (SP).

The SP line is normally isolated from the RCS by three normally closed MOVs (F009/F608, F008, and F006D/B/C/A) on one end and is open to the suppression pool on the other end (normally open MOV F004D/B/C/A, which is operated by a keylock switch and is interlocked with the F006 valve such that one cannot be opened if the other is already open). One of the MOVs (F008) is normally kept deenergized for Appendix R requirements (within 4 h of securing shutdown cooling, the fuses must be removed). Valves F008 and F009 are interlocked to automatically close if the RPV water level drops below Level-3 (193.4 ins.), or the RPV pressure rises above 89.5 psig (shutdown cooling interlock pressure). Valve F608 is operable only via a keylock switch, and valves F009 and F008 are interlocked to prevent opening if the RPV pressure is greater than 89.5 psig. Therefore, a number of events must occur in order to overpressurize this line. Valve F008 must be energized (fuses installed), and the RPV pressure interlock defeated. Valve F608 key-lock switch must be operated or F009 interlock defeated and the valved opened. While it is feasible to bypass the interlocks, indications are that this is not routinely done even during shutdown. The following is taken from the plant training manual for the RHR system.

> "Technical Specification allow the isolation signal that closes the shutdown cooling inboard and outbound isolation valves on low reactor water level to be bypassed in the cold shutdown and refueling modes. This is because the containment isolation function of the valves is not needed in these modes. However, to ensure protection against inadvertent vessel draining, the low reactor water level isolation signal and valve interlocks should remain in effect during all modes of operation.
>
> "If exceptions arise that make it necessary to bypass the isolation signal or the valve interlocks, contingency measures should be taken in the form of having other emergency core cooling system trains, such as one or more low pressure core spray trains, available to provide water to the vessel. Calculations should be performed to confirm that the rate of draining through the largest potential drain path would not exceed the rate of addition by the emergency core cooling system equipment. Special monitoring and preplanned isolation actions should also be considered. Nuclear Safety Analysis Center Report 88 addresses other contingency measure that should be considered. In addition, senior plant management review and approval should be obtained before bypassing any isolation signals or interlocks that protect against vessel draining."

Assuming only internal rupture to be a credible failure mode for these MOVs (because of interlocks), a bounding calculation yields the following:

Frequency of RHR ISLOCA Initiating Event

$$= F009 + F608) * F008$$

$$= (MOV\text{-}NCFO + MOV\text{-}NCFO)$$

$$* MOV\text{-}NCFO$$

$$= (1E\text{–}7/hr * 8760\ hr/yr + 1E\text{–}7/hr$$

$$* 8760\ hr/yr) * 1E\text{–}7/hr * 8760/2\ hr/yr$$

$$= 7.7E\text{–}7/yr\ .$$

If in 50% of the occurrences F006 MOV is closed and 50% of the occurrences F006 is open (and F004 is closed). Then

Frequency of RHR rupture $= 7.7E\text{–}7/yr$

$$* (0.5 * 0.074 + 0.5 * 0.023)$$

$$= 3.7E\text{–}8/yr\ .$$

Allowing a 90% success rate for operator recovery and isolation of ruptures (because of the leak detection system and good procedures) yields a bounding estimate for core damage frequency of 3.7E–9/yr. Therefore, the RHR system is judged to not be of concern with respect to ISLOCA.

Concern has also been raised about the possibility of backleakage through the LPCI injection line.[26] In this scenario, a loss and subsequent restoration of power causes spurious engineered safety feature (ESF) actuation. The ESF actuation causes the LPCI injection MOVs to open, thereby reducing the redundancy in the PIB and increasing the chance of overpressurizing and possibly rupturing some low pressure rated LPCI components. Although an inadvertent ESF actuation along with the failure of the testable check isolation valve (F050B/A) would allow high pressure RCS water to enter the LPCI system, based on the information presented in Table 6, no ruptures would be expected. In order to produce the possibility of a rupture the pump discharge check valve (F031D/B/C/A) would also need to fail. This would allow backleakage into the pump suction line. Estimating the frequency of an inadvertent ESF actuation that opens the LPCI injection MOV at 0.01 (based on four events in about 500 BWR-yrs), the following bounding calculation is made:

$$\begin{aligned}
\textit{Frequency of PIB failure} &= F050B * F015B \\
&\quad * F031D/B/C/A \\
&= TCV\text{-}NCFO * ESF\text{-}act \\
&\quad * (4 * CV\text{-}NOFO) \\
&= 9E\text{–}3 * 0.01/yr * (4 * 1E\text{–}3) \\
&= 3.6E\text{–}7/yr\,.
\end{aligned}$$

$$\begin{aligned}
\textit{Frequency of rupture} &= 3.6E\text{–}7/yr * 0.074 \\
&= 2.7E\text{–}8/yr\,.
\end{aligned}$$

Allowing a 90% success rate for operator recovery and isolation of ruptures (because of the leak detection system and good procedures) yields a bounding estimate for core damage frequency of 2.7E–9/yr. Therefore, the LPCI system is judged not to be of concern with respect to ISLOCA.

4.2.5 Reactor Water Cleanup (RWCU). The RWCU system continuously purifies the reactor water. The system removes water from the suction line of each recirculation pump and from the reactor bottom head and returns it to the feedwater system. Water may also be sent to the main condenser (preferably) or the radwaste system. A regenerative heat exchanger is provided to maintain thermal efficiency. The major equipment of the RWCU system, located in the reactor building, includes pumps, regenerative and nonregenerative heat exchangers, and two filter-demineralizers with supporting equipment. A simplified diagram of the system is shown in Figure 7.

Reactor water is cooled in the regenerative and nonregenerative heat exchangers, then filtered, demineralized, and returned to the reactor feedwater system through the shell side of the regenerative heat exchanger. Because the maximum temperature of the filter-demineralizer units is limited by the ion exchange resin operating temperatures, the reactor coolant must be cooled before being processed in the filter-demineralizer units. The regenerative heat exchanger transfers heat from the influent water to the effluent water. The nonregenerative heat exchanger cools the influent water further by transferring heat to the reactor building closed cooling water (RBCCW) system. The nonregenerative heat exchanger is designed to maintain the required filter-demineralizer operating temperature, even when the effectiveness of the regenerative heat exchanger is reduced by diversion of excess reactor water from the filter-demineralizer effluent to either the main condenser or the radwaste system.

The suction line of the reactor coolant pressure boundary (RCPB) of the RWCU system contains two motor-operated isolation valves which automatically close in response to signals from the RCPB leak detection system (see below). The outermost isolation valve also automatically closes to prevent removal of liquid poison in the event of standby liquid control system actuation and to prevent damage of the filter-demineralizer resins if the outlet temperature of the nonregenerative heat exchanger is high. These isolation

Table 6. RHR system design specifications.

Component	Design pressure (psig)	Design temperature (°F)	Material	Dimension	Median pressure fragility[a] (psi)	Rupture probability[b]
Class 1 piping, pump discharge	1500	575	900-lb carbon steel	24 in. Sch. 80	6,222	Negligible
Class 2 piping, pump discharge	480	335	300-lb carbon steel	3 in. Sch. 40 4 in. Sch. 40 20 in. Sch. 40 24 in. Sch. 40	7,155 6,036 3,411 3,301	Negligible Negligible 4E–4 6E–4
RHR pumps	500	360	Cast steel	Single stage, vertically mounted centrifugal (10,000 gpm at 256 psid)	>>1,000	Negligible
RHR heat exchangers[a]	450 675[b]	470	Carbon steel (stainless steel tubes)	Vertically mounted shell-and-tube	2,985 2,830	4E–6 8E–5
RHR HX tube sheet flange	450	470	Bolts: ASTM A193 GrB7, gaskets: flexitallic	64 1-1/4-in. dia. studs torqued to 1000 ft-lb	1,760[c]	5E–5
Pump suction line	150	335	300-lb carbon steel	20 in. Sch. 40	3,411	4E–4
Shutdown suction line	150	335	150-lb carbon steel	20 in. std 3/8 in. wall thickness	2,065	0.023
Suppression pool to pump suction line	150	335	150-lb carbon steel	24 in. std 3/8 in. wall thickness	1,710	0.074

a. Both shell side (primary) and tube side (service water) of heat exchangers are designed to 450 psig. Heat exchanger failure pressures are given in Appendix B.

b. Hydrotest.

c. Pressure required to produce a leak area of 2 sq. in. (i.e., 1.5 times gross leak pressure). Assumes initial bolt stress of 55,000 psi and joint relaxation of 15%. Tabulation of stresses is given in Appendix B.

valves may be remote manually operated to isolate the system equipment. A remote manually operated gate valve on the return line to the reactor provides long-term backup isolation of the system for the reactor. Instantaneous reverse-flow isolation is provided by two check valves in the RWCU return line (in addition to one feedwater system check valve).

4.2.5.1 RWCU Leak Detection System. The RWCU system is designed for high pressure, high temperature operation. The safety of the

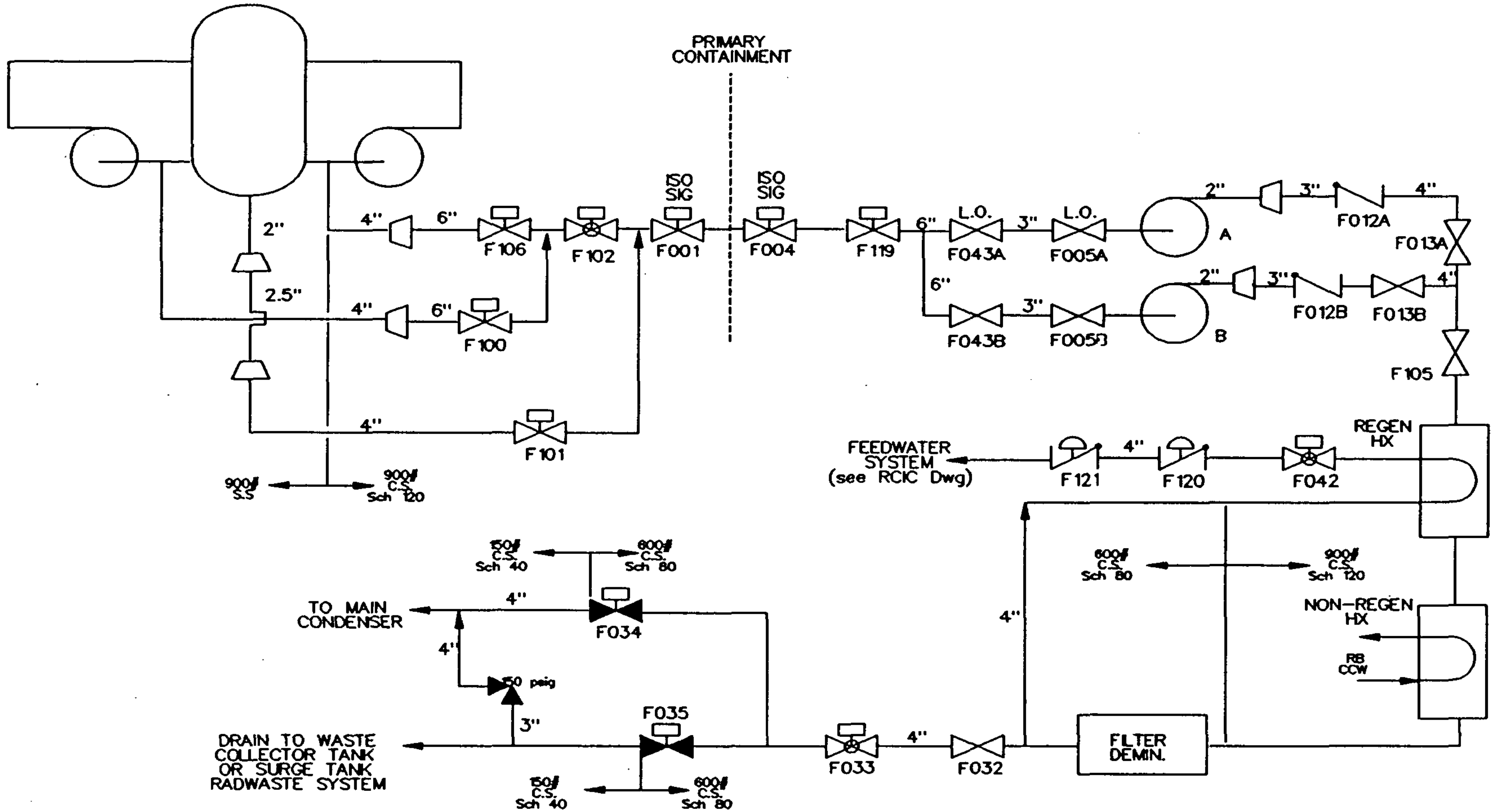

Figure 7. RWCU simplified diagram.

system is enhanced by a leakage detection system (LDS). The limiting pressure boundary is the return line to the chemical radioactive waste tank and the shell side of the nonregenerative heat exchanger. If a failure were to occur in either of these areas, the LDS is designed to automatically isolate the RWCU from the RCS. This isolation would terminate the intersystem LOCA without the need for operator actions.

The automatic isolation signals are initiated by the leakage detection system. The leakage of the reactor coolant into the RWCU compartments is detected by several means, which are: (a) compartment temperature measurements, (b) inlet and outlet flow comparisons, and (c) sump level measurements.

The RWCU process removes chemical impurities from the high temperature reactor coolant. Leakage from the RWCU process lines into the surrounding compartment results in an increase in the temperature of the compartment gas volume. The leakage detection system makes use of the increase in the compartment gas temperature to detect leaks. The leakage detection system uses temperature sensors located in the inlet and outlet ventilation ducts to determine leakage. These sensors measure the temperature rise in the RWCU's compartments. Local ambient temperature sensors are located in all compartments containing equipment for the RWCU. If the temperature sensors show an increase in compartment temperature above normal operating temperatures, alarms in the main control room annunciate. In addition to this annunciation, the temperature rise results in automatic isolation of the RWCU from the reactor coolant system.

The temperature dependent leakage detection system is complemented by RWCU process flow measurements. The additional system is necessary since part of the RWCU process uses low temperature coolants. Reactor coolant system leakage into the RWCU compartments is detected by comparison of the inlet and outlet coolant flows to the system. The leakage alarm in the control room annunciates and the RWCU system automatically isolates when the difference in outlet and inlet flow rate exceeds 55 gpm.

An intersystem LOCA through the RWCU system could result in high temperature RCS coolant entering one of the RWCU compartments. The high temperature coolant would result in an increase in the compartment's ambient temperature. This high ambient temperature would result in the automatic isolation of the RWCU system from the RCS. The rupture would also be detected by a difference in the inlet and outlet flow rates. This flow difference, if more than 55 gpm, would also result in the automatic isolation of the RWCU.

4.2.5.2 RWCU Analysis. No portions of the RWCU system are of concern with respect to rupturing when exposed to full reactor pressure (see Table 7).

4.2.6 Control Rod Drive (CRD) Hydraulic System. The CRD hydraulic system supplies pressurized water to operate and cool the control rod drive mechanisms during normal operation. This system implements a scram command from the reactor protection system and drives the control rods rapidly into the reactor. The CRD can also provide makeup water to the RCS. A simplified diagram is shown in Figure 8 and CRD system design specifications are given in Table 8.

During normal operation the CRD pumps provide a constant flow for drive mechanism cooling and system pressure stabilization. Excess water not used for cooling is discharged to the RCS. Control rods are driven in or out by the coordinated operation of the direction control valves. Insertion speed is controlled by flow through the insert speed control valve. Rod motion may be either stepped or continuous.

A reactor scram is implemented by pneumatic scram valves in the CRD system. An inlet scram valve opens to align the insert side of each control rod drive mechanism to its scram accumulator. An outlet scram valve opens to vent the opposite side of each CRD mechanism to the dump tank (or discharge volume). This coordinated action results in rapid insertion of control rods into the

Table 7. RWCU system design specifications.

Component	Design pressure (psig)	Design temperature (°F)	Material	Dimension	Median pressure fragility[a] (psi)	Rupture probability[b]
RWCU pump suction (RC letdown)	1250	575	Carbon steel (900 lb)	3 in. Sch. 160 4 in. Sch. 120 6 in. Sch. 120	17,840 >9,279 8,009	Negligible Negligible Negligible
RWCU pump discharge (to filter/demineralize)	1300	575	Carbon steel (900 lb)	2 in. Sch. 160 3 in. Sch. 160 4 in. Sch. 120	21,417 17,840 >9,279	Negligible Negligible Negligible
WCU pumps (2)	1400	564	—	@ 180 gpm	>>1,020	Negligible
Regenerative heat exchanger (3 in series)			—	133,000 lb/h flow rate		
- Shell side	1450	564			>>1,020	Negligible
- Tube side	1450	564			>>1,020	Negligible
Nonregenerative heat exchanger (2 in series)			—	133,000 lb/h flow rate		
- Shell side	150	370			NA	NA
- Tube side	1450	564			(RBCCW) >>1,020	Negligible
Filter-demineralizer (2 in parallel)	1400	150	—	@ 64,000 lb/h flow rate	>>1,020	Negligible
Lines from F/D to drains and shell side of regenerative heat exchanger	1300	150	Carbon steel (600 lb)	4 in. Sch. 80	9,279	Negligible
Drain lines to main condenser and radwaste system	150	150	Carbon steel (150 lb)	4 in. Sch. 40	6,036	Negligible

a. Median pressure fragility data for carbon steel pipe assumed a corrosion allowance of 0.02 in. and a temperature of 600°F.

b. Rupture probability calculated using an internal pressure of 1,020 psi.

reactor. At the reference BWR, the Alternate Rod Insertion (ARI) system can also initiate a reactor scram.

Although not intended as a makeup system, the CRD system can provide a source of cooling water to the RCS during vessel isolation. The maximum RCS makeup rate of the CRD system is about 200 gpm with both pumps operating.

4.2.6.1 CRD Leak Detection. The CRD system is not covered by the plant leak detection system.

4.2.6.2 CRD Analysis. If any overpressure induced ruptures were to occur in those portions of the CRD system susceptible to failure, the resultant leak would be limited to the size of the 3/4- or 1/2-in. pipes between the rupture and the

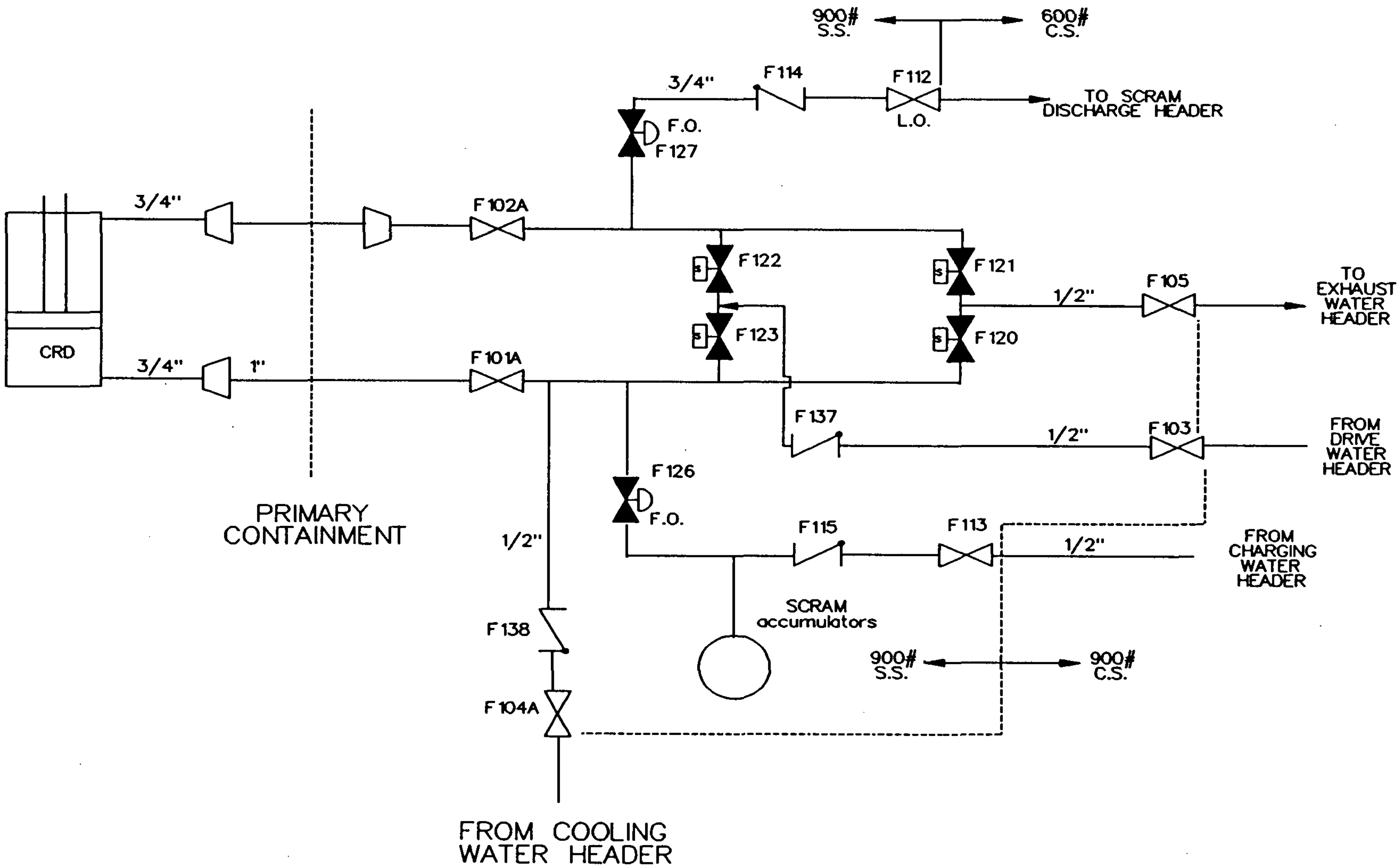

Figure 8. Simplified diagram of the CRD system.

Table 8. CRD system design specifications.

Component	Design pressure (psig)	Design temperature (°F)	Material	Dimension	Median pressure fragility[a] (psi)	Rupture probability[b]
Insert lines withdraw lines	1750	150	900-lb stainless steel, 900-lb carbon steel	1/2 in. Sch. 160 3/4 in. Sch. 160 1 in. Sch. 160	>17,292 (SS) >21,904 (CS)	Negligible Negligible
Control rod drive pumps (2)	1500	150	Cast steel	120 gpm @ 1500 psid centrifugal	>>1,000	Negligible
CRD pump suction line	250	150	150-lb carbon steel	3 in. Sch. 40 4 in. Sch. 40 6 in. Sch. 40	7,155 6,036 4,801	Negligible Negligible Negligible

a. Median pressure fragility data for carbon steel pipe assumed a corrosion allowance of 0.02 in. and a temperature of 600°F.

b. Rupture probability calculated using an internal pressure of 1,020 psi.

RCS. Therefore the CRD system is judged not to be a concern with respect to ISLOCA risk.

4.3 Results

ISLOCA is not a risk concern for the BWR plant examined here. Although portions of the interfacing systems are susceptible to rupture if exposed to full RPV pressure, these are typically pump suction lines that are protected by multiple valves (i.e., two PIVs and the pump discharge check valve). In addition, these lines are open to atmospheric pressure (either the suppression pool or the CST). This feature will limit the amount of pressurization that can take place. Generally, the BWR systems benefit from a much higher relative pressure margin than PWR systems because of the much lower operating pressure of the BWR RCS, about 1,000 psi for BWRs as contrasted with about 2,000 psi for PWRs.

5. PWR AUXILIARY BUILDING ANALYSIS

5.1 Introduction

The ability of plant operators to recover the plant after an ISLOCA depends, among other factors, on the availability of safety injection systems following the break. ISLOCAs are characterized by the blowdown of reactor coolant outside of containment. This generally means inside of the auxiliary building. Equipment in the auxiliary building may or may not be qualified to the environmental conditions created by the break. This is expected to vary widely among operating reactor plants. Some plants will have safety equipment qualified to more severe conditions than can be expected to result from an ISLOCA, while equipment in others may not be qualified beyond the normal operating design basis.

Because auxiliary building designs vary widely, the question naturally occurs as to how the B&W plant that was previously analyzed[1] compares with other plants. Is it more, or less, vulnerable to an ISLOCA than a typical auxiliary building design? If it is typical, is there a plant design that bounds most of the operating plants. To address this question, existing auxiliary building designs were surveyed (by using the insights gained from the B&W plant analysis) to identify the plants that are potentially most vulnerable. A plant was then selected from those considered most vulnerable for detailed thermal hydraulic analysis. The thermal hydraulic analysis was performed to determine equipment survival times assuming an ISLOCA blowdown like that calculated for the B&W plant. The following sections describe the approach used.

5.2 Auxiliary Building Survey

The goal of the survey was to identify the PWR auxiliary buildings that have the greatest potential for propagation of the environmental effects that result from an ISLOCA event. The environmental effects of concern are primarily flooding, but also include high humidity and high temperature. A detailed evaluation procedure is illustrated by the flow chart in Figure 9. Because of the large number of operating PWRs and the amount of information required to quantify the vulnerability of a given auxiliary building, a two-level screening approach was used. The first level was largely subjective. The insights gained from previous steam propagation studies were used to eliminate plants with obvious mitigating features. Plants without mitigating features were studied in more detail, applying as much of the procedure shown in Figure 9 as possible with the available information.

5.2.1 Available Information. The survey required information on the location of ECCS components within the auxiliary building. It also required information about drainage paths, steam flow paths, secondary water supplies (such as firewater spray), and equipment qualification. Some of this information was available in NRC *Nuclear Power Plant System Sourcebooks*.[g] The rest is sometimes available in FSARs, high-energy line break (HELB) analyses, and drainage reports. The first portion of the survey was based largely on the NRC sourcebooks. These were readily available for 42 of the 76 licensed PWR units and contained excellent information on the location of safety related equipment within the plant. This information was supplemented by FSARs and operator training materials when available. The plants surveyed and the information sources used are shown in Table 9. The plants were reviewed using the flooding and temperature criteria described below.

5.2.2 Flooding Criteria. The flooding potential of each plant was evaluated based on the presence of water flow paths between the point of interfacing system rupture and the initially unaffected ECCS trains. These flow paths may be pipe

g. These are a series of reports developed by Science Applications International Corporation under contract to U.S. NRC Office of Reactor Regulation. Sourcebooks available for most but not all U.S. nuclear power plants. They can be obtained from the U.S. NRC Public Document Room.

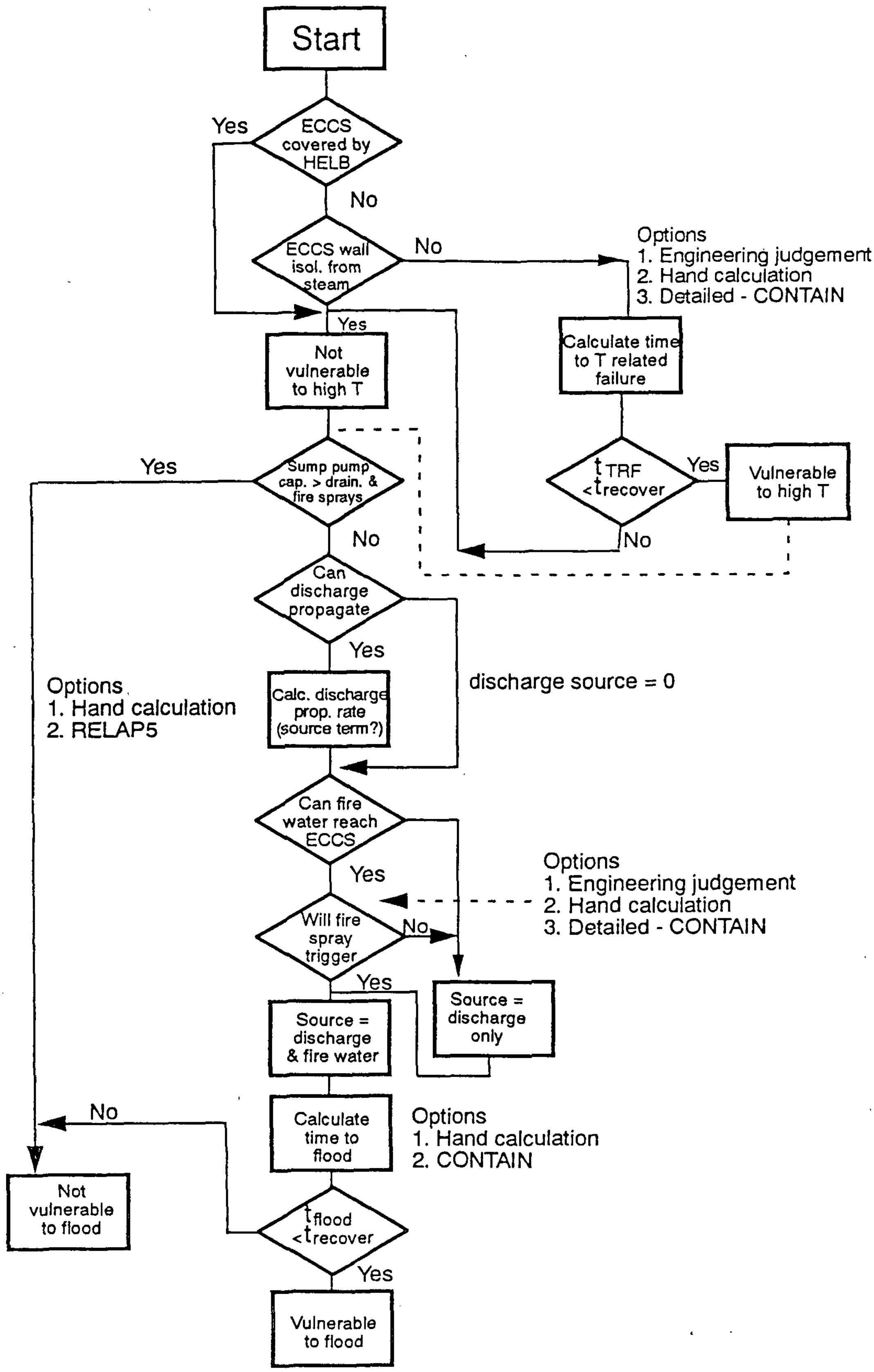

Figure 9. Flow chart for auxiliary building evaluation procedure.

Table 9. List of pressurized water reactor plants surveyed and the information sources used.

Plant name	Sources	Plant name	Sources
Beaver Valley 2	a	Millstone 3	a
Beaver Valley 1	a	Millstone 2	a
Bellefonte 1 and 2	a	Palo Verde 1, 2, and 3	a
Braidwood 1 and 2	a	Rancho Seco	a
Byron 1 and 2	a,b	Seabrook 1	a
Callaway	a,c	Shearon Harris	a
Catawba 1 and 2	a	South Texas 1 and 2	a,b
Comanche Peak 1 and 2	a,b	St. Lucie 1 and 2	a,b
Davis-Besse	a	Three Mile Island 1	a
Fort Calhoun 1	c	Trojan	a,b
Ginna	a,b	Vogtle 1 and 2	a
H. B. Robinson	a	Waterford 3	a,b
Haddam Neck	a	Wolf Creek	c
Indian Point 3	a	Yankee Rowe	a
McGuire 1 and 2	a	Zion 1 and 2	a

Sources:

a = NRC source book

b = FSAR

c = Operator training notes

42 units covered by sourcebooks.

34 units for which we do not have sourcebooks.

chases, stairwells, or open doorways. In reviewing the source books, it appeared that categorizing the buildings as having either spatially diverse or localized placement of ECCS components would provide a good indication of flood potential. Plants with spatially diverse systems are less vulnerable because larger volumes of water will be required to cause flooding. Also, many plants have essential components on separate levels of the building, or widely separated when on the same level of the building. These are plants that were identified as relatively resistant to flooding. Plants with the ECCS located in a single small space in a low level of the building were identified as potentially vulnerable to flooding.

The presence of pipe chases, doors, and stairwells was also considered. For instance, if equipment is located on a high level in the building, and drainage paths are available to lower levels, the essential equipment can be expected to survive for long periods of time. However, a break at a higher level could cause flooding of essential equipment rooms below. Most personnel doors will not hold more than 2 psi pressure differential, the amount provided by a water depth of 5 or 6 ft. Before the doors fail, flooding caused by leakage past the door would likely be prevented, or at least be delayed, by sump pumps. After failure this is very unlikely. Doors close to the break may blow out as a result of overpressure, and doors far from the break are likely to survive. The status of many doors could not be evaluated with certainty from the available information. The doors may routinely be left either open or closed. If some doors are assumed closed in a HELB analysis, or if they are a part of the negative pressure boundary, they may be required to be closed when not in use. Neither can be determined from the information sources used in the survey.

Previous ISLOCA work has shown the most important system with respect to flooding is the LPI or RHR portion of the ECCS. Previous analyses have shown these are both the most likely to be overpressurized, and are the most susceptible to failure when pressurized. Furthermore, the break size will be the largest for any postulated ISLOCA event. This will mean the greatest flood rate will be produced by a failure in one of these systems. Also, the pumps for these systems are often located at the lowest level of the auxiliary building. This is often a small space, resulting in the maximum rate of water level rise. Also, the compartments in which these pumps are located may receive drainage from many other compartments. This drainage can be through floor drains that empty to the sump, or from pipe chases that are usually present in the upper levels. This provides the potential that any break, no matter where, can contribute to flooding these systems. When these systems are located at a higher level in the building, these same drainage paths will tend to mitigate the event instead of aggravating it. The worst possible situation will occur when all ECCS is located low in the building in a relatively small space that receives drainage from higher levels in the building. This would cause all condensate, firewater, and break discharge to collect, flooding the ECCS pump motors.

Another consideration is possible flooding of power distribution system components, service water or component cooling water components, or other essential support system components. In some plants, these are located near or below the LPI/RHR systems. In plants where this situation exists, there is a very high potential for widespread system failures resulting from flooding.

5.2.3 Temperature Criteria. It was very hard to screen for temperature vulnerability on the basis of information provided in the source books. The only way to provide a good evaluation is to review the equipment qualification data. This data is not readily available. Some conclusions were made using the same arguments as for flooding. Plants with spatially diverse systems were considered more resistant than those with localized placement of ECCS equipment. If diverse, there is at least some chance that doorways will hold and provide protection and that blowouts will prevent excessive temperature buildup in the spaces. If the steam environment is minimal because the equipment is far from the break, fire spray systems may provide sufficient cooling to prevent failure. Finally, if auxiliary feed equipment is located in the auxiliary building, there is a reasonable chance that HELB analysis may have resulted in equipment qualification to temperatures higher than those expected to result from an ISLOCA.

Because of the speed with which the steam environment can propagate, there is the potential that nearly all electrical systems within a plant that has no equipment qualification could be degraded within minutes.

5.3 Survey Results

After the first screening 12 plants remained that were considered potentially most vulnerable to the environmental conditions resulting from an ISLOCA. There is no clear ranking of these plants from most vulnerable to least vulnerable. The classifications shown in Table 10 suggest, instead, that the plants in Classes 1, 2, or 3 should all show a similar auxiliary building response. After reviewing the information on these plants, it seemed clear that one plant (hereafter referred to as the "Case-2 plant"[h]) had features that suggested a greater vulnerability than the other plants (the single Class-1 plant in Table 10). Therefore the Case-2 plant was contacted and agreed to provide the information required to perform the auxiliary building sensitivity study described in the following section. The results of this analysis were used in the cost/benefit calculations that support and are documented in the regulatory analysis for the resolution of Generic Issue 105 (NUREG-1463, *Regulatory Analysis for Resolution of Generic Issue-105: Interfacing System Loss of Coolant Accident in Light Water Reactors*).

h. Case-2 plant refers to the sensitivity study nature of this particular analysis. Specifically, Case-1 represents the three PWR analyses of B&W, CE, and W plants (documented in NUREG/CR-5604, NUREG/CR-5745, and NUREG/CR-5744, respectively) with plant-specific auxiliary building designs. Case-2 is a sensitivity study that mated this "Case-2 plant" auxiliary building with the B&W plant examined previously.

5.4 Auxiliary Building Environmental Sensitivity Cases

This section describes a sensitivity calculation[i] in which the B&W plant ISLOCA blowdown (Reference 1) is applied to the Case-2 plant auxiliary building. The objective of this calculation is to determine the environmental conditions in the Case-2 plant auxiliary building for the five break scenarios determined by the B&W plant ISLOCA analysis (Reference 1).

5.4.1 Method of Calculation. The CONTAIN computer code was used to calculate the time response of the auxiliary building parameters stated above. The CONTAIN models are intended to be best estimate models. Based on the auxiliary

i. The calculation is documented by J. A. Schroeder using the CONTAIN computer code (Version 1.12, Rev. 2) in "Calculation Package for the INEL ISLOCA Program Steam Propagation Calculation for a B&W Plant," May 1992.

Table 10. Plant categorization results.

Class	Characteristics	Number of units in class
1	Moderate floor area, little compartmentalization, all equipment on the lowest level.	1
2	Large floor area, significant compartmentalization but consisting of weak or partial barriers, all equipment on lowest level.	3
3	Small floor are, compartmentalization inadequate to prevent flood propagation, all ECCS on lowest level but some injection (charging) on higher level.	5
4	These plants contain elements in more than one of the above classes. These plants were individually evaluated and judged to not represent a bounding case.	3

building plan and section drawings provided by the utility staff, the auxiliary building can be adequately modeled by treating the RHR pit, basement, intermediate floor, and operating floor as separate control volumes (cells) as shown in Figure 10. Accurate representation of the volume, drainage paths, gas flow paths, heat structures, and fire spray systems in each cell should then produce the desired best estimate model. The auxiliary building steam sources resulting from the ISLOCA were calculated in a previous analysis[27] and were not updated for this analysis.

5.4.2 Model Description. The modeling data required for this analysis consists of the gas volume dimensions, gas flow path dimensions, heat structure surface areas and masses, drain system capacities, and fire spray flow data. The following sections describe the auxiliary building features included in the model and the data and data sources used to build the model.

Compartment Gas Volume Data. The essential gas volume information includes the elevation of each floor, the distance from floor to ceiling, the total floor area, and the total free volume. The volume information for this section has been obtain from the Case-2 plant auxiliary building plan and section drawings, without the benefit of a plant visit. Therefore, modeling details that could not be resolved on a best estimate basis were bounded with known information (e.g., if a floor area was not clearly discernible, it was bounded with the area enclosed by the outside walls). This does not in every instance lead to a conservative model, but it is best estimate in that it uses the best available information.

The RHR pit is the lowest part of the auxiliary building. It contains the RHR pumps and a sump to collect drainage from the basement floor. The floor is approximately 23 feet by 21 feet, and the room is about 15 feet high. The RHR pit is connected to the basement by a ladder hatch and by removable equipment hatches. Both of these are equipped with a curb to prevent water spilling into the pit from the basement. Water flow paths into this compartment are discussed in more detail in later sections.

The RHR pit presents modeling difficulties because the room will become completely submerged in some scenarios. CONTAIN will not allow a gas volume to become completely filled with liquid, so in break scenarios where this occurred it was necessary to adjust the volume upward so that the compartment would not fill. Extending the volume height to the top of the basement floor volume allowed the calculation to run to completion for each scenario. The impact of this on the overall model is slight since the RHR pit volume is only about 10% of the basement volume. This will affect rate of temperature rise slightly, but not the rate of flooding since the floor areas are correct.

The basement volume was calculated based on the area enclosed by the exterior walls. Past analyses have shown that occupancy can be about 20% of the volume, but without supporting data no attempt was made to account for occupancy of

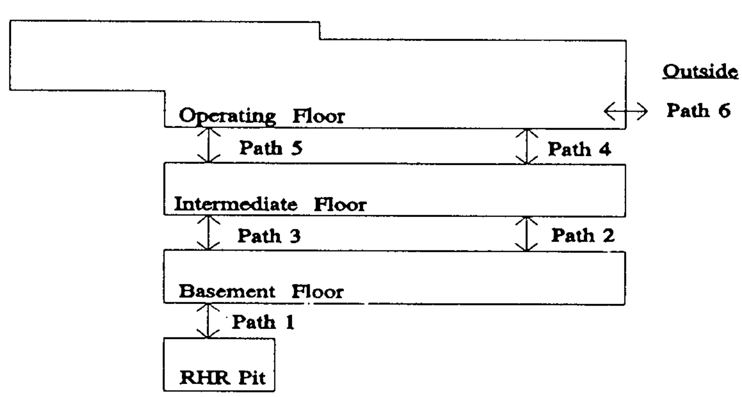

Figure 10. CONTAIN nodalization used to model the auxiliary building.

the interior space by equipment and support walls.

The volume of the intermediate floor was bounded by the volume enclosed within the outer walls of the auxiliary building at the 253′-0″ level. Again, no reduction was made for the area occupied by internal walls, tanks, and other structures.

The operating floor includes two elevations, the main floor and the deck at the top of the fuel storage pool. The roof also is built in two sections at different elevations. The resulting volume is irregular and not easily described by a single floor area and volume height. The total volume was calculated,[j] and the average height obtained by dividing the volume by the floor area.

The outside volume and area are set large to provide a constant pressure and temperature boundary condition.

Blowdown Data. The reactor system blowdown was obtained from a previous calculation[1] The data were obtained using simplified RELAP models of a B&W reactor plant. The plant was a two raised loop 2772 MWt system with a reactor coolant system volume of 11,440 ft^3. The two low pressure injection system pumps had a rated flow of 3000 gpm each. The two high pressure injection system pumps were rated at 500 gpm each. The Case-2 plant is a two-loop Westinghouse plant rated at 1520 MWt. The two low pressure injection system pumps in this plant have a rated flow of 2500 gpm each. The three high pressure injection system pumps are rated at 300 gpm each. At the beginning of the blowdown the steam source is driven by the RCS size and energy content, and later the blowdown becomes a function of the available injection capacity.

Break Sequence 1 (BS-1) is a rupture in the decay heat removal (DHR) system piping. The plant is operating at rated power and normal operating temperature and pressure. Isolation of the DHR letdown line fails, causing pressurization of the DHR system, which results in rupture of a 12-in. line and discharge into auxiliary building at the basement level.

Break Sequence 2 (BS-2) is a rupture in the decay heat coolers. The plant is operating at rated power and normal operating temperature and pressure. Isolation of the DHR letdown line fails, causing pressurization of the DHR system, which results in simultaneous rupture in both decay heat removal heat exchangers and discharge into the basement level of the auxiliary building. The limiting flow area is in the 2.5-in. bypass lines around the DHR letdown line isolation valves.

Break Sequence 3 (BS-3) is a rupture in the suction lines to the DHR pumps. The plant is operating at rated power and normal operating temperature and pressure. Isolation of the DHR letdown line fails, causing pressurization of the DHR system, which results in simultaneous rupture in the low-pressure decay heat removal pump suction piping, resulting in discharge to the auxiliary building at the basement level. The limiting flow area is in the 2.5-in. bypass lines around the DHR letdown line isolation valves.

Break Sequence 4 (BS-4) is a rupture in a DHR cooler. This sequence involves backflow from the RCS through the decay heat cooler discharge piping. The plant is operating at rated power and normal operating temperature and pressure. The check valve isolation on the injection side of the LPI system fails, resulting in a rupture at the decay heat cooler with discharge into the basement level of the auxiliary building.

Break Sequence 5 (BS-5) is a rupture in the high pressure injection (HPI) pump suction piping. This sequence involves backflow from the RCS through the HPI discharge piping. The plant is operating at rated power and normal operating temperature and pressure. The HPI discharge isolation check valves fail, resulting in a rupture in the suction piping to the HPI pump and discharge into the basement of the auxiliary building.

j. The method of calculation is shown in Appendix 2 of J. A. Schroeder, "Calculation Package for the INEL ISLOCA Program Steam Propagation Calculation for a B&W Plant," May 1992.

Compartment Pool Data. Water sprayed or condensed into each level of the auxiliary building will form pools on the floors. When the pool depth on the operating and intermediate floors reaches 6 to 8 in., water will overflow the curbs around floor penetrations. Water draining to the basement through these penetrations will eventually reach the RHR pit collection sump through drain piping in the basement floor. Because of the presence of interconnected drains in the basement floor, water will not build up to any significant depth (1 in. assumed) until the RHR pit fills. When water does start to accumulate on the basement floor, it will backup through the floor drains causing a level rise in all parts of the basement simultaneously. The RHR pit sump is equipped with two level-actuated pumps with a capacity of 50 gpm each. Because this capacity is small compared to the expected break discharge of thousands of gpm, the pumps (and sump volume) are neglected. Reference 27 discusses modifications to the RHR pit sump and floor drains that would prevent flooding of the compartment before water spills over the curbs on the basement floor. It is not clear whether these modifications were ever made, so this analysis assumes the RHR pit starts to flood via the RHR pit floor drain as soon as water drains from the basement level. The drainage models are discussed in more detail in the Engineered System Models section that follows.

The CONTAIN lower cell (pool) model was based on the compartment floor areas. The RHR pit sump was not modeled because it has a small area (and hence volume) compared to that of the pit. The basement floor area data from the Case-2 plant flooding report are used in the basement pool model. The maximum floor depth is about 1 in. (assumed) while water can drain to the RHR pit. When the pit fills, the water depth becomes unlimited. The intermediate and operating floor areas are bounded by the outside wall perimeter.

Compartment Condensing Surface Data. The heat sinks available for steam condensing in each compartment are the walls and ceiling (the floor being included in the lower cell, or pool model). Support walls and piping and equipment on each level were not included in the model. These structures were not included because the available references were not sufficiently detailed. Also, a previous calculation showed the metal masses to have relatively little influence on the calculated results.

Flow Path Data. The major gas flow paths in the Case-2 plant auxiliary building are the stair wells and the ladder hatch into the RHR pit. Other flow paths are present but are very difficult to quantify from the available prints. For example, there are large floor hatches that may be covered with plating, grates, or concrete slabs. The nature and strength of these barriers is not evident from the available documentation. Therefore they are excluded from the model. The six flow paths used in the CONTAIN model are described in the following paragraphs.

Path 1 is the ladder hatch connecting the RHR pit with the basement. It is not clear whether this hatch is normally closed. The analysis assumed it could be open, or will blow open at the beginning of the blowdown. This path provides a vertical connection between the RHR pit and the basement.

Path 2 is the central stairwell from the basement to the intermediate floor. This stairwell provides a vertical connection with the intermediate floor, the opening is 3.5 by 14 ft.

Path 3 is the west stairwell connecting the basement and the intermediate floor. This stairwell is the same as the central stairwell (path 2).

Path 4 is the central stairwell connecting the intermediate floor with the operating floor. This stairwell provides a vertical connection with the intermediate floor, the opening is 3.5 by 14 ft.

Path 5 is the west stairwell connecting the intermediate floor with the operating floor. This stairwell is the same as the central stairwell (path 4).

Path 6 is the truck door on the operating floor. The door is 12 ft 8 in. by 15 ft.

Engineered Systems Models. The engineered systems models are used to model water overflow through openings in each floor, water flow through drain piping, and firesprays. The floor drains on the basement level are routed to a 748-gallon (100 ft^3) sump in the RHR pit. The sump is equipped with two 50-gpm pumps that pump to a 21,444-gallon waste holdup tank. The holdup tank, in turn, overflows to a 375-gallon tank in the RHR pit. Actions that would prevent water in the sump system from backing up into the RHR pit involve sealing the sump cover and installing a check valve in the RHR pit floor drain. It is not clear whether the tank has been sealed and if a check valve has been installed in the pit floor drain. If these actions have been taken the RHR pit will not fill until water on the basement floor overflows the curbs on the ladder hatch or the equipment hatch. This analysis assumed the tank was not sealed. If this is incorrect, it will affect the time at which LPI is lost, but not the time at which HPI is lost, because the time to flood the basement above 8 in. will be the same either way. In the first instance, the basement floor will remain dry until the RHR pit fills, and then it will begin to flood. In the second instance, the basement will fill to 8 in. and hold at this level until the RHR pit is filled, and then continue to rise. The rise above 8 in. will occur at the same time in either case.

The water movement from the operating and intermediate floors into the basement and RHR pit is modeled using the overflow, pipe, and spray models described in the following sections.

Overflow Data. The operating and intermediate floors can each hold up significant volumes of water before water starts to drain into the basement and RHR pit. The depth of water on these floors is limited to the height of curbs around stair wells and other floor penetrations. The plant section drawings show the curb height to be 6 to 8 in. on both levels. Therefore water movement from these floors was simulated with the CONTAIN overflow model that allowed flow after a depth of 6 in. was reached.

Pipe Data. The basement is connected to the RHR pit sump by a large number of drains. The sump is, in turn, connected to the RHR pit floor through a floor drain. The basement to RHR pit water flow connection is modeled as an 8-in. pipe. Should the cover on the sump lift, then all the basement drains become connected to the RHR pit pool. However, modeling just the single path through the floor drain does not appear to introduce an appreciable delay in the rate at which the RHR pit pool level rises.

Spray Data. The firewater sprays on the basement and intermediate levels are included in the model. The sprays are activated by fusible links (assumed to fuse at 212°F). The basement sprays provide 244 gpm of water from heads scattered around the basement, and the intermediate level sprays provide 1226 gpm from heads on the intermediate level.[27] The spray nozzles were assumed to be located near the ceiling at each level, and to produce a droplet size of 0.001 meters.

Material Properties. The material properties required for this analysis are those of steam/water and concrete. Appropriate material properties for steam/water and concrete are built into the CONTAIN code.

5.4.3 Results. The predicted maximum auxiliary building pressurization is about 3 psig for the larger break scenarios. This is sufficient to blow open any doors that might have been closed and justifies the assumption that each level of the auxiliary building can be modeled as a single volume. The smaller breaks show pressurization of less than 1 psig. Doors might provide some protection for these sequences, however, they are not the limiting cases of concern.

The maximum room temperature of all break scenarios is about 220°F. The highest temperatures are produced by the sequences with the greatest pressurization. In general, though, the auxiliary building atmosphere temperatures are limited to 212°F.

The relative humidity predictions were similar in each scenario. All the auxiliary building floors experienced periods of 100% relative humidity.

The rate of flooding varied considerably between the sequences. However, all sequences resulted in rapid flooding of the RHR pit. The large breaks caused flooding of the HPI and charging equipment as well.

The following sections describe the plant response for each sequence.

Break Sequence 1. The auxiliary building response to this blowdown is shown in Figures 11 through 13. The pressure predictions, shown in Figure 11, indicate an increase in auxiliary building pressure of 3 psig. The temperature, shown in Figure 12, increases to a maximum of about 218°F in a few minutes, and then drops to 212°F until cooling of the RCS discharge causes temperature to drop. This occurs early in the transient.

The pool depth predictions are shown in Figure 13. The RHR pit will flood to a depth sufficient to cover the RHR pumps in 5 to 10 min. The RHR pit depth plot shows depths greater than the compartment height because the height of the room has been extended into the basement. Subtracting the difference in floor elevations between the basement and RHR pit shows that the water level in the pit follows the basement floor water depth after the pit fills. The water depth in the basement is sufficient to flood all safety injection equipment within 15 to 20 min.

Break Sequence 2. The predicted auxiliary building response during this sequence is shown in Figures 14 through 16. The pressure rise is only a few tenths of a psi. The temperatures reach 212°F very quickly and remain there until the RCS is cooled down.

The flooding that results from the break is shown in Figure 16. Note that the RHR pit fills sufficiently to submerge the RHR pumps in about 5 min. The entire pit is filled within 30 min. After 30 min the water level in the basement begins to rise. At 50 min, the safety injection and charging pumps would be immersed in a foot of water.

Break Sequence 3. The predicted auxiliary building response during this sequence is shown in Figures 17 through 19. There is essentially no pressure rise during this transient. The temperatures in the basement and intermediate levels reach 212°F in 10 min. Temperatures in the RHR pit and on the operating floor increase much more

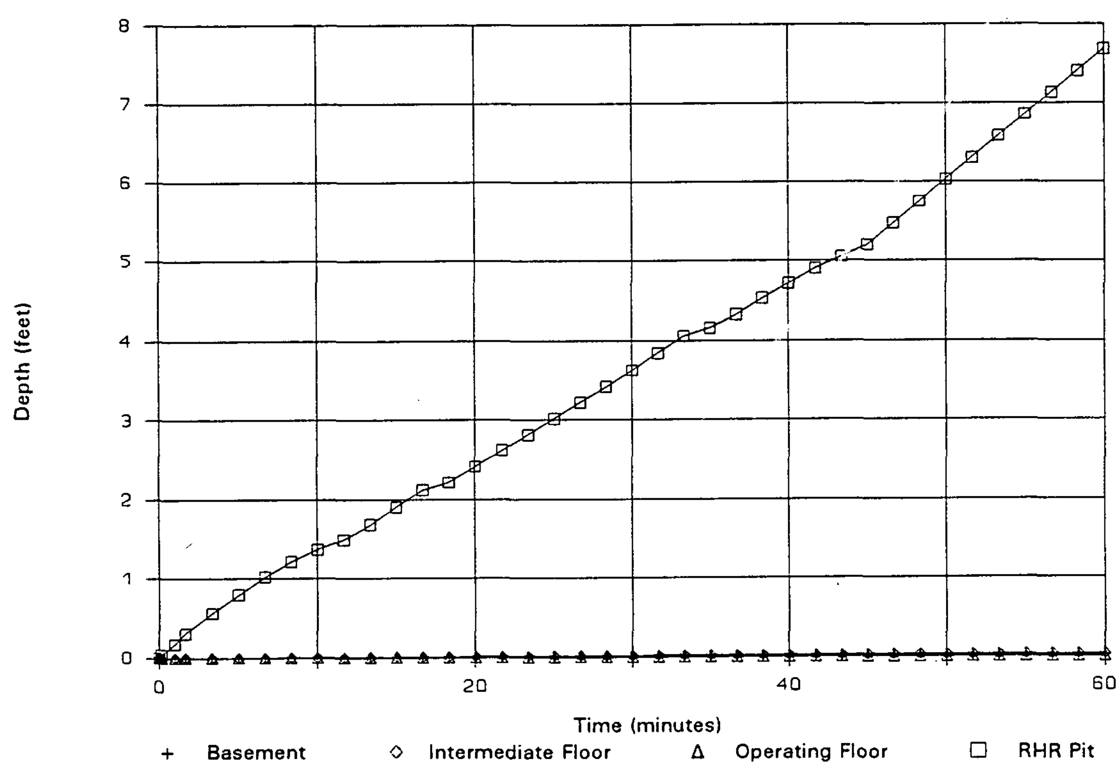

Figure 11. Auxiliary building pool depths for break sequence 3.

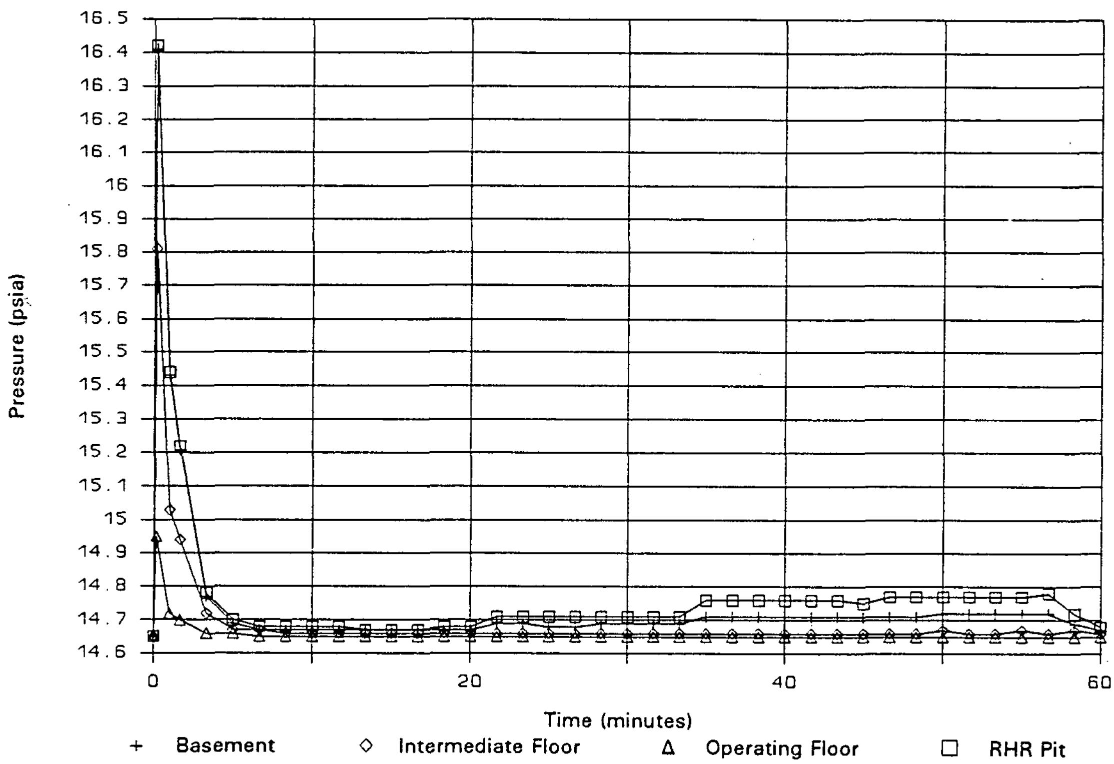

Figure 12. Auxiliary building pressures for break sequence 4.

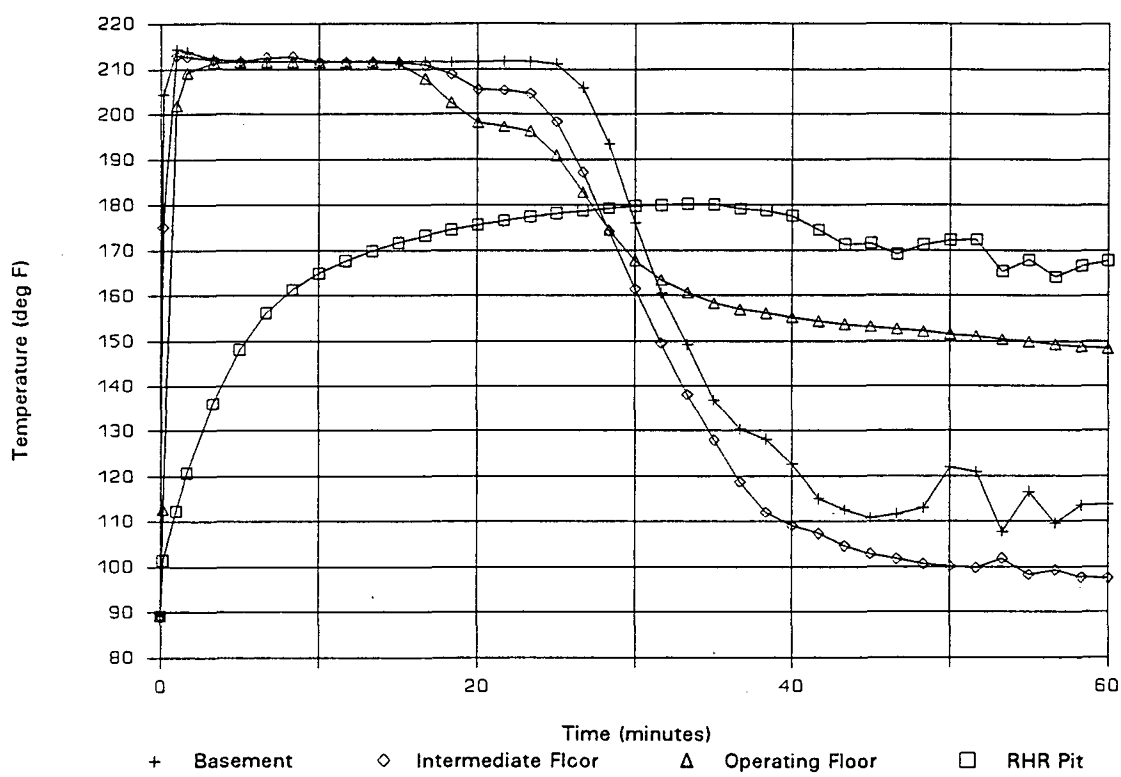

Figure 13. Auxiliary building temperatures for break sequence 4.

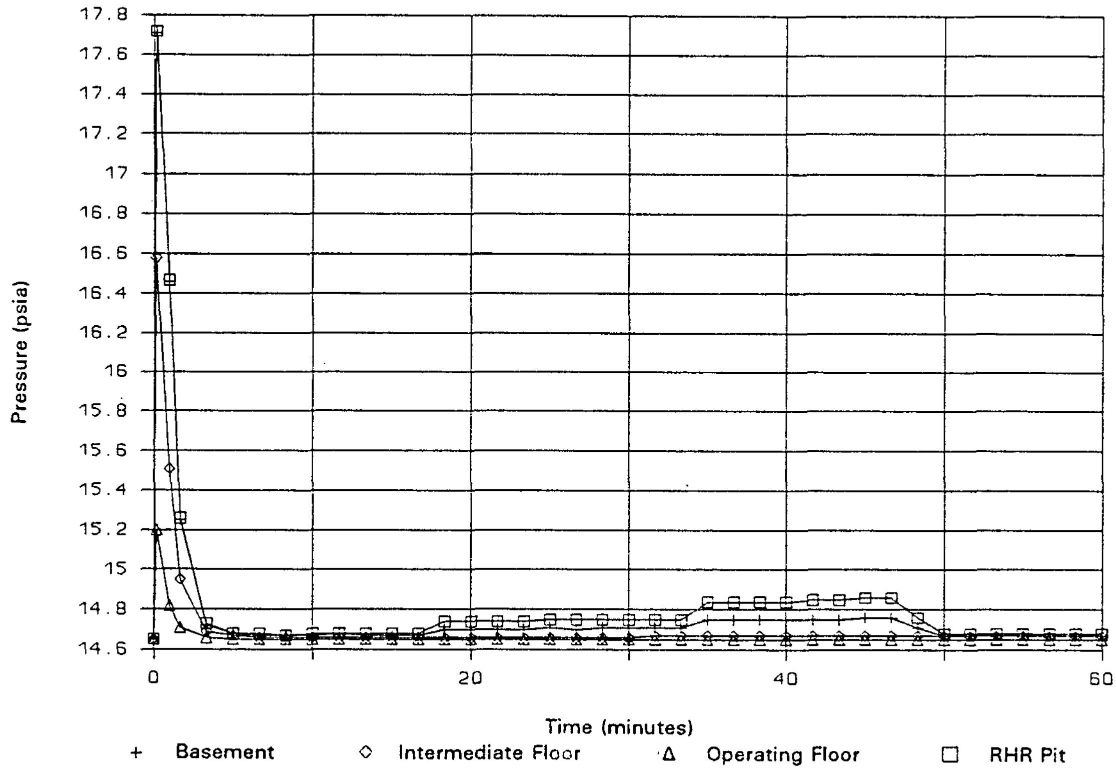

Figure 14. Auxiliary building pressures for break sequence 1.

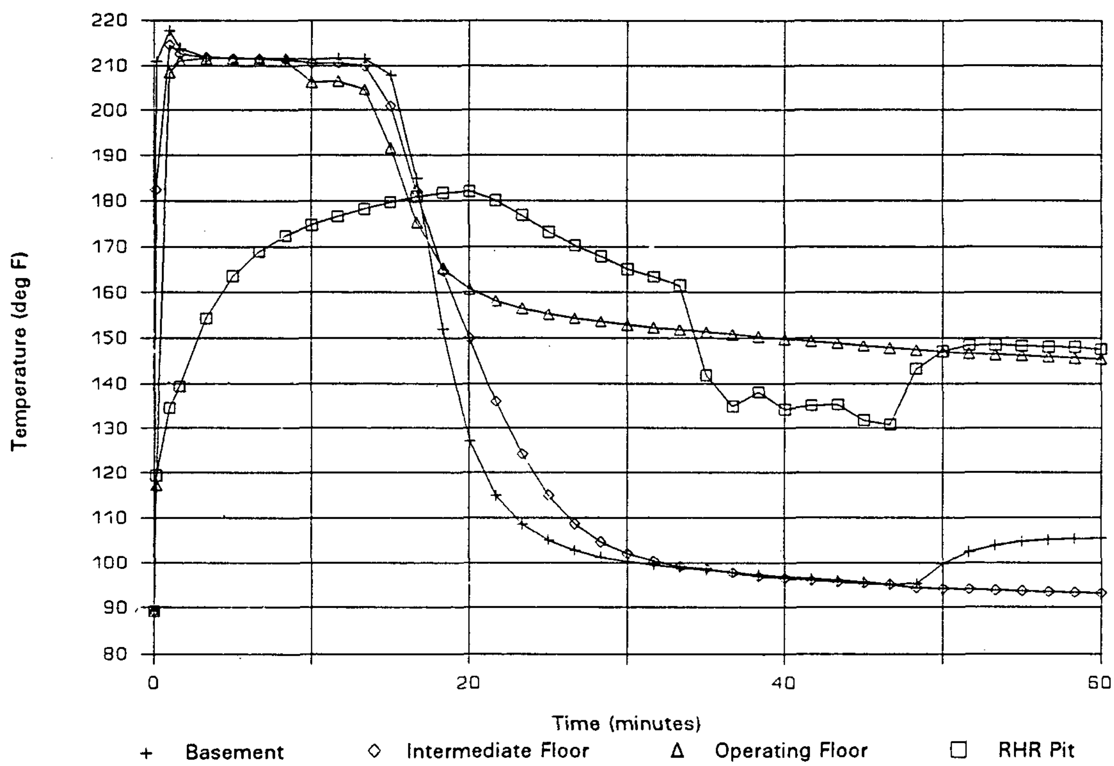

Figure 15. Auxiliary building temperatures for break sequence 1.

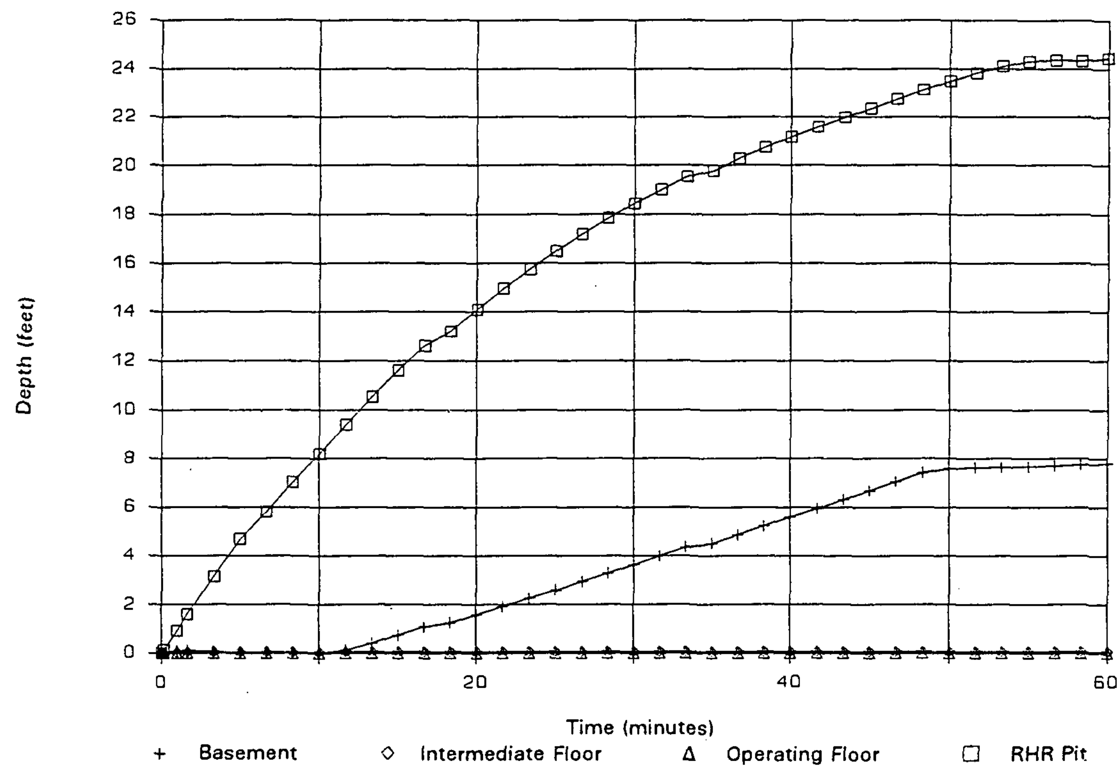

Figure 16. Auxiliary building pool depths for break sequence 1.

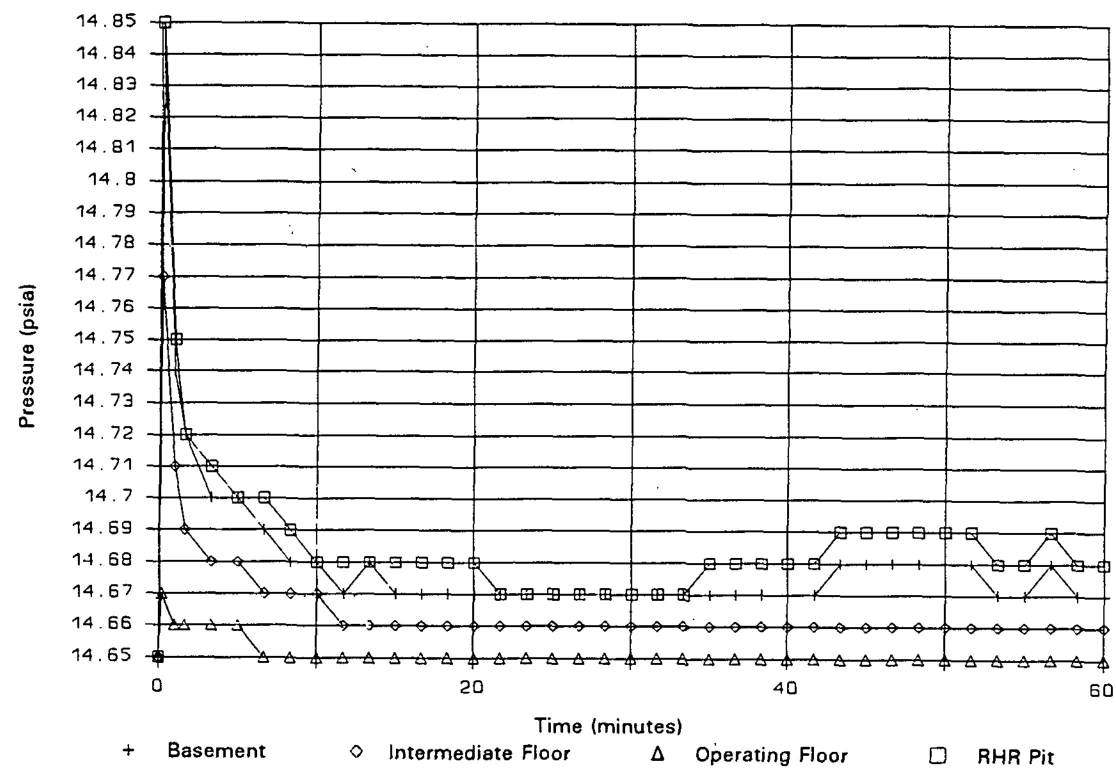

Figure 17. Auxiliary building pressures for break sequence 2.

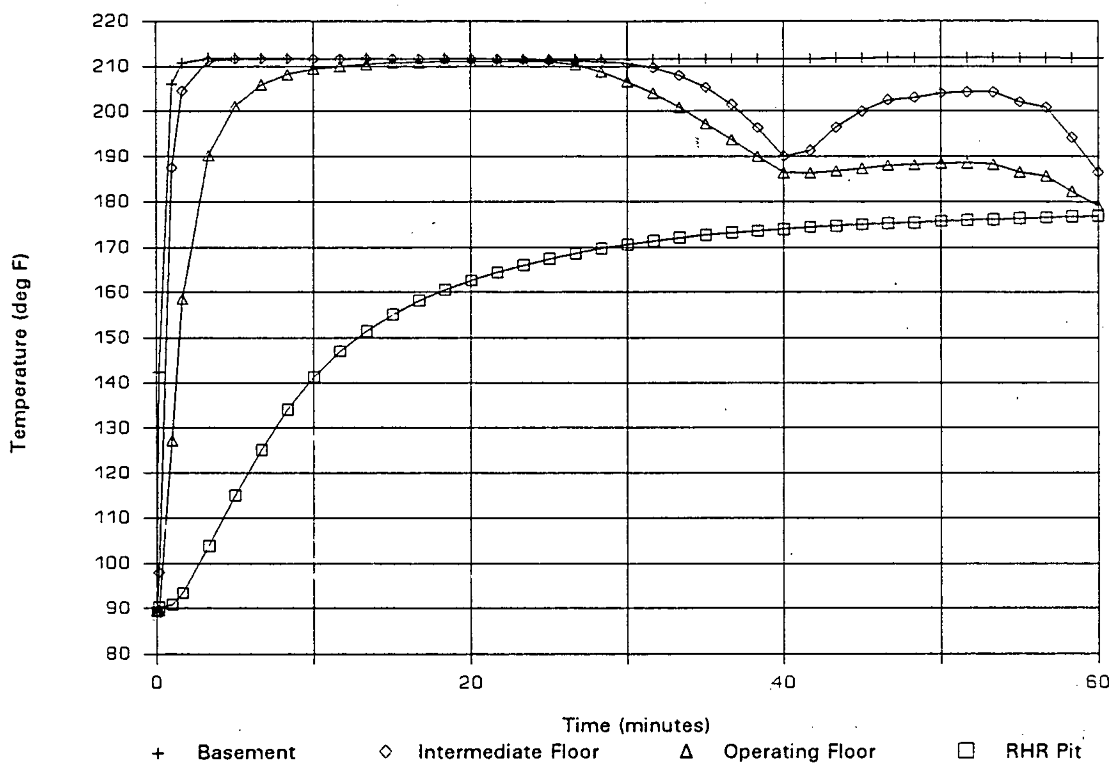

Figure 18. Auxiliary building temperatures for break sequence 2.

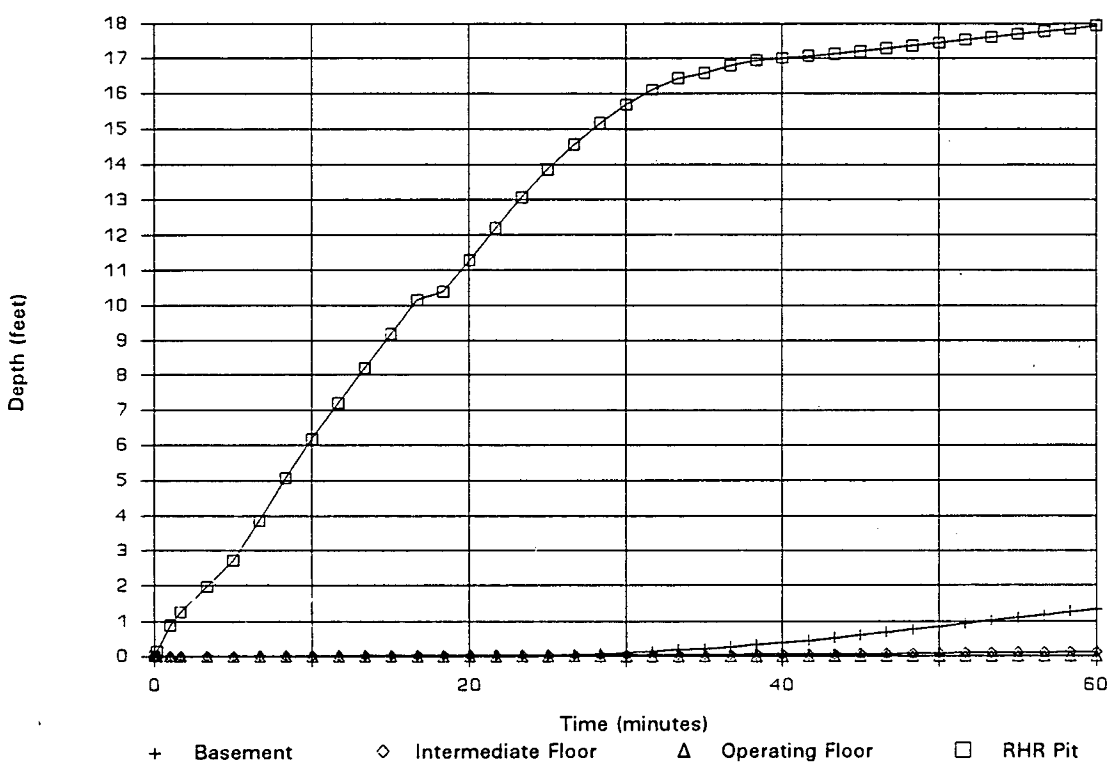

Figure 19. Auxiliary building pool depths for break sequence 2.

slowly. The RHR pit temperature rise is limited by the amount of steam that flows into the compartment. Since this is a dead ended volume, there is very little convection involved. The operating floor temperature rise is controlled by mixing of the steam with the large amount of air present. This transient is less severe than the others in that the break flow is not large enough to flush the air out of the building, leaving only a steam atmosphere behind.

The flooding that results from this transient is shown in Figure 19. The RHR pumps are still threatened in 10 to 20 min, however flooding will not result on the basement floor during the 60 min this transient was run. The flooding that results from this transient is mostly caused by discharge from the sprinkler systems.

Break Sequence 4. The predicted auxiliary building response during this sequence is shown in Figures 20 through 22. The maximum pressure rise is about 1.5 psig. The temperatures in the basement, intermediate floor, and operating floor reach 212°F within a minute or two. Temperatures in the RHR pit increase much more slowly for the reasons discussed above.

The flooding that results from this transient is shown in Figure 22. The RHR pumps will flood in several minutes. The basement floor will start to flood in about 25 min, and all safety injection and charging will likely be failed from submergence by 30 min.

Break Sequence 5. The predicted auxiliary building response during this sequence is shown in Figures 23 through 25. The maximum pressure rise is about 0.5 psig. The temperatures in the basement and intermediate floor reach 212°F at about 10 min. Temperatures in the RHR pit and on the operating floor increase more slowly for the reasons discussed above.

The flooding that results from this transient is shown in Figure 25. The RHR pumps will flood in 10 to 20 min. The basement floor will not start to flood until about 50 min, and safety injection and charging will not be threatened by flooding during the first 60 min of this transient.

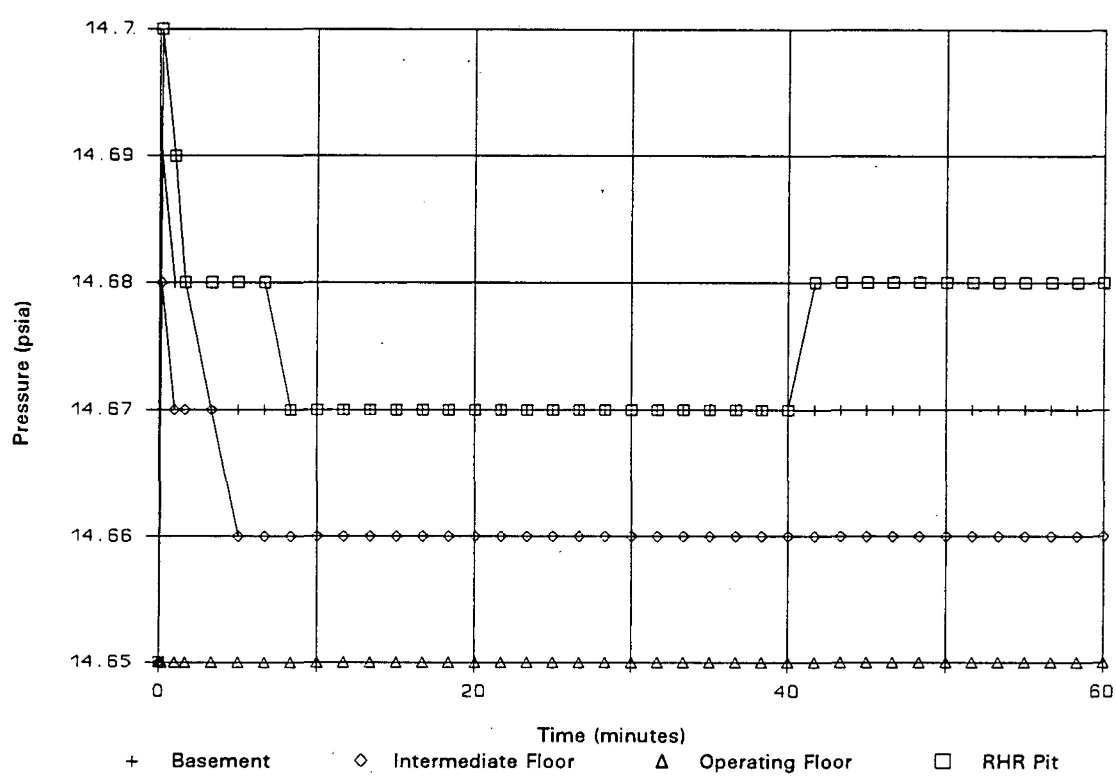

Figure 20. Auxiliary building pressures for break sequence 3.

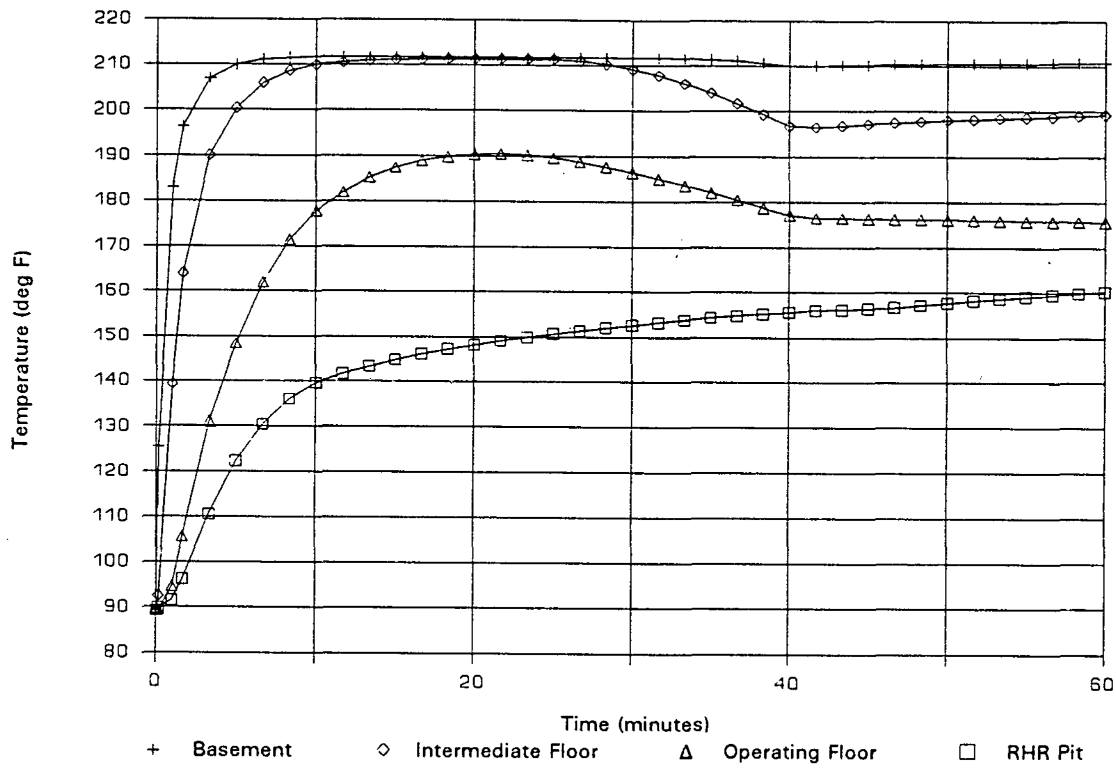

Figure 21. Auxiliary building temperatures for break sequence 3.

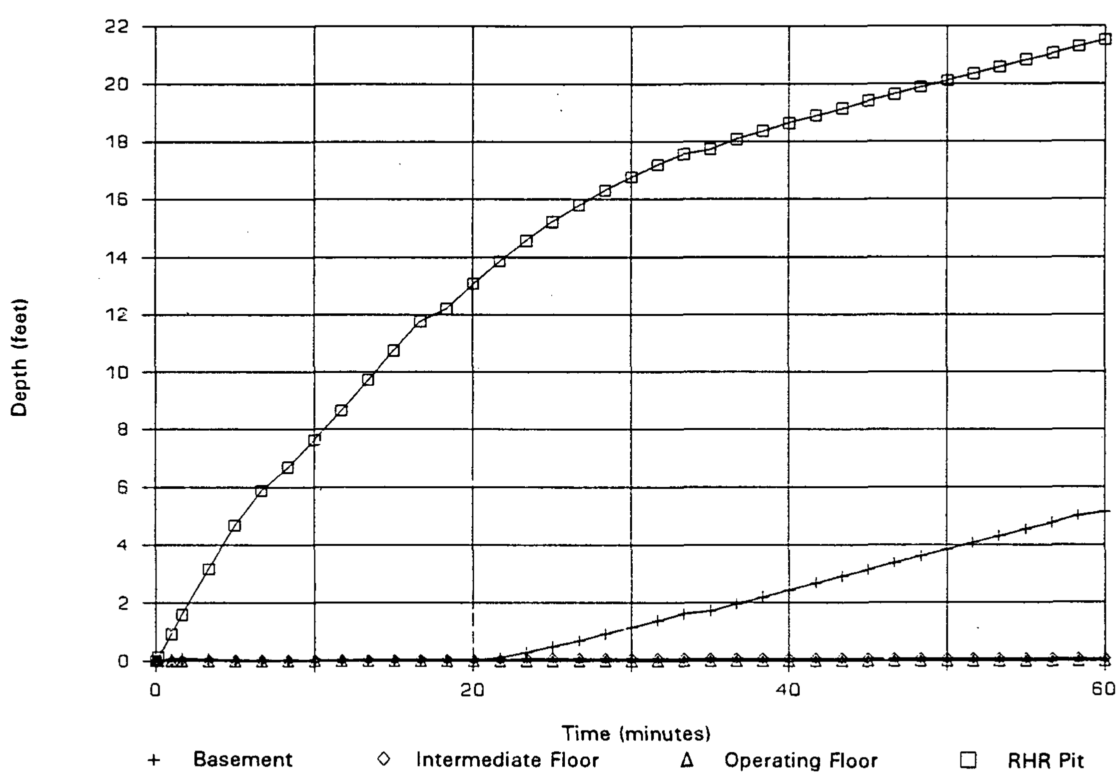

Figure 22. Auxiliary building pool depths for break sequence 4.

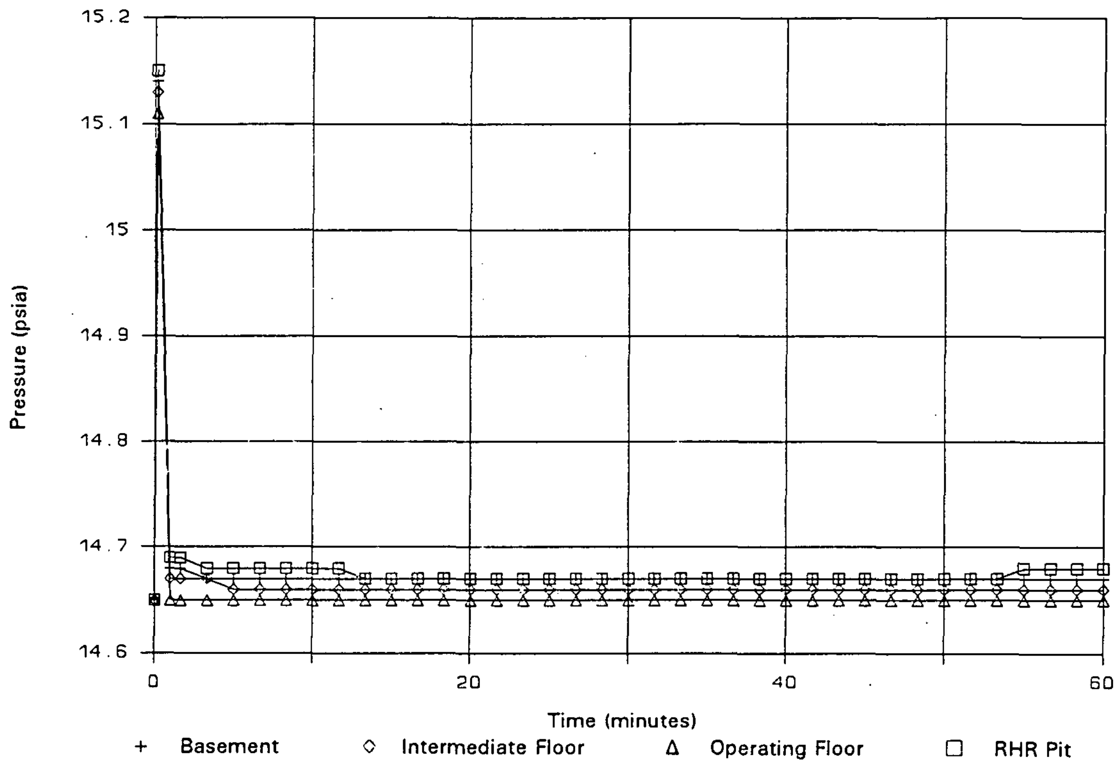

Figure 23. Auxiliary building pressures for break sequence 5.

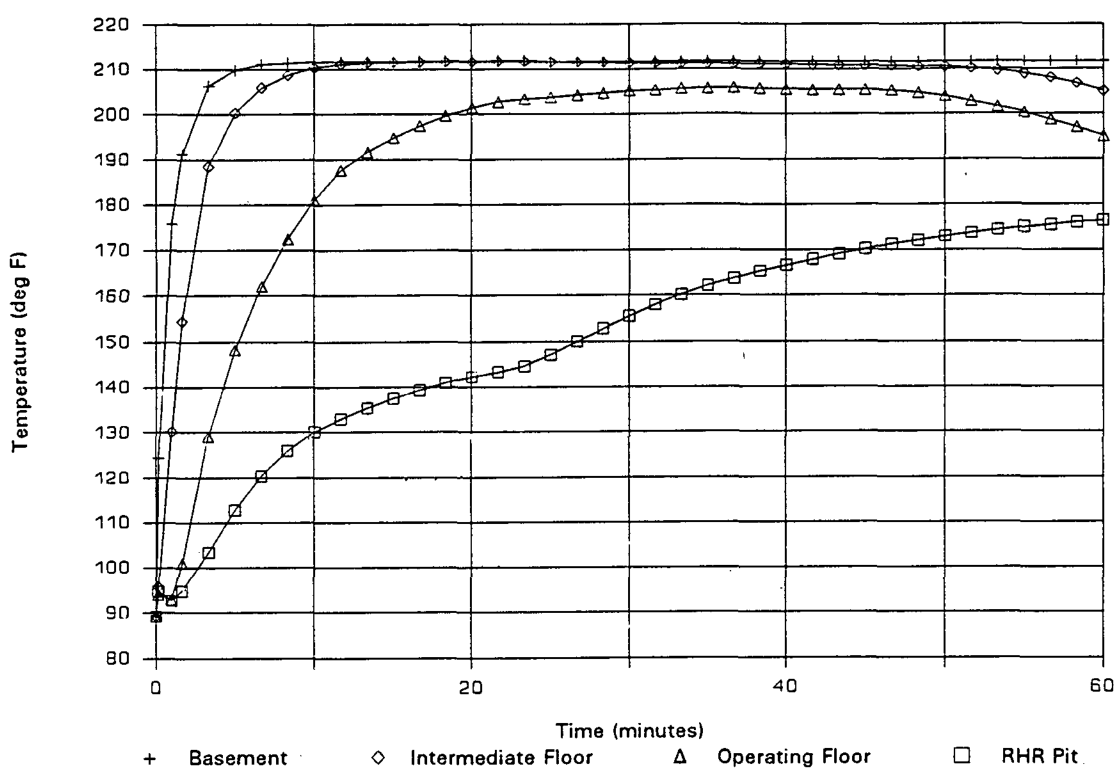

Figure 24. Auxiliary building temperatures for break sequence 5.

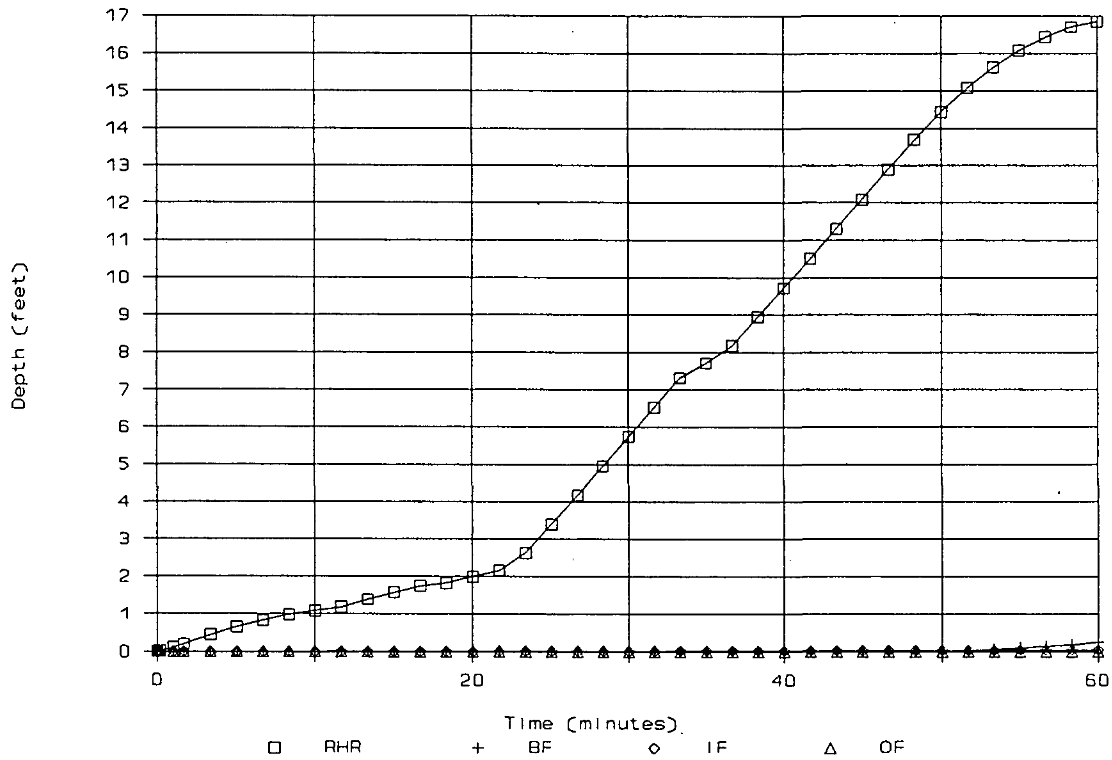

Figure 25. Auxiliary building pool depths for break sequence 5.

5.4.4 Findings. A number of generalizations can be made from the results described in the preceding section. Pressurization of the auxiliary building is limited to about 3 psig. The maximum pressurization occurs in scenario 1, and pressurization is much less in the other scenarios. This result suggests that for the worst scenarios, personnel doors would not be capable of isolating rooms from the steam environment. The smaller breaks would not have the blowdown force to open doors, and in these scenarios closed doors might isolate some spaces from the steam environment.

Temperatures in the auxiliary building do not exceed 220°F. This result is very dependent on the presence of saturated steam in the blowdown. If dry steam should be discharged, higher temperatures are possible.[27]

The relative humidity in the auxiliary building will be 1.0 for much of the transient in all the break scenarios analyzed here.

The flooding results show the RHR pumps will not likely survive any ISLOCA scenario considered here for more than a few minutes. This is because all drainage paths end up in the RHR pit eventually. The safety injection and charging pumps will survive longer, although the larger breaks will cause submergence of these in 20 to 30 min.

One final consideration should be discussed that is not mentioned above. That is the destructive force of the blowdown itself. Vapor velocities in the stair wells some distance from the break can reach hundreds of miles per hour; greater than hurricane wind forces will be generated. The destructive force of the blowdown near the break will be immense. It is likely that any power cables or motor control centers near the break will be torn loose, or destroyed. The Case-2 plant safety systems are particularly vulnerable because so much of the equipment is located near to, and on what appears to be line-of-sight, from the likely break locations.

6. INSIGHTS FROM PWR ISLOCA ANALYSIS

6.1 ISLOCA PWR Results

This section provides a general summary of the insights from the PWR ISLOCA assessments.[1,2,3] Table 11 provides a brief overview of the ISLOCA results obtained to date, and is based on the three plant specific studies conducted during the ISLOCA research program. This table indicates that the frequency of ISLOCA occurrence for the PWR plants appears to be in the range of 2×10^{-6} per reactor year. These results could be conservative in that the failure rates for normally closed check valves are based on generic data in contrast to the calculation performed in Reference 32, Table 18, which documents a Baysian update assuming no failures in 1.02E+8 check valve hours.

The application of the ISLOCA methodology to three U. S. nuclear power plants provided the following results. The reader is cautioned that extrapolating these results to other plants at this time without further analysis would be highly speculative.

6.1.1 B&W Plant Findings.

1. The total ISLOCA core damage frequency was estimated to be $\sim 2 \times 10^{-6}$ per reactor-year.

2. Potential human errors during mode changes were found to be the dominant contributors to ISLOCA core damage frequency and risk. ISLOCA initiating events driven by human error were composed of errors of commission that could occur during execution of normal procedural tasks, such as entering shutdown cooling.

3. Scenarios initiated primarily by hardware failures of pressure isolation valves were also important contributors to ISLOCA core damage frequency and risk.

4. Break isolation would be required to prevent core damage, because the makeup capacity to the borated water storage tank (BWST) is insufficient to maintain an adequate reactor coolant inventory for breaks larger than 2 in. in diameter. Although the analysis indicates that hardware would probably be available to isolate these breaks, specific procedures and training were not in place at the time of the plant visit to ensure that this hardware is used.

5. Operational experience related to ISLOCA made credible the specific human errors of commission that were the dominant contributors to ISLOCA risk.

Table 11. ISLOCA risk profile summary.

Item/Vendor	Babcock & Wilcox	Combustion Engineering	Westinghouse
Core damage frequency (/yr)	$\approx 2.0 \times 10^{-6}$	$\approx 2.0 \times 10^{-6}$	$\approx 2.0 \times 10^{-6}$
Potential for human error	High	Low	Low
Risk reduction from procedural changes	Yes	Yes	No
Break isolation required	Yes	Yes	Yes
Hardware failures	Medium	High	High

The large contribution to ISLOCA risk from human errors that could occur during mode changes highlights the importance of including a comprehensive assessment of the role of plant personnel in an ISLOCA risk evaluation. To be complete, this assessment should consider the potential for errors of commission and the effect that possible latent errors could have on the normal execution of procedures.

The CONTAIN calculations performed for this plant show that pressurization of the auxiliary building is limited to less than 1 psig, because the modeled auxiliary building compartments are very well connected to one another. The modeling parameters that affect the calculated pressure rise among compartments are flow area, discharge coefficient, and L/D. These parameters were all accurately modeled for the ECCS rooms, so any uncertainties in the flow model (and hence the pressure calculation) would stem from the number of flow junctions neglected in the balance-of-plant portion of the model. The largest pressure drop in the auxiliary building model should occur as the fluid passes from the break compartment to the immediately adjacent compartments (assuming roughly equal flow areas connecting each compartment). At each successive flow junction, the mass flow will be reduced by the mass condensed in passing through each compartment, and the effective area associated with the flow will increase as more and more parallel flow paths become available to the fluid. Therefore, pressure drops at each succeeding junction will decrease as distance from the break increases. This effect leads to the conclusion that refinements to the CONTAIN model would not have an appreciable effect on the calculated pressures.

CONTAIN predicts that, for the range of break sizes analyzed, temperatures in the auxiliary building will not exceed 212°F. However, there is one modeling uncertainty that could change this result significantly: the quality of the water-steam mixture discharged from the break into the auxiliary building. For all of the break sequences analyzed, the break discharge is a two-phase mixture with steam quality no higher than approximately 0.90. Because CONTAIN models the RCS blow down as an isenthalpic expansion from RCS pressure to compartment pressure, the resulting compartment temperatures will always be at, or very near, the saturation temperature associated with the calculated compartment pressure, as long as the break quality does not exceed approximately 0.93. At break qualities higher than 0.93, the enthalpy of the discharged fluid will be high enough (greater than 1150 Btu/lbm) to produce superheated steam in the compartment. The maximum steam temperature obtainable by this process is approximately 320°F, and occurs when dry saturated steam at approximately 500 psia is discharged from the RCS through the break. It must be emphasized that none of the simplified model predictions indicate that dry steam will be present in the break discharge. However, *recent best-estimate Oconee SBLOCA calculations do show prolonged periods during which the discharge is very nearly dry.* Given the uncertainty inherent in any calculation of this type, one cannot rule out the possibility that high quality steam will be discharged for a long enough period of time to superheat the steam in the break compartment.

The relative humidity predictions were similar in each sequence. All of the auxiliary building rooms that were evaluated experienced periods of 100% relative humidity.

The flooding results can be summarized with two general statements:

1. For the large-break sequences (RHR letdown line interface), flooding will occur in the break compartment and in adjacent compartments at a rate that will cover essential ECCS components within one hour.

2. For small-break sequences (HPI and LPI injection interfaces), flooding will occur slowly and could be delayed by operation of the compartment sump pumps. A period of many hours would pass before essential ECCS components would be threatened.

Of the various factors controlling pool formation, two dominate. The first is the rate of discharge of unflashed fluid from the break. The second is the extent to which firewater and condensate from the balance-of-plant find their way into rooms housing ECCS equipment. For the large-break sequences, the principal contributor to pool formation is the discharged liquid that does not flash to steam. For large-break sequences, as the RCS cools down, this becomes essentially the runout flow of the surviving ECCS. In small-break sequences, the discharge of fire protection sprays provides a greater flooding hazard than the accumulation of condensate or unflashed break discharge.

The results from the steam propagation analysis show that operator entry into the auxiliary building would be prohibited by the live steam environment that forms within minutes of system rupture. The results also show that the most limiting environmental factor is water pool formation in the ECCS compartments. When the pools reach a depth of 2 ft, the ECCS pump motors will be submerged, failing the ECCS pumps. The time at which this failure occurs becomes the limiting time available for operator recovery of the plant. The temperature and humidity effects were not important at the B&W study plant because the ECCS equipment was qualified for the postulated environment produced by a high energy line break, a more severe environment than predicted here.

6.1.2 Westinghouse Plant Results.

1. The total ISLOCA core damage frequency was estimated to be ~2 × 10^{-6} per reactor-year.

2. Human errors that could occur during startup and shutdown of the plant were negligible contributors to ISLOCA core damage frequency and risk. This is in contrast to the finding for the B&W plant and is a direct result of the high quality of the administrative controls and safety culture found at the Westinghouse plant.

3. Sequences initiated by hardware failures of pressure isolation check valves were the only significant contributors to ISLOCA core damage frequency and risk.

4. As in the B&W plant, break isolation would be an important recovery action, because the makeup capacity from the refueling water storage tank (RWST) is insufficient to maintain an adequate reactor coolant inventory for breaks that are larger than approximately 2 in. in diameter. The analysis indicates that hardware would be available to isolate these ISLOCA breaks, and, in contrast with the finding for the B&W plant, adequate procedures and training are generally available to ensure that this hardware is used.

5. At the time of the plant visit, a general survey was made of the interfacing system flow paths to *qualitatively* estimate the impact on equipment of ruptures in various locations. This survey could not verify that the ECCS components are adequately separated such that any postulated rupture would not affect redundant ECCS trains. In particular, in the case of the residual heat removal (RHR) system, the pumps for units 1 and 2 (the analyzed plant is located at a two-unit site) are located in the basement of the auxiliary building and a common corridor runs outside of the individual pump rooms. If there were a pipe break and blowdown of steam and liquid from the RCS into one of the RHR pump rooms, this configuration may not ensure that at least one train of ECCS would still be available following the rupture. In other words, because of the common corridor, a rupture in the RHR pump room of one unit could conceivably disable the RHR pumps for the other unit also.

6. A significant reduction in ISLOCA risk through relatively simple changes to procedures, training, and instrumentation does not appear achievable.

6.1.3 Combustion Engineering Plant Results.

1. The total ISLOCA core damage frequency was estimated to be ~2 × 10^{-6} per reactor-year.

2. As at the Westinghouse plant, human errors that could occur during startup and

shutdown of the plant were negligible contributors to ISLOCA core damage frequency and risk. Again, this is a direct result of the high quality of the administrative controls and safety culture found at the plant.

3. Sequences initiated by hardware failures of pressure isolation check valves were the dominant contributors to ISLOCA core damage frequency and risk. However, at the CE plant, exposure of the low-pressure interfacing system is precipitated by stroke-testing of a normally closed injection flow control valve.

4. As in the other two plants, break isolation would be an important recovery action, because makeup capacity to the refueling water storage pool (RWSP) is insufficient to maintain an adequate reactor coolant inventory for breaks that are larger than approximately 2 in. in diameter. The analysis indicates that hardware would be available to isolate these ISLOCA breaks; however, as for the B&W plant, procedures were not available at the time of the plant visit to ensure that this hardware is used in all sequences.

5. At the time of the plant visit, a general survey was made of the interfacing system flow paths to qualitatively estimate the impact on equipment of ruptures in various locations. As for the Westinghouse plant, this survey could not verify that the emergency core cooling systems (ECCS) are adequately separated such that any postulated rupture would not affect redundant ECCS trains.

6. It appears that relatively simple changes to procedures and training could reduce ISLOCA risk by reducing the initiator frequency and by increasing the likelihood of successfully isolating an intersystem break.

6.2 Plant Operational Data

Pressure isolation valve testing varies between PWR units. The PWR PIV trains usually consist of two check valves inside containment and one motor-operated check valve outside of the containment. The motor-operated valves are usually left open to enhance the reliability of the ECI safety systems. A series of motor-operated valves may be required in some facilities in which an injection path serves several functions. In these cases the motor-operated valves are normally closed and only opened when the specific function is required. These normally closed valves are commonly subject to stroke testing. Stroke testing of valves generally takes places either during a refueling outage or quarterly. Leak testing of PWR PIVs tends to take place during refueling outages. The practices and procedures used during leak testing, however, vary widely between units.

Several incidents have occurred at domestic and foreign PWRs involving overpressurization of the interfacing systems. Descriptions of these events can be found in previous U.S. NRC sponsored ISLOCA reports[1,6,7] and in an EPRI sponsored report.[28]

The hardware failures involved failure of the valve to properly seat or failure of the valve's position indicators. The following items must be considered to properly analyze valve failure in an ISLOCA assessment.

- Internal leakage
- Internal rupture
- Failure to hold
- Mispositioned
- Spurious failure
- Failure to reseat.

The review of the PWR ISLOCA precursor incidents indicates that the following items or insights were contributors to the incidents:

- Motor and check valves mispositioned or leaking
- Testing and maintenance actions

- Lack of EOPs to cover ISLOCA events for all operational modes
- Conflict between protection of low pressure systems and ECCS valve operational testing
- Diagnosis failure.

The review of the ISLOCA precursor events indicates that the causes can be traced to three general areas. These three areas are: (a) human errors—typically the result of miscommunication, (b) hardware failures—such as a leaking or partially stuck open valve, and (c) testing and maintenance (T&M) operations—either in conjunction with a hardware or human error, or during a system alignment or plant evolution that inadvertently conflicts with the T&M activity.

6.3 Emergency Protection Guidelines

One difference between the NSSS vendors is the support given to the utility by the owners' groups in developing procedures to diagnose and isolate LOCA events outside of the containment. The B&W Owners Group supplies no information to the utility in the abnormal transient operating guidelines (ATOGs) for isolating and diagnosing breaks outside of the containment. The development of EOPs to treat B&W ISLOCA events is the responsibility of the specific utility, and in the case of the plant examined, takes the form of alarm response procedures. The Combustion Engineering Owners Group provides EPGs to isolate an interfacing system LOCA. The EPGs do not provide detailed guidance in diagnosing the location of the break outside of the containment. The Westinghouse Owners Group provides the most detail in its emergency response guidelines (ERGs) to diagnose and isolate the ISLOCA events.

6.3.1 Babcock & Wilcox Owners Group. The B&W Owners Group have developed ATOGs for procedural guidance in emergency situations.[29] A review of the ATOGs indicates that there is no *specific* guidance regarding ISLOCA sequences or operator recovery responses for the interfacing system LOCAs.

The B&W plants have two levels of emergency procedures. The first level procedures are plant specific ATOGs that contain information required to mitigate abnormal transients and restore safety functions. The second level procedures are emergency operating procedures or EOPs. The EOPs are designed to mitigate the events that caused the plant operational technical specification parameters to be violated.

The individual B&W plants have developed EOPs that address ISLOCA events to various degrees. Small RCS leaks outside containment are directly addressed in the EOPs. The procedures identify interfacing systems as potential leak paths. However, large leaks are only addressed through the alarm response procedures (e.g., loss of RHR pump).

6.3.2 Combustion Engineering Owners Group. The operator response to the ISLOCA is driven by symptom-based LOCA procedures. The Combustion Engineering emergency procedures guidelines instruct the operators to monitor pressurizer level, containment pressure, and steam plant activity to diagnosis the ISLOCA events.[30] The Combustion Engineering emergency procedure guidelines do not provide a *precise* diagnosis of the ISLOCA event.

The Combustion Engineering emergency protection guidelines are structured and contain four items. These four items are:

- Posttrip actions or SPTAs
- Diagnostic actions
- Optimal recovery guidelines or ORGs
- Functional recovery guidelines.

The ISLOCA events are addressed in the optimal recovery guidelines (ORG) for the loss-of-coolant accidents. The LOCA ORGs provide actions to support long term core cooling. These actions include break isolation and use of the safety injection systems. The procedures, however, do not make use of the conditions present in the auxiliary building to diagnose the ISLOCA event. The Combustion Engineering emergency procedure guidelines do not provide a *precise* diagnosis of the ISLOCA event because of a lack of attention to the auxiliary building environmental conditions.

6.3.3 Westinghouse Owners Group. The Westinghouse emergency response guidelines (ERGs) explicitly address the ISLOCA accident.[31] The procedures are found in the Emergency Contingency Action Guidelines (ECAs). The ECA which address the ISLOCA events is ECA-1.2, titled "LOCA outside of the containment." The procedures instruct the operators to (a) verify valve alignment, (b) identify and isolate the break, and (c) verify the break is isolated.

The ISLOCA procedures first instruct the operations crew to verify valve alignment of the interfacing systems. This step in the ISLOCA procedure can identify any PIV left open because of testing or maintenance errors. The operator is also instructed to identify and isolate the break. The isolation is accomplished by closing in sequence all the valves that isolate the low-pressure/high-pressure lines from the containment.

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Appendix A

Bounding Calculations and Data

Appendix A

Bounding Calculations and Data

This appendix contains information used in BWR interface bounding calculations. This analysis, which is documented in Section 4 of this report, comprises an initial screening of the interfaces. Those that survive are subject to a bounding estimate using the failure rates presented here.

Background Information—Obtained from plant procedures.

- Isolation valve stroke tests are performed only during shutdown.
- Valve positions are checked monthly. Therefore, spurious operation type faults are assumed to produce a conditional unavailability of one month duration.
- Isolation valve leak tests are performed during shutdown. However, the procedures specify that only "normal means" must be used to close the valves (that is, no manual closing) to satisfy test requirements.

Motor-Operated Valves (MOVs)

Normally Closed, Fails Open (NCFO)

- Spurious operation (SO): 5E–8/hr Ref. 1
- Internal rupture (IR): 1E–7/hr Ref. 1
- Human error—unspecified (HE)
- Omission—left open (Om): 3E–3/yr estimated
- Commission—opened inappropriately (Co): 1E–3/yr estimated

*MOV-NCFO = (SO * 1-month) + (IR * 1-yr)*

+ HE-Om + HE-Co

*MOV-NCFO = (5E–8/hr * 730 hr) + (1E–7/hr*

** 8760 hr) + 3E–3 + 1E–3*

MOV-NCFO = 5E–3.

Check Valves (CV)

Normally Closed, Fails Open (NCFO)

- Internal rupture: 5E–7/hr Ref. 1

Normally Open, Fails Open (NOFO)

- Fail to close: 1E–3/demand Ref. 1

Testable (pneumatic) Check Valves (TCV)

Normally Closed, Fails Open (NCFO)

- Spurious operation: 5E–7/hr Ref. 1
- Internal rupture: 5E–7/hr Ref. 1
- Human error (unspecified)
- Omission (left open): 3E–3/yr estimated
- Commission (opened inappropriately) : 1E–3/yr estimated

*TCV-NCFO = (SO * 1-month) + (IR * 1-yr)*

+ HE-Om + HE-Co

*TCV-NCFO = (5E–7/hr * 730 hr) + (5E–7/hr*

** 8760 hr) + 3E–3 + 1E–3*

TCV-NCFO = 9E–3.

Normally Open, Fails Open (NOFO)

- Fail to close: 1E–3/demand Ref. 1

Appendix B

RHR Heat Exchanger Pressure Fragilities

Appendix B

RHR Heat Exchanger Pressure Fragilities

Table B-1. Residual heat removal heat exchanger failure pressures.[a]

Temperature (°F)	Shell side cylinder ρ (psig)	Shell side cylinder β	Shell side head buckling[b] ρ_o	Shell side head buckling[b] β
Room temperature	2940	0.16	3540	0.22
400	3120	0.22	3060	0.25
600	2985	0.24	2830	0.27
800	2440	0.20	2600	0.30

a. All leak areas are large, uncontrolled leaks.

b. Assume 0.2 probability of crack formation given head buckling occurs.

Table B-2. Residual heat removal heat exchanger tube sheet.

	Effective gasket stress (psi)	Act gasket stress (psi)	Gasket deflectivity (in.)	Gross leak pressure (psi)	Leak rate at GLP (mg/s)	Leak rate at 0.25 GLP (mg/s)	Leak rate at 0.50 GLP (mg/s)	Leak rate at 0.75 GLP (mg/s)	Leak area at 1.25 GLP (sq in.)	Leak area at 1.5 GLP (sq in.)	Leak area at 1.75 GLP (sq in.)	Leak area at 2.0 GLP (sq in.)	Bolt stress at 2.0 GLP (psi)
(1)	32037	32037	0.087	1252	0	0	0	0	1.03	2.06	3.09	4.12	79784
(2)	35241	35241	0.096	1377	0	0	0	0	1.13	2.26	3.40	4.53	87763
(3)	38445	38445	0.104	1503	0	0	0	0	1.24	2.47	3.71	4.94	95741
(4)	29955	29955	0.096	1170	0	0	0	0	0.96	1.92	2.89	3.85	79835
(5)	26431	26431	0.096	1031	0	0	0	0	0.85	1.70	2.55	3.40	74550

Gasket: O.D. = 57.125 in. I.D. = 55.875 in. Thickness = 3/16 in. K = 369,000 psi/in.

Bolts: (64) 1-1/4 in. diameter SA 193-B7

(1) Initial bolt stress = 50,000 psi. Joint relaxation = 0%.

(2) Initial bolt stress = 55,000 psi. Joint relaxation = 0%.

(3) Initial bolt stress = 60,000 psi. Joint relaxation = 0%.

(4) Initial bolt stress = 55,000 psi. Joint relaxation = 15%.

(5) Initial bolt stress = 55,000 psi. Joint relaxation = 25%.

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11. ABSTRACT *(200 words or less)*
This report contains a compilation of information generated during the ISLOCA research program. Presented is a screening analysis and a procedures guide for performing an ISLOCA evaluation. This methodology has been distilled from past analyses performed for the U.S. Nuclear Regulatory Commission and documented in a series of NUREG/CR reports. The methodology comprises five distinct steps: (a) containment penetration screening, (b) interfaces for ISLOCA analysis, (c) mechanisms for failing the pressure boundary, (d) construction of event trees and estimation of rupture probabilities, and (e) quantification of the event tree. Included in the methodology are steps required for a detailed human reliability analysis. In addition, this report presents a BWR ISLOCA evaluation, a survey of PWR auxiliary building designs and identification of one design deemed most disadvantageous with respect to ISLOCA risk, and a PWR ISLOCA cost/benefit analysis.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*
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