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Assessment of ISLOCA Risk-Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

Inter-system loss-of-coolant accidents (ISLOCAs) have been identified as important contributors to offsite risk for some nuclear power plants. A methodology has been developed for identifying and evaluating plant-specific hardware designs, human factors issues, and accident consequence factors relevant to the estimation of ISLOCA core damage frequency and risk. This report presents a detailed description of the application of this analysis methodology to a Westinghouse four-loop ice condenser plant.

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EXECUTIVE SUMMARY

Inter-system loss-of-coolant accidents (ISLOCAs) have been identified in some probabilistic risk assessments (PRAs) as major contributors to offsite risk at nuclear power plants (NPPs). They have the potential to result in core damage and containment bypass (thus, they are beyond the design basis of the plant), which may lead to the early release of large quantities of fission products to the offsite environment. Recent events at several operating reactors have been identified as ISLOCA precursors. These events have raised concerns over the frequency of occurrence, plausible initiators, and means of identifying and mitigating this potential accident. In response to these concerns, a June 7, 1989, memorandum, "Request for Office of Nuclear Regulatory Research (RES) Support for Resolution of the ISLOCA Issue," was transmitted from Dr. Thomas E. Murley to Dr. Eric S. Beckjord. The ISLOCA research program

described in this report was initiated in response to this memorandum.

The objective of the ISLOCA research program is to provide the U.S. Nuclear Regulatory Commission with qualitative and quantitative information on the hardware, human factors, and accident consequence issues that contribute to ISLOCA risk. To meet this objective, a methodology has been developed to estimate the core damage frequency (CDF) and offsite consequences associated with an ISLOCA, and this methodology is being applied to individual NPPs. This report describes the ISLOCA methodology and the results of its application to a Westinghouse four-loop ice condenser NPP.

An eight-step methodology was developed to evaluate the ISLOCA issue qualitatively and quantitatively. These steps and their relationships to one another are shown in Figure ES-1. This

Figure ES-1. Approach for plant-specific evaluation of ISLOCA.

methodology was applied to a Westinghouse plant by a team of PRA and human factors specialists. The important results, *specific to this plant*, are

1. Human errors that could occur during startup and shutdown of the plant were not found to be significant contributors to ISLOCA CDF and risk.
2. ISLOCA sequences initiated by hardware failures of check valves were the dominant contributors to CDF and risk.
3. Isolation of the break would be an important recovery action during an ISLOCA. Refueling water storage tank makeup capacity is insufficient to maintain an adequate reactor coolant inventory for breaks outside the containment that are larger than approximately one inch in diameter. The analysis indicates that hardware would be available to isolate these ISLOCA breaks, and adequate procedures and training are generally available to ensure that this hardware is used.
4. 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 emergency core cooling systems (ECCSs) 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 piping break and blowdown of steam from the reactor coolant system (RCS) into one of the RHR pump rooms, this configuration may not ensure that at least one train of ECCS would still be available after a rupture had occurred. 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. This issue is still under study at the Idaho National Engineering Laboratory.
5. A significant reduction in ISLOCA risk through relatively simple changes to procedures, training, and instrumentation does not appear achievable.
6. The ISLOCA methodology has been successful in providing important insights on the relative contribution of both hardware faults and human actions to ISLOCA CDF and risk. In particular, the extensive task analysis performed as part of the human reliability analysis provided many valuable insights that would have been missed in a less detailed analysis.

While caution must be exercised when using these results to draw general conclusions about the ISLOCA risk at other NPPs, the perspective provided by the aforementioned important insights, along with suitable sensitivity studies and companion PRAs, provide additional technical bases upon which a regulatory decision for resolution of ISLOCA as a generic issue can be considered.

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All of the personnel at the analyzed plant were most helpful, handling all of the logistics of the data-gathering visits in an admirable fashion. The difficulty of their task can only be appreciated by someone who has had to interface with a large U.S. Nuclear Regulatory Commission inspection team like the one put together for this project. They coordinated all of our many requests for information and equipment data, both during the inspection and after. They managed to do this while still performing their regular duties. Without their assistance, this project would not have been possible.

At the Idaho National Engineering Laboratory, thanks are due to John Schroeder, who performed the RELAP5 system calculations and assisted in the rupture probability calculations, to Craig Kullberg, who provided the core uncover time estimates, to Dan Pafford, who aided in calculating source terms and offsite consequences, and to Debbie Johnson, who provided the simplified system flow diagrams.

Finally, we would like to thank all the members of the inter-system loss-of-coolant accident inspection team for their aid in gathering and interpreting plant data.

ACRONYMS

AFW	Auxiliary feedwater	ND	Plant designation for low-pressure safety injection/residual heat removal system
BNL	Brookhaven National Laboratory		
CDF	Core damage frequency	NI	Plant designation for high-pressure safety injection system
CRS	Control room supervisor		
DF	Decontamination factor	NPP	Nuclear power plant
DHR	Decay heat removal	NRC	U.S. Nuclear Regulatory Commission
ECCS	Emergency core cooling systems	NSSS	Nuclear steam supply system
EPRI	Electric Power Research Institute	P&ID	Piping and instrumentation diagram
GLP	Gross leak pressure	PIV	Pressure isolation valve
HCR	Human cognitive reliability	PORV	Pilot-operated relief valve
HEP	Human error probability	PRA	Probabilistic risk assessment
HRA	Human reliability analysis	PSF	Performance shaping factor
INEL	Idaho National Engineering Laboratory	PWR	Pressurized water reactor
ISLOCA	Inter-system loss-of-coolant accident	RCS	Reactor coolant system
IST	In-service testing	RHR	Residual heat removal
LER	Licensee Event Report	RHT	Recycle holdup tank
LOCA	Loss-of-coolant accident	RO	Reactor operator
LTOP	Low-temperature overpressure protection	RPV	Reactor pressure vessel
MAAP	Modular Accident Analysis Program	RWST	Refueling water storage tank
MACCS	MELCOR Accident Consequence Code System	SHARP	Systematic Human Action Reliability Procedure
MOV	Motor-operated valve	TALENT	Task Analysis-Linked Evaluation Technique
NC	Plant designation for reactor coolant system	THERP	Technique for human error rate prediction

Assessment of ISLOCA Risks—Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant

1. INTRODUCTION

The *Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400, identified a class of accidents that can result in overpressurization and rupture of systems that interface with the reactor coolant system (RCS).¹ These events were postulated to be caused by failure of the check valves and motor-operated valves (MOVs) normally used for system isolation. In a subset of these inter-system loss-of-coolant accidents (ISLOCAs), called V-sequences or event V, the system rupture occurred outside of the containment building. In cases where the rupture led to severe core damage, ISLOCAs were found to be significant contributors to risk because fission products released from the RCS bypassed the containment and were discharged directly to the environment. Subsequent probabilistic risk assessments (PRAs), including the NUREG-1150 results for Surry and Sequoyah nuclear power plants (NPPs), have identified ISLOCAs as important contributors to public health risk.² Researchers at Brookhaven National Laboratory (BNL) have evaluated the vulnerability of several reactor designs to an ISLOCA and identified improvements that could reduce ISLOCA frequency.^{3,4}

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

The objective of the ISLOCA research program is to provide the U.S. Nuclear Regulatory Commission (NRC) with qualitative and quantitative information on the hardware, human factors, and accident consequence issues that contribute to ISLOCA risk. This information is to be used in

- Developing a PRA framework for evaluating the ISLOCA issue and identifying insights with respect to the risk contribution from both hardware and human factors, along with recommendations for reducing the ISLOCA risk.
- Highlighting the effects of specific types of human errors and their root causes on ISLOCA risk, along with recommendations for risk reduction.
- Evaluating the fragility of low-pressure systems exposed to high-pressure, high-temperature reactor coolant. This evaluation will include identification of likely failure locations and failure probabilities.
- Identifying and describing potential ISLOCA sequences with respect to timing, possible accident management strategies, and effects on other plant equipment and systems.
- Estimating the fission product source terms and offsite consequences for postulated ISLOCAs. Again, important issues will be identified and recommendations will be made on possible consequence reduction actions.

To meet the above program objectives, a methodology has been developed to estimate the

Introduction

ISLOCA core damage frequency (CDF) and offsite risk, and this methodology is being applied to a limited sample of NPPs of different design. This report describes the ISLOCA methodology and documents the results from its application to a Westinghouse four-loop ice condenser plant. These results emphasize the effect of hardware failures and human actions on the ISLOCA CDF. The offsite risk measures are useful in comparing results from the sensitivity studies. Major uncertainties in this estimate also are identified.

Section 2 of this report describes the methodology developed to evaluate ISLOCAs, the approach taken for its application to a specific plant, and a description of the plant systems that were identified as potential ISLOCA flow paths. Section 3 contains a more detailed description of the interfacing systems and the possible ISLOCA scenarios. Section 4 describes the plant-specific results, and Section 5 contains the conclusions and recommendations from this assessment. Appendices A–I are used to document the details of the separate analyses.

2. APPROACH

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

Before beginning the individual plant evaluations, historical plant operating information was reviewed to gather insights on ISLOCA issues. The major emphasis of this evaluation was to identify and evaluate Licensee Event Reports (LERs) that (a) involved valve failures resulting from either hardware failures or human errors or

(b) indicated that an ISLOCA had occurred. The results from this search provided information on the causes and frequencies of valve failures and provided important insights on the systems involved and the potential causes of those ISLOCAs that have occurred. This information was used during the plant visits to help identify systems to be reviewed, develop the event trees, and quantify the failure rates of some interfacing system valves. Appendix A in *Assessment of ISLOCA Risks—Methodology and Application to a Babcock and Wilcox Plant* summarizes the results of this evaluation.⁵

2.1 Assessment of ISLOCA Potential

The first step in the individual plant evaluation approach is a preliminary assessment of the

Figure 1. Approach for plant-specific evaluation of ISLOCA.

potential for an ISLOCA to occur. Plant-specific information on the systems that could be involved in an ISLOCA is obtained during a short data-gathering visit to the plant. Detailed information is gathered on the hardware and operation of a range of low- and high-pressure interfacing systems. Examples of information collected include plant procedures, piping and instrumentation diagrams (P&IDs), isometric drawings, and training manuals. This information is then reviewed by a team of PRA and human factors 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 RCS are identified during this preliminary assessment. A determination is then made of the maximum interfacing system break size that would not be expected to result in core damage. The systems are screened to identify those with interfacing pipe sizes larger than this maximum value with the potential to bypass containment. The systems that meet this screening criterion are analyzed further to identify specific ISLOCA initiators and scenarios. The identified scenarios are developed in sufficient detail to guide the team in obtaining detailed information during a subsequent extended plant visit.

2.2 Gathering of Detailed Plant-Specific Information

An extended visit to the plant allowed the analysts to gather the information needed to complete the above reviews and to develop and analyze the candidate ISLOCA scenarios. Members of the team that developed the candidate scenarios obtained information by interviewing plant personnel and walking down the systems of interest. This task was performed in conjunction with a special ISLOCA team inspection conducted by the Office of Nuclear Reactor Regulation of the NRC. The types of information that were obtained during this visit include detailed information on

- Hardware that would be involved in an ISLOCA, for example, data on control

valves, relief valves, piping, flanges, pumps, and heat exchangers

- Procedures, guidelines, and practices followed by plant personnel during startup, normal power operation, and shutdown of the plant, as well as detailed information on maintenance and in-service testing
- Factors that could influence performance of plant personnel as related to initiation, detection, diagnosis, prevention, or mitigation of an ISLOCA.

2.3 Development of Event Trees

After the plant-specific information was collected, a final list of low-pressure interfaces and candidate scenarios was compiled and the detailed accident sequence analysis begun. This analysis was a joint effort of PRA and human factors specialists. The scenarios were modeled using (primarily) component level event trees combining the hardware faults and human errors that constitute each sequence in the scenario. In general, each event tree comprised three phases:

1. The initiating events, which are those combinations of failures, both hardware and human related, that breach the RCS pressure isolation boundary and expose the low-pressure interfacing system to the RCS.
2. The rupture events, which model a break in the interfacing system, its size, and location.
3. The post-rupture events, which model the performance of the control room and auxiliary operators in recovering from or mitigating the consequences of the ISLOCA. The list of possible ISLOCA sequences contains both hardware-based sequences (as found in typical PRAs) and sequences initiated by human error. The potential human errors in both types of sequence comprised errors of omission, commission, and pre-existing or latent errors.

2.4 Estimation of Rupture Potential

It is important to realistically assess the performance of those components designed for low-pressure conditions that are exposed to the high pressures associated with an ISLOCA. The basic approach for performing this assessment is

- An event tree model is built that models the failure mode of each of the important low-pressure components. This model is input to the EVNTRE computer code,⁶ which was developed as part of the *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants* program specifically for the evaluation of complex event trees.²
- The failure probability of each hardware component of the interfacing system is described by a lognormal distribution with a specified median failure pressure and logarithmic standard deviation.
- Thermal-hydraulic response of the systems is simulated to estimate the pressure distribution in the system based on the expected initiating event, initial primary system conditions, and the expected performance of relief valves designed to protect the systems from overpressurization.
- Each node in the event tree is evaluated by (a) randomly sampling a failure pressure from the failure-pressure distribution of the appropriate component and (b) comparing this sampled failure pressure with a randomly (and independently) sampled system pressure. The system pressure was randomly sampled based on the expected operating conditions, assuming a normal distribution of system pressure with a calculated mean and estimated standard deviation. If the sampled component failure pressure is below the sampled system pressure, the component is assumed to fail and vice versa. Each component in the low pressure-rated system is evaluated in this manner until all nodes in the event tree have

been examined. This process is repeated 10,000 times in a Monte Carlo process, which is feasible in terms of computer time because of the relatively small size of the EVNTRE model.

- The output of this Monte Carlo simulation is used to estimate the relative frequency of equipment failure due to system overpressurization.

The component and piping failure pressures and distributions used in the rupture calculations were developed from an independent structural analysis performed by Impell Corporation (see Appendix I). Not only were failure pressures calculated for components and piping in the interfacing systems but likely leak rates and leak areas, also. In this respect, bolted flanges were found to be unique in that there are actually two failure pressures of interest. First, there is the estimated gross leak pressure (GLP), at which a measurable leak area develops. At pressures below the GLP, seepage around the flange gasket is possible, but the leak rates are very small (of the order of mg/s). Once pressure exceeds the GLP, the flange bolts begin to stretch (elastically deform) and the flange surfaces begin to separate. At some higher pressure (P_o), the bolts begin to deform plastically. At this point, large leak areas begin to develop with correspondingly large leak rates. These three pressure regions (below GLP, between GLP and P_o , and greater than P_o) are associated with three leak sizes: spray leaks, small leaks, and large leaks, respectively.

Other components, specifically pipes and vessels (including heat exchangers), were estimated to fail catastrophically at their failure pressure. This is the case because the likely failure mode includes the formation of cracks that grow very rapidly, resulting in large ruptures.

2.5 Human Reliability Analysis

The predominant human errors for each scenario in the ISLOCA PRA were modeled using the techniques of human reliability analysis (HRA). HRA is a methodological tool used for prediction, evaluation, and quantitative analysis

Approach

of work-oriented human performance. As a diagnostic tool, HRA can estimate the error rate anticipated for individual tasks and can identify where errors are likely to be most frequent.

The general methodological framework for the ISLOCA HRA was based on guidelines (under development) from the NRC-sponsored Task Analysis-Linked Evaluation Technique (TALENT) Program,⁷ which recommends the use of task analyses, time-line analyses, and interface analyses in a detailed HRA. The technique for human error rate prediction (THERP) from the *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications*,⁸ recommends similar techniques and, in addition, provides a data base that can be used for estimating human error probabilities (HEPs). Finally, the ISLOCA HRA integrated the steps from the *Systematic Human Action Reliability Procedure* (SHARP)⁹ and *A Guide for General Principles of Human Action Reliability Analysis for Nuclear Power Generation Stations*.¹⁰

From this combination of approaches, the analysts identified 11 basic steps to be followed in performing the HRA:

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

Because most of the human actions in this HRA entailed the use of written normal, abnormal, and emergency operating procedures, THERP-type HRA event trees were used to model most of the human actions in the detailed analysis.⁸ However, not all ISLOCA scenarios were best represented by THERP event trees alone. In those cases, HRA fault trees were used in conjunction with the THERP event trees. The fault trees and THERP event trees were used in a detailed analysis to estimate the probability of human error for each of the dominant human actions.

Traditionally, human reliability analysts model human performance through the use of an event tree like that shown in Figure 2, with operator error generally placed along the descending right branches of the event tree and successful operator

Sequence 1A & 1B
OP-FTC-1 (Path 2, reliefs lift)

Figure 2. Example of HRA event tree.

Approach

actions sequenced on the left side of the tree. For example, on the top left, event a [RO (reactor operator) diagnoses need to isolate ND (plant designation for low-pressure safety injection/residual heat removal (RHR) system)] is the success path. Failure to accomplish this task is modeled as event A (RO fails to diagnose need to isolate ND). When a second operator is involved, such as in event B [CRS (control room supervisor) fails to diagnose need to isolate ND], the action of this second operator may be modeled in a recovery branch, as shown in Figure 2. Since the CRS is in the control room in this scenario, the CRS also has an opportunity to diagnose the need to isolate the RHR or ND system. If successful, this becomes a recovery action because it would bring the model back to the success path (via the dotted lines in Figure 2).

The basic, or unmodified, HEPs for branches in the HRA event trees were estimated using techniques from the *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Application* and human cognitive reliability (HCR).¹¹ These basic HEP estimates were then revised by using performance shaping factors (PSFs) to more realistically model the work process at the plant. Each PSF was either positive or negative and, accordingly, either decreased or increased the likelihood of a given human error. For example, an analog meter, like a pressure gauge, if it does not have easily seen limit marks, would be judged to be a negative PSF. Thus, there would be a higher-than-normal probability for error in reading the gauge. Individual PSFs were derived from task analyses, time-line analyses, evaluation of the human/machine interface, and direct observations of operator performance. They are presented as part of the *ISLOCA Inspection Report*.¹²

Specific PSFs that were investigated include

- The quality of the human/machine interface
 - Written procedures (emergency, abnormal, maintenance, etc.)
 - P&IDs
 - Response times for systems and personnel
 - Communication requirements
 - Whether the operator actions were skill, rule, or knowledge-based
 - Crew experience
 - Levels of operator stress in different scenarios
 - Feedback from the systems in the plant
 - Task dependence and operator dependence
 - Location of the task (control room, auxiliary building, etc.)
 - Training for individual operator actions including those required in ISLOCA situations.
- Finally, the combination of all identified failure paths (i.e., sequences that included either single or multiple human errors leading to a failure of the action modeled by the HRA tree) gave the failure probability for the top-level action modeled in the HRA tree. The guidelines of THERP were followed in identifying the individual error paths. Probabilities for each unique error path were calculated by multiplying each HEP on a given error path by other HEPs on the same path.
- As depicted by Figure 2, each human error event tree may have several unique error paths. For example, event A and event B constitute an error path in which the RO fails to diagnose the need to isolate the RHR system. In a similar manner, failure path A–b–C–D models a sequence where the RO fails to diagnose the need to isolate the RHR system, then recovers (e.g., the CRS diagnoses the need to isolate) from this first failure (event b), only to have both the RO and CRS omit (fail) the steps that would close the RHR isolation valves. Probabilities for each unique error path were calculated by multiplying each HEP on a given error path by other HEPs on

the same path. For example, the error rate for path A–B would be calculated by multiplying the HEP of failure A by that for failure B, resulting in a nominal HEP for that specific path. Other error paths for this event tree include A–b–c–E–F, a–c–E–F, and a–C–D. The individual error path failure probabilities were then summed to give the total event tree failure probability. Comprehensive details of this process are provided in Appendix C for each event, and the results are summarized in Section 4.2.

2.6 Quantification of Event Trees

The top events on the ISLOCA accident sequence event trees are quantified by separate calculations that generate the conditional probabilities of occurrence of each event for each path through the tree. The means of obtaining the rupture event probabilities and the probabilities relating to failure of plant personnel have already been discussed. Hardware failure probabilities were developed using the data base documented in Appendix B of *Assessment of ISLOCA Risks—Methodology and Application to a Babcock and Wilcox Plant*.⁵ The ISLOCA event trees were constructed using the ETA-II PC computer code.¹³

2.7 Consequence Evaluation

The ISLOCA CDFs are multiplied by the corresponding consequences (conditional on the occurrence of core damage) calculated using the MELCOR Accident Consequence Code System (MACCS) to obtain the ISLOCA annual risk estimates.¹⁴ The conditional consequences were generated with MACCS using a hybrid input deck. The fission product source terms were obtained from the SEQSOR parametric source term generation code.¹⁵ Like the analyzed plant, Sequoyah (the plant for which SEQSOR was developed) has a four-loop Westinghouse nuclear steam supply system (NSSS) with an ice condenser containment, and the source terms generated are the ones identified with the containment bypass V-sequence in NUREG-1150.² The site information was taken from the Surry MACCS input deck used in the NUREG-1150 program. The Surry site was chosen by reviewing the Sandia Siting Study and calculating an average site based on weather-weighted population density.¹⁶ This average population density was then compared to the five NUREG-1150 sites² and Surry was chosen because it most closely matched the calculated average population density. Further details of the consequence calculations can be found in Appendix G.

3. DESCRIPTION OF THE INTERFACING SYSTEMS

The unit analyzed is a 3,411 MWt pressurized water reactor (PWR), with a four-loop Westinghouse NSSS. It is equipped with an ice condenser containment and a separate auxiliary building and turbine building. An overview of the interfacing systems is presented in the following section. For more details on the interfacing systems, see Appendix A.

3.1 Interfacing Systems

All interfacing systems were screened to identify those systems that required further evaluation. The first criterion used in screening was that any system with an interfacing pipe diameter larger than one inch should be evaluated. The one-inch pipe size was selected based on an estimation of the discharge from a one-inch high-pressure pipe break, which was about 200 gpm. A 200-gpm leak rate outside of the containment is considered to be critical based on the capacity of the refueling water storage tank (RWST) (minimum Technical Specification volume of 350,000 gal), the capacity of a single charging pump (~100 gpm), and the normal makeup rate to the RWST (~150 gpm). Based on these considerations and the number of hours it would take for the plant to achieve cold shutdown (conservatively assumed to be about 10 hours), leak rates of 200 gpm or less were judged not to be risk-significant. The second criterion was that systems whose low-pressure portions were isolated from reactor pressure by three or more normally closed valves or periodically leak-tested check valves in series would not be analyzed. The basis for screening out such systems is the low expected frequency of occurrence of failure of the pressure isolation boundary.

The initial screening resulted in the selection of the high-pressure safety injection (NI) discharge lines, the low-pressure safety injection (ND) discharge lines, and the shutdown cooling (ND) suction lines. Figure 3 shows a schematic diagram of the safety injection system hardware configuration and Figure 4 provides similar information for the RHR system. Additional

details on these systems are provided in Appendix A. The safety injection system interface comprises eight separate reactor pressure vessel (RPV) injection lines, four each to the RCS (the RCS is referred to as the NC system by the plant) hot legs and cold legs. Starting from the RPV, each injection line contains two check valves in series, an MOV, and the safety injection system pump discharge check valve. The RHR system interface consists of four low-pressure injection lines to each of the RCS cold legs, two high-pressure recirculation lines to the RCS hot legs, and two suction lines from the RCS hot legs used for shutdown cooling.

3.2 Potential ISLOCA Scenarios

Potential scenarios were developed by examining the system interfaces and plant operational information. A team of PRA and human factors specialists was involved in the scenario development. In some cases (e.g., the RHR system injection lines), the sequences are strictly hardware-driven; that is, the ISLOCA potential is a function of the hardware failure rates of the pressure isolation boundary valves. In other cases, (e.g., the RHR system suction lines), human errors can initiate an ISLOCA. Table 1 summarizes the ISLOCA scenarios identified.

3.2.1 Residual Heat Removal System Suction Lines. When the plant operates in one of its shutdown modes (i.e., Modes 4 or 5, as defined in the plant Technical Specifications), the RHR system is used for removing core decay heat. It operates via two 12-inch lines connected to the RCS hot legs, with each RHR system loop isolated from the RCS by two 12-inch motor-operated gate valves in a series configuration.

Three potential ISLOCA scenarios involving the RHR system suction lines were evaluated. First, there are two scenarios that could occur during plant startup. In these scenarios (one with the RCS solid, the other with a steam bubble in the pressurizer), the RHR system suction isolation valves are inadvertently left open as RCS pressure

Figure 3. Simplified flow diagram of safety injection system.

Figure 3. (continued).

Figure 4. Simplified flow diagram of RHR system.

Table 1. List of potential ISLOCA scenarios.

Interface	Description	Notes
RHR suction lines	Failure to close valves during startup or premature opening during shutdown	Three scenarios modeled: two startup and one shutdown (SEQs-1A, 1B, and 2)
RHR cold leg injection lines	Failure of two check valves with normally open MOV	Only hardware failures considered (SEQ-3)
Safety injection cold leg interface	Failure of two pressure isolation check valves with normally open MOV, plus failure of safety injection pump discharge check valve	Only hardware failures considered (SEQ-4)
Safety injection hot leg interface	Failure of two pressure isolation check valves and stroke test of normally closed MOV, plus failure of safety injection pump discharge check valve	Only hardware failures considered (SEQ-5)
RHR hot leg recirculation lines	Failure of two check valves and normally closed MOV	Only hardware failures considered (SEQ-6)

is raised above 385 psig. In the third scenario, which occurs during plant shutdown, the RHR system suction isolation valves are opened prematurely, with RCS pressure and temperature above 385 psig and 350°F. In each of these scenarios, the RHR system is exposed to pressures and/or temperatures that could possibly result in the failure of low pressure-rated components in the system.

3.2.2 Residual Heat Removal System Cold Leg Interface. This scenario involves a failure of the two pressure isolation check valves in one of the low-pressure injection lines from the RHR system to the RCS cold legs. Each RHR system loop connects to the safety injection system through a normally open MOV. The lines then divide to each RCS cold leg (4); each of these injection lines has two check valves in series, as shown in Figure 4.

3.2.3 Safety Injection System Cold Leg Interface. This scenario involves a failure of the two pressure isolation check valves in one of the high-pressure injection lines from the safety injection system to the RCS cold legs. There are two safety injection system trains, with the trains cross-connected to the cold leg injection lines via normally open MOVs. The common cold leg injection line passes through another normally open MOV before splitting into four separate lines, with one line injecting into each RCS cold leg. Each of the four injection lines has two pressure isolation check valves in series.

3.2.4 Safety Injection System Hot Leg Interface. This scenario involves failure of the safety injection system/RCS pressure isolation boundary at the high-pressure injection interface with the RCS. As shown in Figure 3, there are

four injection lines, with two check valves in series in each line. There is a piping code class break at normally closed MOVs NI-121A and 152B, with a change in pressure rating from 2,500 psia on the RCS side to 1,765 psia on the safety injection system side. These two MOVs are stroke-tested quarterly under the plant's in-service testing (IST) program. Because the safety injection system discharge piping is hydrostatically tested to a pressure greater than rated RCS pressure, this code class break is not expected to be of concern in terms of an ISLOCA. The transition of concern occurs at the safety injection pumps, where the discharge-side

piping is rated to 1,765 psia, but the suction-side is rated to only 235 psia.

3.2.5 Residual Heat Removal System Hot Leg Recirculation Lines. This scenario involves a failure of the RCS/RHR system pressure isolation boundary in the hot leg recirculation portion of the RHR system. As shown in Figure 4, there are two injection lines, with two check valves in series in each line. The piping code class break (high/low pressure) occurs on the RHR system side of normally closed MOV NI-183B. This MOV is stroke-tested as part of the plant's IST program only during cold shutdown.

4. RESULTS

Because of the unique nature of the ISLOCA sequence, a detailed understanding of the capabilities of the plant hardware and personnel is needed to accurately analyze the ISLOCA challenge. For this report, an ISLOCA is considered to involve a loss of reactor coolant outside containment. Since the supply of water available for makeup to the RCS is essentially limited to the available inventory in the RWST, a high-priority item for the control room operators should be to expeditiously isolate the rupture and terminate the loss of reactor coolant. If the rupture were isolated in a timely manner and the loss of RCS inventory terminated, the plant could be cooled down safely using the auxiliary feedwater (AFW) system (secondary cooldown) or the RHR system (primary cooldown).

Before discussing the detailed results, some general comments can be made that are applicable to all the postulated ISLOCA scenarios. During the course of the plant visit, particular attention was paid to the issue of local environmental effects resulting from ruptures in the interfacing systems. At the time of the plant visit, the probabilistic system rupture calculations had not been completed; hence, a general survey was made of the interfacing system flow paths to *qualitatively* estimate the impact of ruptures on equipment in various locations. This survey included walk-downs of the emergency core cooling systems (ECCS) systems to examine likely break locations. For example, the assumption was made for this analysis that all equipment in the compartment where a break occurs will be rendered unavailable for use in isolating/mitigating the ISLOCA. Therefore, equipment in compartments judged to be candidate locations for an ISLOCA break was inventoried. This survey could not verify that the ECCS are adequately separated such that any postulated rupture would not affect redundant ECCS trains. In particular, in the case of the 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 piping break and blowdown of steam 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 after a rupture had occurred. 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. Conversely, a break in one of the RHR heat exchanger rooms was judged to be unlikely to disable redundant equipment. The basis for this plant-specific judgment is the presence of an access hatch in the heat exchanger compartment overhead that would open to relieve pressure to the environment (external to the auxiliary building) at a small differential pressure. However, we stress that these conclusions are not based on mechanistic heat and mass transfer calculations and are therefore qualitative in nature. This issue is still under investigation at the Idaho National Engineering Laboratory (INEL).

4.1 Event Trees

The following sections describe the event trees developed for the postulated ISLOCA scenarios. The event trees are quantified on a yearly basis, as reflected in the frequency of the initiating event. The event trees are constructed such that the downward branch depicts the failure event listed at the top of the event tree and the upward branch denotes the complement of the event (typically success). The top events are a combination of individual component failures, human errors, and functional failures that describe the progression of the ISLOCA from initiation to core damage or recovery.

All event tree quantification is performed using mean failure probabilities. An uncertainty analysis of CDF is presented in Section 4.5. In addition, sensitivity studies were performed for selected issues that are believed to dominate the risk or about which there is significant uncertainty.

Finally, each event tree end state was assigned to one of the source term bins as follows.

- **OK**—No overpressurization of the low-pressure system occurred (no fission product release).
- **OK-op**—Scenario results in overpressurization of the interfacing system but the system does not rupture or leak (no fission product release).
- **LK-ncd**—Scenario results in RCS leakage from the interfacing system, through either a break or an open relief valve, but severe core damage (sufficient to cause offsite health effects) does not occur because the leak is either isolated before core uncover or the leak is too small to interfere with core cooling (no fission product release).
- **LOCA-ic**—Identifies scenarios that produce a loss-of-coolant accident (LOCA) inside containment. Because these sequences are enveloped by the design basis analysis of the plant, they are not fully developed on the event trees and are not considered to be core damage events for the purposes of this analysis.
- **REL-mit**—An ISLOCA with core damage occurs but the radioactive fission product release is mitigated through some means, such as scrubbing through an overlying water pool or general area fire sprays in the auxiliary building.
- **REL-ig**—An ISLOCA with core damage occurs and results in a large unmitigated radioactive release.

4.1.1 Residual Heat Removal System/Reactor Coolant System Interfacing LOCA During Startup (Solid)—SEQ-1A.

Figure 5 shows a simplified flow diagram of the RHR system. Figure 6 shows the event tree for an interfacing LOCA between the RCS and RHR system during startup (Modes 5 and 4) with the RCS solid. During such a startup, the operators

will be using the charging pumps to slowly raise RCS pressure by collapsing the air that is trapped in the steam generator U-tubes. This process usually lasts for several hours, with the RCS pressure rise being hyperbolic with time as long as the air in the U-tubes can be modeled as an ideal gas, and the compression is approximately isothermal.

The initiating event for this scenario is a plant startup from Mode 5 in which the pressurizer and the RCS are filled with water (solid), except for air trapped in the steam generator U-tubes that was not removed in the preceding RCS fill-and-vent. The RHR system suction isolation valves from the RCS are open, with the RHR system operating to remove decay heat from the reactor. Such a startup is estimated to occur once every 18 months. Two scenarios were developed. In the first scenario, the plant is assumed to be starting up with the wide range RCS pressure transmitters isolated, as in the March 1990 event described in *NRC Special Inspection Report 50-413/90-10 and 50-414/90-10*.¹² In the second scenario, the new configuration controls developed in response to the March event are modeled.

With the RCS solid and temperature below 285°F, the pressurizer pilot-operated relief valves (PORVs) are required by plant Technical Specifications to be in the low-temperature overpressure (LTOP) mode of operation. In this mode, the set point of two of the three PORVs is lowered to approximately 400 psig. Therefore, overpressurization of the RCS and RHR system can be terminated if the PORVs open to relieve pressure. However, in the March 1990 overpressurization event, the PORVs could not function in the LTOP mode because the RCS pressure transmitters from which they receive a signal were isolated.

The RHR system suction isolation valves receive an autoclosure signal when RCS wide range pressure reaches 600 psig. If the valves close successfully, the scenario is terminated. However, with the RCS pressure transmitters isolated as in the March event, the RHR system suction MOVs will not receive an autoclosure signal.

Figure 5. Simplified flow diagram of RHR system.

Figure 6. RHR/RCS interfacing LOCA during startup (solid).

Given failure of both the PORVs and the auto-closure interlock on the RHR system suction isolation valves, the scenario can proceed to core damage through several paths. In each of these paths, the RHR system suction relief valves must fail to open. Given this failure, the RHR system discharge reliefs may open at approximately 600 psig to protect the system from overpressurization. If these relief valves stick open, a drainage path is created from the RCS to the recycle holdup tank (RHT) in the auxiliary building. If the operators fail to detect, diagnose, and isolate this flow path, core uncover and core damage could ensue. If the RHR system discharge relief valves fail to open, and the operators do not detect the abnormally high pressure in the RHR system, a break could occur outside containment. Core damage will occur if the operators then fail to detect, diagnose, and isolate this break.

4.1.2 Residual Heat Removal System/Reactor Coolant System Interfacing LOCA During Startup (Bubble)—SEQ-1B.

Figure 7 shows the event tree used to model an interfacing LOCA between the RCS and RHR system during startup (Mode 4) with a steam bubble in the pressurizer. In this scenario, pressurizer heaters are being used to raise RCS pressure and temperature, under the assumption that RCS temperature is above 285°F, so that LTOP is no longer needed. The RHR system suction isolation valves from the RCS are open initially because the RHR system is being used to remove decay heat from the reactor. The remainder of the scenario is similar to the solid scenario, with the exception that the pressurizer PORVs will not function in the LTOP mode to prevent overpressurization of the RHR system.

4.1.3 Residual Heat Removal System/Reactor Coolant System Interfacing LOCA During Shutdown—SEQ-2.

A risk-significant scenario at the Babcock and Wilcox plant involves premature entry into shutdown cooling, with RCS pressure and temperature above the open permissive set point of the decay heat removal (DHR) system suction isolation valves.⁵ This scenario is considered credible at the Babcock and Wilcox plant because the plant procedures allow operators to bypass the open

permissive interlock for one of the two shutdown cooling isolation valves. This allows an error of commission to be postulated in which, once the decision is made to enter shutdown cooling early, the operators will be led to bypass the interlock for the other valve, also, even though the procedure does not instruct them to do so. For the Westinghouse plant, the HRA does not reveal any circumstances that would lead to an analogous scenario. Therefore, this scenario is not developed further.

4.1.4 Residual Heat Removal System/Reactor Coolant System LOCA At Low-Pressure Injection Cold Leg Interface—SEQ-3.

This scenario involves a failure of the two pressure isolation check valves in one of the low-pressure injection lines from the RHR system to the RCS cold legs (the classic V-sequence). Each RHR system loop connects to the safety injection system through a normally open MOV. The lines then divide to each RCS cold leg (4); each of these injection lines has two check valves in series, as shown in Figure 5. Figure 8 shows the event tree used to model this scenario.

The initiating event for this scenario is internal rupture (with the reactor at power) of a pair of pressure isolation check valves in one of the four injection lines to the RCS. The frequency of this event is calculated under the assumption that the reactor will be at power for the entire year.

In accordance with the plant Technical Specifications, the check valves are leak-tested prior to startup, so they are assumed to be properly seated, eliminating failure-to-close events from consideration. The failure mode of interest is the time-dependent probability that the valves will allow significant (>200 gpm) back-leakage. The assumption is made that either valve in a pair can fail at any time during the year, regardless of whether the other valve has failed.

Given catastrophic internal failure of the check valves, the RHR system discharge relief valves are too small (400 gpm each) to significantly reduce the pressure rise in the RHR system. The

Figure 7. RHR/RCS LOCA at cold leg injection interface.

Figure 8. Simplified flow diagram of safety injection system.

Figure 8. (continued).

flanges in the RHR system are not likely to fail from overpressurization, nor are the valves (see Appendices F and I). The most likely location for a large break is the tube-side cylinder of the RHR system heat exchangers. If there is a rupture in the RHR system, the scenario will proceed to core damage unless the operators can detect, diagnose, and isolate the break.

4.1.5 Safety Injection System/Reactor Coolant System Interfacing LOCA At Cold Leg Interface—SEQ-4. Figure 8 shows a simplified flow diagram of the safety injection system. This scenario involves a failure of the two pressure isolation check valves in one of the injection lines from the safety injection system to the RCS cold legs. Figure 9 shows the event tree used to quantify this scenario. There are two safety injection trains, with the trains cross-connected to the cold leg injection lines via normally open MOVs. The common cold leg injection line passes through another normally open MOV before separating into four lines, with one line injecting into each RCS cold leg. Each of the four injection lines has two pressure isolation check valves in series.

The discharge piping of the safety injection system is rated at 1,765 psia and is hydrostatically tested to 1.5 times this value, so overpressure failure of the safety injection system discharge piping was not modeled. The suction piping to the safety injection pumps is rated at 235 psia, so this piping could be threatened by overpressurization. However, in order for the suction piping to be overpressurized, given failure of the pressure isolation check valves, one of the safety injection pump discharge check valves (NI-116 or 148) must fail to close, also. Because the safety injection pumps are flow-tested quarterly, the discharge check valves periodically see flow in the forward direction but are normally in the “free” state. That is, most of time there is no flow and no differential pressure across the valves. Therefore, in a situation that exposes the valves to reverse flow, they are demanded to close and isolate the safety injection pumps from the RCS.

Three relief valves protect the safety injection system from overpressurization, one on each of

the discharge headers and one on the common suction header. The capacity of these relief valves is probably not sufficient to protect the safety injection system suction piping from overpressurization should the injection check valves fail catastrophically at normal RCS operating pressure; however, these valves relieve to the RHT in the auxiliary building, so their opening may aid operators in detecting overpressurization of the safety injection system.

4.1.6 Safety Injection System/Reactor Coolant System Interfacing LOCA At Hot Leg Interface—SEQ-5. This scenario involves failure of the safety injection system/RCS pressure isolation boundary at the high-pressure injection interface with the RCS. As shown in Figure 8, there are four injection lines, with two check valves in series in each line. There is a change in the piping code class at normally closed MOVs NI-121A and 152B, with a change in pressure rating from 2,500 psia on the RCS side to 1,765 psia on the safety injection system side. As before, because the safety injection system discharge piping is hydrostatically tested to a pressure greater than rated RCS pressure, this code class change is not expected to be of concern in terms of an ISLOCA. The transition of concern occurs at the safety injection pumps where the discharge-side piping is rated to 1,765 psia, but the suction side is rated to only 235 psia. Figure 10 shows the event tree used to model this scenario.

Given failure of the pressure isolation check valves, the low-pressure portion of the safety injection system can be threatened during the quarterly stroke test of NI-121A and 152B. Overpressurization of the safety injection system suction piping would, as above, require a failure on demand of one of the safety injection pump discharge check valves.

4.1.7 Residual Heat Removal System/Reactor Coolant System Interfacing LOCA At Hot Leg Interface—SEQ-6. This scenario involves a failure of the pressure isolation boundary between the RHR system and RCS

Figure 9. Safety injection system/RCS LOCA at cold leg injection interface.

Figure 10. Safety injection/RCS LOCA at hot leg injection interface.

in the hot leg recirculation portion of the RHR system (see Figure 5). As this figure shows, there are two injection lines, with two check valves in series in each line. The piping code class change (high/low pressure) occurs on the RHR system side of normally closed MOV NI-183B. Figure 11 shows the event tree used to model this scenario.

The initiating event is catastrophic failure of a pair of injection check valves with the plant at power. For the RHR system to be overpressurized, normally closed valve NI-183B would have to fail internally or be inadvertently left open. The latter failure mode is not considered credible because the plant Technical Specifications require that this valve be verified to be closed once every 12 hours with the reactor at power. As part of this surveillance, there is a verification that the valve operator is de-energized. The removal of power from the valve operator precludes an inadvertent transfer to the open position. The valve is stroke-tested during cold shutdown only, so the probability of failure equates to the probability that the valve fails internally (no stroke-testing and no inadvertent transfer to the open position).

4.2 Human Reliability Analysis

This section summarizes the results of the ISLOCA HRA efforts. Appendix C provides detailed information regarding HRA fault trees, event trees, tabulated HEP values, and discussions of the HRA process. The HEPs presented as part of the HRA analysis are estimates based upon the best contemporary models and quantitative techniques. As in any HRA, these HEPs must be considered in light of hardware failure information contained within this report and should not be used in isolation.

HRA was used to model the predominant human errors for each scenario in the ISLOCA PRA. As discussed in Section 2.5, HRA is a methodological tool that involves the quantitative analysis, prediction, and evaluation of work-oriented human performance. The ISLOCA HRA diagnosed those factors within the plant's sys-

tems that could lead to less than optimal human performance in the initiation, detection, diagnosis, and mitigation of ISLOCA scenarios. HRA was used as a diagnostic tool to isolate the error rate anticipated for individual tasks and to determine where errors were likely to be most frequent.

Because most of the human actions in this HRA involved the use of various written normal, abnormal, and emergency operating procedures, THERP-type HRA event trees were chosen for modeling most of the human actions in the detailed analysis. However, in several ISLOCA scenarios, HRA fault trees were used in conjunction with the typical THERP event trees to provide the best representation of the modeled events. Detailed analyses were conducted using the fault trees and/or THERP event trees to estimate the error probabilities of the dominant human actions.

Individual error branches for each of the HRA event trees (see Section 2.5 or Appendix C for details) were quantified using techniques from THERP and HCR. Specific human actions on each error branch were assigned an estimate of the basic HEP. These basic HEP estimates were then modified using PSFs to realistically describe the work process at the plant. Finally, possible failure paths (i.e., sequences that included either single or multiple human errors leading to a failure of the action modeled by the HRA tree) were identified and combined to estimate the total failure probability for the HRA tree in accordance with the THERP guidelines. Individual PSFs were derived from task analyses, time-line analyses, evaluation of the human/machine interface, and direct observations of operator performance. The majority of these PSFs were presented in the *ISLOCA Inspection Report* for the analyzed plant.¹² Each PSF was seen as casting either a positive or negative influence on the basic HEP, that is, as either decreasing or increasing the probability of failure for a given human action. For example, some of the positive PSFs in this HRA included the following:

- “The human-machine interface at [the plant] minimized the probability of an

Figure 11. RHR/RCS LOCA at hot leg injection interface.

operator error initiating an ISLOCA event...because most of the pressure isolation valves (PIVs) were check valves which could not be manipulated by the operators. Additionally, for those PIVs which the operators could manipulate at power, there were interlocks and administrative controls to prevent inadvertent operation of these valves at power.”¹²

- “The control room at [the plant] appeared to be well designed in that it had a good control display integration, effective panel layouts, and proper labeling.”¹²
- There were ISLOCA-related training programs and additional training modules being considered for development.

Examples of negative PSFs were

- “Based on past events, the operators did not readily recognize the significance of interfacing systems configurations which could potentially initiate an ISLOCA event.”¹²
- “In the control room, there was only one alarm (the RHR system high discharge pressure annunciator) that would warn the operators of a potential ISLOCA condition and this alarm did not function during a recent RHR system overpressurization event. Although the inspection did identify other instrumentation in the control room which could be used by operators to detect a potential ISLOCA event, their use, without the aid of annunciators or alarms, could easily delay the operator detection of and response to an ISLOCA event.”¹²
- “Observations concerning the relevance of procedures to human performance during detection and mitigation of an ISLOCA included: various procedures did not attribute high pressures in the interfacing systems, or high levels in tanks which receive flow from relief valves found in these interfacing systems to a potential ISLOCA event....The annunciator response procedure (ARP) for high PRT pressure, tempera-

ture, and level did not attribute the cause of the high or increasing PRT level to leaking pressure isolation valves. Instead, the ARP for high PRT level identified the pressurizer PORV as the only probable source of flow into the PRT.”¹²

- Operators’ ISLOCA diagnostic abilities were centered on the presence of EMF-41 (auxiliary building vent high radiation alarm), which has the undesirable effect of increasing operator stress due to initiating preparation for a site evacuation, and the sealing and positive pressurization of the control room.

A detailed HRA was performed for each of the significant scenarios identified in the ISLOCA PRA. Tables 2 through 5 summarize the results of these analyses, which are extensively described in Appendix C. These tables provide the identifier and description for each significant human error, as well as both nominal and mean HEPs. Nominal HEPs in these tables are *median point estimates from a lognormal distribution* (using guidelines from THERP),⁸ while mean values are *mean HEPs from a lognormal distribution*, which were derived using the following formula:

$$\text{mean HEP} = \exp\left(\mu + \frac{\sigma^2}{2}\right)$$

where

$$\mu = \ln \hat{x}$$

$$\hat{x} = \text{the median HEP}$$

$$\sigma = \frac{\ln(\text{error factor})}{1.645}$$

The conversions to mean values were carried out as a result of mathematical concerns where median values from a lognormal distribution should not be multiplied by mean values in estimating the mean CDF, a process that has been followed in some past HRAs. Converting median HEPs to means also allows uncertainties in human error to be included in calculations of the uncertainty in CDF in the PRA (see Appendices C and H for further discussion).

Results

Table 2. Estimated HEPs for Scenario 1A.

Scenario 1A: Overpressurization of RHR system during startup—pressurizer solid.			
Identifier	Human error	Nominal HEP	Mean HEP
ND-OPEN	Operator fails to close isolation valves ND1B & ND2A/ND36B & ND37A	0.0063	0.0082
OP-FTD-1	Operator fails to detect problem: PATH 2 = reliefs stuck open; PATH 3 = reliefs fail to open	0.000063	0.000098
		0.000014	0.000022
OP-FTC-1	Operator fails to isolate RHR PATH 2 = reliefs stuck open PATH 3 = reliefs fail to open	0.00093	0.0024
		0.0199	0.0844
FTD-LOCA	Operator fails to detect LOCA	0.025	0.0595
DIAG-LOCA	Operator fails to diagnose ISLOCA	0.025	0.067
OP-FTC-2	Operator fails to isolate ISLOCA	0.018	0.020

Table 3. HEPs for Scenario 1B.

Scenario 1B: Overpressurization of RHR system during startup—bubble in pressurizer.			
Identifier	Human error	Nominal HEP	Mean HEP
ND-OPEN	Operator fails to close isolation valves ND1B & ND2A/ND36B & ND37A	0.0063	0.0082
OP-FTD-1	Operator fails to detect problem: PATH 2 = reliefs stuck open; PATH 3 = reliefs fail to open	0.000002	0.000005
		0.000014	0.000022
OP-FTC-1	Operator fails to isolate RHR PATH 2 = reliefs stuck open PATH 3 = reliefs fail to open	0.00093	0.0024
		0.0199	0.0844
FTD-LOCA	Operator fails to detect LOCA	0.025	0.0595
DIAG-LOCA	Operator fails to diagnose ISLOCA	0.025	0.067
OP-FTC-2	Operator fails to isolate ISLOCA	0.018	0.020

Table 4. HEPs for Scenario 2.

Scenario 2: Premature entry into shutdown cooling.			
Identifier	Human error	Nominal HEP	Mean HEP
ND-OPEN	Operator opens RHR suction MOVs too soon. (The remaining human errors for this scenario were not analyzed since the HEP for the initiator, ND-OPEN, was estimated as negligible.)	Negligible	Negligible

Table 5. HEPs for Scenario 3.

Scenario 3: Failure of check valves at RCS/RHR cold leg pressure isolation boundary.

Identifier	Human error	Nominal HEP	Mean HEP
FTD-LOCA	Operator fails to detect LOCA	0.026	0.0607
DIAG-LOCA	Operator fails to diagnose ISLOCA	0.025	0.067
OP-FTC-2	Operator fails to isolate ISLOCA	0.018	0.020

Inspection of these tables reveals that HEPs are largest for actions that occur following an interfacing system rupture. These actions are detection of the LOCA, diagnosis of the ISLOCA, and isolation of the break. These comparatively high error rates reflect the following circumstances identified for the plant. First, procedures may not effectively lead operators to the control room indications that are most relevant for detection of an ISLOCA, and may not provide definitive guidance as to applicable procedures or procedure sections. Second, procedures and training rely very much on a single indicator for detection and diagnosis of ISLOCA. This indicator is annunciator EMF-41, an auxiliary building stack high radiation alarm. Procedures may not instruct the operator to verify the presence, absence, or significance of this alarm in a timely manner. Furthermore, the detection of EMF-41 in an alarmed state is identified by the relevant annunciator response procedure as a cue to initiate preparations for a site evacuation while ensuring isolation and positive pressurization of the control room. Therefore, operator workload is increased and significant stress (threat level) is likely to be experienced by the operators at the time when ISLOCA isolation actions are required.

4.3 Quantification of ISLOCA Model

Based on the event trees described in Section 4.1 (and in more detail in Appendix B), the total mean ISLOCA CDF for the plant is estimated to be 2.3×10^{-6} per reactor-year of

operation. Table 6 provides a breakdown of this frequency by sequence and release category. The dominant scenario is hardware dominated, involving failure of the pressure isolation check valves in the RHR system cold leg interface with the RCS (SEQ-3). This scenario is equivalent to the classical event-V category of core damage sequences that has been examined in some past PRAs. The relative insignificance of the human error-initiated sequences is due to the excellent administrative controls and safety culture present at the plant, for example, the practice of not defeating (jumpering out) equipment interlocks during normal operations and the tight control of keys needed to restore power to isolation valves.

4.4 Risk Assessment

As described in Section 2.6, the offsite consequences of ISLOCA core damage sequences were estimated utilizing the V-sequence source term from the June 1989, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants* NUREG-1150 analysis of Sequoyah (see Appendix G for details of the consequence analysis).² The conditional consequences for the base case analysis are listed in Table 7. Based on information from the severe accident risks program that estimated decontamination factors (DFs) for both dry and wet containment bypass releases, a DF of 1 (no decontamination) is assumed for the release from the auxiliary building (large dry release) in the base case. Additional work on estimating DFs for the auxiliary building has been sponsored by the Electric Power Research Institute (EPRI) using the Modular Accident Analysis

Results

Table 6. ISLOCA CDF (per reactor-year).

Scenario	CDF	REL-lg	REL-mit	LOCA-ic	LK-ncd	OK-op
1A No LTOP	ϵ^a	ϵ	0.0	$1.6E-05$	$5.5E-03$	$4.8E-08$
1A LTOP	ϵ	ϵ	0.0	$8.3E-13$	$5.5E-03$	$1.2E-06$
1B	ϵ	ϵ	0.0	$2.0E-06$	ϵ	ϵ
2	ϵ	ϵ	0.0	ϵ	ϵ	ϵ
3	$2.3E-06$	$2.3E-06$	0.0	0.0	$1.4E-05$	ϵ
4	ϵ	ϵ	0.0	0.0	ϵ	0.0
5	ϵ	ϵ	0.0	0.0	ϵ	0.0
6	ϵ	ϵ	0.0	0.0	ϵ	0.0
Totals	$2.3E-06$	$2.3E-06$	0.0	$1.8E-05$	$5.5E-03$	$1.3E-06$

a. $\epsilon < 10^{-8}$.

Table 7. Base case ISLOCA consequences conditional upon core damage.

Mean early fatalities	Mean latent fatalities	Mean 50-mi dose (person-rem)
$9.99 E + 01$	$5.36 E + 03$	$6.12 E + 06$

Program (MAAP) code.^a This work would seem to support DFs for a dry release in the range of 3 to 80, depending on the specific configuration of the auxiliary building. Wet release DFs, either due to a flooded break location or scrubbing by general area fire sprays in the auxiliary building, ranged from 40 on up. However, the MAAP code, when the core flow blockage feature is used (as it was for the EPRI work), tends to predict relatively little hydrogen generation in-vessel.

a. Electric Power Research Institute, *Evaluation of the Consequences of Containment Bypass Scenarios*, EPRI-NP-6586-L, November 1989. This report is not publicly available; however, it was made available to the INEL analysts for review.

Hydrogen generated in-vessel, if released into the auxiliary building, could burn potentially opening pathways for free convective exchange with the outside environment, thus reducing the effective auxiliary building DF. The effect of a credible range of DFs on the offsite consequences is examined in Section 4.5.

When reviewing the ISLOCA consequence and risk estimates, several aspects of this calculation should be kept in mind. Many measures of risk are available and have been used in recent studies. However, to produce these estimates, many sequence-specific and site-specific assumptions must be made, from the cost of land to the warning time available to activate the offsite emergency response plan before a release occurs. These assumptions can have significant effects on the consequences calculated with MACCS. The base case ISLOCA risks are shown in Table 8.

4.5 Uncertainty and Sensitivity Study Results

An uncertainty analysis was performed for the dominant ISLOCA scenario to estimate the range over which the CDF might vary. The details of

Table 8. Base case ISLOCA risk (per reactor-year).

Mean early fatalities	Mean latent fatalities	Mean 50-mi dose (person-rem)
2.3 E – 04	1.2 E – 02	1.38 E + 01

this analysis are presented in Appendix H. The results are shown in Table 9.

The uncertainty in CDF spans approximately four orders of magnitude. The dominant contributor to this uncertainty is the uncertainty in the initiating event: failure of a pair of check valves in one of the low-pressure injection lines.

Because human errors do not dominate the results, the major effort in evaluating the effects of uncertainty, and the sensitivity of risk to major issues, was devoted to the probability of pipe rupture and the effect on risk of a credible range of auxiliary building DFs.

4.5.1 Component Rupture Pressure Uncertainty.

The base case analysis used a logarithmic standard deviation of 0.36 for the pipe rupture pressure distribution (which was modeled as lognormal). As discussed in Appendix I, this value is derived by assuming that the probability of component failure is 10^{-3} when applied stress equals yield stress. Because this may be an overly conservative assumption, two sensitivity cases were examined in which the probability of component failure at yield stress was taken to be 10^{-4} and 10^{-5} . These values correspond to a logarithmic standard deviation for the pipe rupture pressure of 0.30 and 0.26, respectively. The probability of event ND-RUPT was recalculated, producing the results shown in Table 10.

As this table shows, varying the probability of component failure at yield stress does not affect the overall large break probability. The reason for this is the relatively high probability of failure of the RHR system heat exchanger tube-side cylinder, which is not as sensitive to this uncertainty as is piping failure. Because this is the predominant

failure, the ISLOCA CDF is insensitive to the uncertainty in component rupture pressure.

4.5.2 Auxiliary Building DF Uncertainty.

Uncertainty due to lack of knowledge was treated via a sensitivity analysis that examined the effects of a range of credible auxiliary building DFs on fission product source terms and offsite consequences. The details of this analysis can be found in Appendix G. The important aspects and results of this analysis are summarized below.

For the dry ISLOCA sequences, the base case DF for all release classes had a uniform value of 1 and the release was at ground level. The sensitivity analysis involved calculating new fission product source terms with uniform-valued DFs of 5, 10, 50, and 100 for all release classes except the noble gases, for which the base case DF of 1 was retained. For the first two calculations, with DFs of 5 and 10, a ground-level release was assumed. For the last two calculations, with DFs of 50 and 100, the release elevation was specified as 10 m. The reason for the change in release elevation is that industry-sponsored analyses of auxiliary building DFs show that higher DFs, in the absence of water scrubbing, are produced when there is a lack of free convective exchange with the outside environment. This generally corresponds to having an opening high up in the auxiliary building, with a release from the reactor coolant system in the lower elevations of the building.^b

Sensitivity cases also were run to examine the potential effects of auxiliary building fire sprays on the release (there are no such sprays at the plant analyzed in this report). In these so-called wet ISLOCA sequences, the base case DF was specified as a distribution that was sampled upon, as in the *Severe Accident Risks Report* analysis of Sequoyah.² The distribution is shown in Table 11 for the base case analysis of the wet sequences. As

b. Electrical Power Research Institute, *Evaluation of the Consequences of Containment Bypass Scenarios*, EPRI-NP-6586-L, November 1989. This report is not publicly available; however, it was made available to the INEL analysts for review.

Results

Table 9. Results of CDF uncertainty analysis (per reactor-year).

Sequence	5%	50%	Mean	95%
RHR cold leg	5.3 E – 10	5.1 E – 08	2.3 E – 06	6.9 E – 06

Table 10. Probabilities of a large break (ND-RUPT) for various probabilities of failure at yield stress.

Probability of failure at yield	Logarithmic standard deviation for piping	Large break probability
10 ⁻³ (base case)	0.36	0.98
10 ⁻⁴	0.30	0.98
10 ⁻⁵	0.26	0.98

Table 11. Distribution of DF for the base case analysis of the wet ISLOCA sequences as specified in the SEQSOR input.

Distribution of DF (%)	DF
0	5.1E + 03
1	4.5E + 03
5	4.1E + 03
25	1.3E + 02
50	6.2E + 00
75	3.0E + 00
95	1.8E + 00
99	1.7E + 00
100	1.6E + 00

was done for the dry sequences, sensitivity analyses were performed with uniform-valued DFs of 5, 10, 50, and 100 for all release classes except the noble gases, for which the DF remained at the base case value of 1. For the first two sensitivity calculations, with DFs of 5 and 10, a ground-level release was assumed. For the calculations with DFs of 50 and 100, calculations were performed with the release elevation specified both at ground level and 10 m.

Tables 12 and 13 present the mean conditional consequence results for the dry and wet ISLOCA sequence sensitivity cases, respectively. These results are also presented graphically in Figures 12 through 17. Comparisons of the dry and wet sequence consequence results are also presented graphically in Figures 18 through 20.

Table 12. Mean MELCOR accident consequence code system results for each dry ISLOCA sequence sensitivity.

<u>Sensitivity case</u>				
<u>DF</u>	<u>Release elevation</u>	<u>Mean early fatalities</u>	<u>Mean latent fatalities</u>	<u>Mean 50-mi dose (person-rem)</u>
1	0.0	9.99E + 01	5.36E + 03	6.12E + 06
5	0.0	1.63E + 00	1.82E + 03	2.15E + 06
10	0.0	4.03E - 01	1.17E + 03	1.55E + 06
50	10.0	7.12E - 02	5.16E + 02	8.83E + 05
100	10.0	6.10E - 02	3.97E + 02	7.24E + 05

Table 13. Mean MELCOR accident consequence code system results for each wet ISLOCA sequence sensitivity.

<u>Sensitivity case</u>				
<u>DF</u>	<u>Release elevation</u>	<u>Mean early fatalities</u>	<u>Mean latent fatalities</u>	<u>Mean 50-mi dose (person-rem)</u>
Base ^a	0.0	3.69E + 00	1.71E + 03	2.08E + 06
5	0.0	9.92E - 01	1.49E + 03	1.79E + 06
10	0.0	4.01E - 01	9.85E + 02	1.37E + 06
50	0.0	1.76E - 01	4.43E + 02	7.78E + 05
50	10.0	1.18E - 01	4.51E + 02	7.86E + 05
100	0.0	1.59E - 01	3.51E + 02	6.40E + 05
100	10.0	1.06E - 01	3.56E + 02	6.46E + 05

a. Wet sequence base case DF is a sampled distribution as given in Table 11.

Figure 12. Mean early fatality consequence results as a function of DF for the dry ISLOCA sequences.

Figure 13. Mean latent fatality consequence results as a function of DF for the dry ISLOCA sequences.

Figure 14. Mean 50-mi population dose consequence results as a function of DF for the dry ISLOCA sequences.

Figure 15. Mean early fatality consequence results as a function of DF for the wet ISLOCA sequences (all release elevations are 0.0 m).

Figure 16. Mean latent fatality consequence results as a function of DF for the wet ISLOCA sequences (all release elevations are 0.0 m).

Figure 17. Mean 50-mi population dose consequence results as a function of DF for the wet ISLOCA sequences (all release elevations are 0.0 m).

Figure 18. Comparison of dry and wet ISLOCA sequence mean early fatality consequence results as a function of DF (release elevation is 10.0 m for DFs of 50 and 100, otherwise elevation is 0.0 m).

Figure 19. Comparison of dry and wet ISLOCA sequence mean latent fatality consequence results as a function of DF (release elevation is 10.0 m for DFs of 50 and 100, otherwise elevation is 0.0 m).

Figure 20. Comparison of dry and wet ISLOCA sequence mean 50-mi population dose consequences as a function of DF (release elevation is 10.0 m for DFs of 50 and 100, otherwise elevation is 0.0 m).

5. CONCLUSIONS

The methodology for evaluating ISLOCA risk that was developed in *Assessment of ISLOC Risks—Methodology and Application to a Babcock and Wilcox Plant* has been applied to a Westinghouse four-loop plant with an ice condenser containment.⁵ This methodology has been successful in providing insights regarding the *relative* contributions of both hardware faults and human actions to ISLOCA CDF and risk. The results indicate that human errors, latent faults of equipment, and normal procedural tasks can combine in an ISLOCA sequence. However, the methodology also was used to identify potential means of reducing these contributions to risk. Conclusions are presented below, followed by a preliminary discussion of the relationship of these results to the other NPPs.

5.1 Plant-Specific Conclusions

A PRA of the analyzed plant has been performed by the licensee. The full results of this PRA were not made available to the analysts for comparison; however, some of the ISLOCA CDF results were provided, so limited comparisons can be made.

The licensee used a screening analysis to identify those potential ISLOCA paths that required more detailed analysis, similar to what was done for this report. Based on the results of their screening analysis, the licensee focused on essentially the same scenarios as were examined in this report.

The licensee's detailed analysis of the individual scenarios differed in a number of respects from the analysis done for this report. First, the failure of the pressure isolation check valves in the injection lines was treated differently. The licensee assumed that the second check valve could not fail until the first valve failed (the valve nearer the RCS). Thus, the licensee modeled the failure probability of the second check valve as an unavailability, giving the following expression for the probability that both valves fail during the exposure time, τ :

$$P(\text{both valves fail}) = (\lambda\tau)\left(\frac{\lambda\tau}{2}\right).$$

In this report, the assumption was made that either valve can fail at any time during the exposure time, τ . The failure rate, λ , used for this report is higher than the value used by the licensee, based on the expected accelerated corrosion caused by the harsh boric acid environment to which the valves are exposed.³ Also, because the same failure rate is applied to both valves, the failures are treated as correlated. This gives a higher probability for the failure of both valves than would be obtained assuming uncorrelated failure data.¹⁷ The licensee calculated an initiating event frequency of 9.2×10^{-8} /year, whereas this report calculated a frequency of 1.6×10^{-5} /year. Using the higher failure rate from *Interfacing Systems LOCA: Boiling Water Reactors*,⁴ but modeling the valve pair as the licensee did would result in a frequency of 2.9×10^{-7} /year.

The licensee modeled recovery by estimating the probability that the operator would fail to isolate the RHR system by closing the discharge MOVs within one hour after the initiating event (one hour is the licensee's estimate of the time to empty the RWST). A more detailed HRA was performed for this report, factoring in the procedural direction available to the operators. The recovery was subdivided into detection, diagnosis, and isolation, with detailed HRA event trees used to quantify each subdivision (see Sections 2.5, 4.2, and Appendix C). The licensee's estimate of the probability of operator failure to isolate the RHR system is significantly lower than the estimate in this report; however, because the details of the licensee's HRA were unavailable for review, a detailed examination of the differences could not be made.

For the interface at the RHR system suction lines, the licensee modeled only internal rupture of the isolation MOVs; errors of commission were not considered. Errors of commission were considered in the INEL analysis, but the HRA found

Conclusions

sufficient administrative barriers to eliminate this sequence from further consideration.

The licensee's results for the other RCS interfaces are similar to those obtained in this report, with the exception of the treatment of correlated check valve failure data noted above. The licensee found one dominant contributor (frequency $> 10^{-8}$ /year) to ISLOCA CDF: the scenario involving internal rupture of the MOVs in the RHR system suction lines. This sequence had a point estimate CDF of 3.9×10^{-8} /year in the licensee's analysis. Sudden catastrophic internal failure of closed motor-operated gate valves was not considered credible by the INEL analysts and, thus, was not included in this report. Note that the mean frequency of the dominant sequence in this report is 2.3×10^{-6} /year, with a 90% confidence interval of 5.3×10^{-10} /year to 6.9×10^{-6} /year.

The licensee's estimates of the ISLOCA offsite consequences were not available, so comparisons with the INEL results could not be made.

Another difference from the licensee's analysis is the estimation of the rupture probability of components in the RHR system. The licensee performed simple hoop stress calculations for the RHR system piping and the RHR system heat exchanger tubes and found that the hoop stress at normal RCS pressure was below the ultimate tensile strength of the material, and concluded from this that the piping and RHR system heat exchanger tubes would not fail under ISLOCA conditions. The detailed component fragility analysis performed by Impell (see Appendix I) concluded that there was a significant probability that the RHR system heat exchanger tube-side cylinder (to our knowledge, not analyzed by the licensee) would fail, as well as a smaller, but still significant probability of pipe rupture. In contrast, the flange gaskets in the RHR system are unlikely to fail; therefore, the likely break location is the tube-side cylinder of the RHR system heat exchanger, and the break is likely to be large.

In the pressure fragility analysis of the interfacing systems, existing relief valves were found to provide very little protection against the domi-

nant ISLOCA initiator. Typically, relief valves in the interfacing systems are designed to mitigate the occasional pressure transient associated with routine valve realignments and pump starts and stops. The pressures generated in ISLOCA events simply overwhelm the relatively small relief capacity of these valves.

The HRA also found that operator error could contribute to ISLOCA initiators. However, risk-significant human error initiators were judged unlikely during operations involving interfacing systems. This is due, in part, to the existence of significant administrative procedures and related operator training, as well as the presence of well controlled interlocks that prevent the inadvertent operation of those pressure isolation valves under operator control. The HRA found a higher probability for operator error during detection, diagnosis, and isolation of an ISLOCA. This results from an interaction of the following variables:

- Limited number of clear control room indications for ISLOCA
- Procedures that may occupy operators for some time with steps related to the diagnosis of leaks or ruptures inside containment before directing them to indications of ISLOCA
- Significant reliance on a single annunciator, EMF-41, which is not referenced until the last page of the appropriate procedure and that limits the operators' ability to detect and diagnose ISLOCA in a timely manner
- Limited amount of time in the dominant scenario for the detection, diagnosis, and isolation of ISLOCA before core uncover
- Operator exposure to high workload and threat stress at the time when actions to isolate an ISLOCA are needed and directed by procedure.

5.2 General Conclusions

Extreme caution should be exercised when attempting to extrapolate the results of a single

analysis to estimate the performance of the entire commercial nuclear power industry. The analysis of the Westinghouse plant in this report has identified some potential ISLOCA issues, but the completeness and typicality of the results, even for other Westinghouse plants, has not been determined. The analysis of this plant indicates that the most important concern regarding ISLOCA risk centers on the reliability of the pressure isolation check valves, rather than on the plant personnel. However, it is imprudent to conclude that human errors, while not important ISLOCA initiators at the plant analyzed in this report, will not dominate the ISLOCA risk for other plants. Therefore, a major emphasis in any evaluation of ISLOCA should be the assessment of the potential for

human errors. Specifically, this involves judging the adequacy of plant procedures and personnel training and awareness of the potential for and consequences of an ISLOCA. To generalize, the understanding by the plant personnel of the importance of maintaining the pressure isolation boundary, and recognition of the potential for an ISLOCA and its consequences, can have a dramatic impact on ISLOCA risk. In the case of the Westinghouse plant analyzed in this report, the detailed HRA considering these effects eliminated human error-initiated ISLOCAs from detailed consideration, whereas at the Babcock and Wilcox plant,⁵ the result was just the opposite; the dominant ISLOCA sequence was initiated by a human error of commission.

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Appendix A

System Descriptions

Appendix A

System Descriptions

The plant analyzed for this report is a two-unit site. Both units are 3411 MWt pressurized water reactors (PWRs), with a 4-loop NSSS supplied by Westinghouse. Each unit has an ice condenser containment and a separate auxiliary building and turbine building.

The plant differs from the majority of Westinghouse plants in that it has an ice condenser containment, rather than the more common dry containment. It is similar to other Westinghouse plants in the number and type of charging and high pressure safety injection pumps.

A.1 Reactor Coolant System

The reactor coolant system (RCS) transfers energy from the reactor core to the secondary water in the steam generators. The RCS pressure boundary acts as a barrier (one of several) against the uncontrolled release of radioactive material from the reactor core and primary coolant.

During power operation, primary coolant in the RCS is circulated by one reactor coolant pump in each of the four loops. Pressure is maintained within a prescribed band by the combined action of the pressurizer heaters and sprays. RCS inventory is maintained within a prescribed band by the chemical and volume control system, otherwise known as the charging system.

Component Information

- A. RCS
 - 1. Volume: 12,516 ft³ including pressurizer and surge line
 - 2. Nominal operating pressure: 2235 psia
- B. Pressurizer
 - 1. Normal water volume: 1350 ft³
 - 2. Normal gas volume: 450 ft³
- C. Safety valves (3)
 - 1. Set pressure: 2485 psig

- 2. Relief capacity: 420,000 lbm/h each
- D. Pilot-operated relief valves (3)
 - 1. Design pressure: 2485 psig
 - 2. Relief capacity: 210,000 lbm/h each
- E. Steam generators (4)
 - 1. Type: vertical shell and U-tube
 - 2. Model: Westinghouse
- F. Pressurizer heaters
 - 1. Capacity: 1800 kW

A.2 Interfacing Systems

All interfacing systems were screened to identify those systems that needed further evaluation. The criterion used in screening was that any system with an interfacing pipe diameter larger than one inch should be evaluated. The one-inch pipe size was selected based on an estimation of the discharge from a one-inch high pressure pipe break, which was about 200 gpm. A 200-gpm leak rate outside of the containment is considered to be critical based on: the capacity of the RWST (Technical Specification minimum volume of 350,000 gal), the capacity of a single charging pump (~100 gpm), and the normal makeup rate to the RWST (~150 gpm). Based on these considerations and the number of hours it would take for the plant to achieve cold shutdown (conservatively assumed to be about 10 hours), leak rates of 200 gpm or less were judged not to be risk-significant.

The initial screening resulted in the selection of the high pressure safety injection system, the low pressure safety injection system, and the residual heat removal (RHR) system. Figures A.1 and A.2 are schematic diagrams showing the hardware configuration of the safety injection system and Figure A.3 provides similar information for the RHR system.

Figure A.1 Simplified Flow Diagram of Safety Injection System

Figure A.2 Simplified Flow Diagram of Safety Injection System (cont.)

Figure A.3 Simplified Flow Diagram of RHR System

A.2.1 Safety Injection System

The safety injection system provides high pressure coolant injection capability. There are two safety injection system trains, each with one safety injection pump. The safety injection pumps are normally aligned for cold leg injection (to all four RCS cold legs), but are capable of supplying flow to the hot legs, also. The pumps start automatically upon receipt of a safety injection actuation signal. During injection, the RWST supplies borated water which the safety injection pumps deliver to the cold legs via a common discharge header that branches into four lines, one for each RCS loop.

During cold leg recirculation, the safety injection pumps are realigned by operator action to take suction from the discharge of the RHR pumps and deliver flow through the same flow path as during the injection phase. The RHR pumps take suction directly from the containment sump during recirculation.

For hot leg recirculation, the discharge flow path is realigned by operator action to deliver flow to each RCS loop. Safety injection pump A provides flow to hot legs B and C, and pump B provides flow to hot legs A and D.

The safety injection system also contains four cold leg accumulator injection tanks. Each tank contains borated water with a pressurized cover gas. The borated water is forced into the respective cold legs when RCS pressure decreases below the cover gas pressure. The contents of three accumulators are required to limit the peak fuel cladding temperature to less than or equal to the maximum allowable value during a design basis accident.

Testing of the safety injection system is specified in the plant Technical Specifications and in the in-service testing (IST) program. The pumps are flow-tested on a quarterly basis. The discharge MOVs are stroke-tested quarterly. Functional actuation tests of the safety injection system are performed during cold shutdown.

Table A.1 Safety Injection Pump Data

Type	Centrifugal
Design pressure	1750 psig
Design temperature	300° F.
Design flow	400 gpm
Maximum flow	650 gpm
Design head	2540 ft
Head at maximum flow	1650 ft
Discharge head at shutoff	3545 ft

A.2.2 Residual Heat Removal System

The primary purpose of the RHR system is to remove thermal energy from the reactor core and RCS during plant cooldown and refueling operations. The RHR system is used as part of the ECCS for low pressure injection, and a portion of the RHR system flow may be diverted to the spray headers of the containment spray system.

The RHR system pumps are capable of providing relatively large flow rates at relatively low pressures. During the injection phase of an accident, the RHR system is aligned to deliver cooling water directly to the RCS cold legs. During the recirculation phases, the RHR pumps are aligned to provide flow to the safety injection pump suction lines and may be providing flow to the RCS cold legs, the RCS hot legs, or the containment spray headers.

The RHR system, like the safety injection system, consists of two parallel trains. Each train contains one pump, one heat exchanger, and the associated valves, piping, and instrumentation required for operational control. During operation, reactor coolant passes through the respective pump and heat exchanger and is returned to the RCS cold legs. The RHR system train

A suction is from the RCS loop B hot leg and the flow returns to the RCS loop C and D cold legs. Train B takes a suction from RCS hot leg C and returns flow to the loop A and B cold legs.

Two MOVs in series isolate each RHR pump suction from the RCS. These valves are interlocked to prevent inadvertent mispositioning, to ensure RCS pressure is reduced below 385 psig prior to opening the valves, and to automatically close the valves if RCS pressure increases above 600 psig with the valves open.

Table A.2 Residual Heat Removal Pump Data

Type	Vertical centrifugal
Design pressure	600 psig
Design temperature	400° F.
Design flow	3000 gpm
Design head	375 ft

Table A.3 Residual Heat Removal System Heat Exchanger Data

Type	Shell and U-tube
Installation	Vertical
Design pressure	150 psig (shell) 600 psig (tube)
Design temperature	200° F. (shell) 400° F. (tube)
Design flow rate	1.48×10^6 lbm/h (tube)
Shell material	Carbon steel
Tube material	Austenitic stainless steel

Appendix B

ISLOCA Event Trees

Appendix B

ISLOCA Event Trees

This appendix describes in detail the ISLOCA scenarios developed for the Westinghouse plant and the event trees used to calculate the core damage frequency. The event trees are quantified on a yearly basis. The downward branch at each node depicts the failure event listed at the top of the event tree and the upward branch denotes the complement of the event (typically success). The top events represent a combination of individual component failures, human errors, and functional failures that describe the ISLOCA progression.

All event tree quantification is performed using mean failure probabilities. An uncertainty analysis of core damage frequency is presented in Section 4.5 of the main report. In addition, sensitivity studies were performed for selected issues that are believed to dominate the risk or about which there is significant uncertainty.

Finally, each event tree end state was assigned to one of the consequence bins listed below.

OK - No overpressurization of the low pressure system occurred.

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

LK-ncd - Scenario results in RCS leakage from the interfacing system, through either a break or an open relief valve, but no severe core damage (sufficient to cause offsite health effects) occurs because the leak is either isolated before core uncover or the leak is too small to interfere with core cooling.

LOCA-ic - Identifies scenarios that produce a LOCA inside containment. Because these sequences are enveloped by the design basis analysis of

the plant, they are not fully developed on the event trees and these scenarios are not considered to be core damage events.

REL-mit - An ISLOCA with core damage occurs but the radioactive release is mitigated through some means, such as scrubbing through an overlying water pool or general area fire sprays in the auxiliary building.

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

B.1 Residual Heat Removal System/Reactor Coolant System Interfacing LOCA During Startup (Solid) - SEQ-1A

Figure B.1 shows a simplified flow diagram of the RHR system. Figure B.2 shows the event tree for an interfacing LOCA between the RCS and RHR system during startup (Modes 5 and 4) with the RCS solid. During such a startup, the operators will be using the charging pumps to slowly raise RCS pressure by collapsing the air that is trapped in the steam generator U-tubes. This process usually lasts for several hours, with the RCS pressure rise being hyperbolic with time as long as the air in the U-tubes can be modeled as an ideal gas, and the compression is approximately isothermal.

M4-SUS. The initiating event is a plant startup from Mode 5 in which the pressurizer and the RCS are filled with water (solid), except for air trapped in the steam generator U-tubes. The RHR system suction isolation valves from the RCS are open, with the RHR system operating to remove decay heat from the reactor. Such a startup is estimated to occur once every 18 months. Two values were used for this event. In the first case, the plant is assumed to be starting up with the wide range RCS pressure transmitters isolated, as in the March 1990 event described in NRC Special Inspection Reports 50-413/90-10 and 50-414/90-10. The frequency of event M4-SUS is taken to be 0.67/y (one solid startup every 18 months) multiplied by the probability of starting up with the pressure transmitters isolated, which is estimated to be 8.25×10^{-3} , given the configuration controls in place at the time of the March event. This gives a frequency for event M4-SUS of 5.5×10^{-3} in the first case. In

the second case, the new configuration controls developed in response to the March event are modeled. With these modified controls, the probability of starting up with the pressure transmitters isolated is estimated to be 8.25×10^{-5} . Therefore, the frequency of event M4-SUS in the second case is simply taken as 0.67/y.

ND-OPEN. This event models the possibility that RCS pressure will be increased above 385 psig (the point at which the valves are required to be closed by procedure) with the RHR system suction isolation valves open. For the case in which the pressure transmitters are isolated, the probability of this event is estimated at essentially 1.0. If the transmitters are not isolated, the probability is estimated at 8.2×10^{-3} .

PORV-FTO. According to the plant Technical Specifications, the pressurizer PORVs are required to be in the low temperature overpressure (LTOP) mode when the RCS is solid and the temperature is below 285° F. In this mode, two of the three pressurizer PORVs have their opening set point lowered to 400 psig. This event models the failure of the PORVs to open at this set point. The hardware failure probability is estimated at $2.1 \times 10^{-4}/d$, calculated by multiplying the demand failure probability of 3.0×10^{-3} from [B-1] by the estimated beta factor of 0.07 from [B-2].

Note that there also may be a human error contribution to this failure probability, as there was in the March event described in the above NRC reports. This contribution has been quantified for two cases. In the first case, the configuration control system that was in place prior to the March event was modeled. In the second, the analysis modeled the new system developed in response to the March event. For the first case, the failure probability, given that the pressure transmitters needed for LTOP are left isolated during RCS pressurization, is 1.0. In the second case, the failure probability is taken to be the hardware failure probability of 2.1×10^{-4} .

Figure B.1 Simplified Flow Diagram of RHR System

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MOV-FTC. The RHR system suction isolation valves receive an autoclosure signal when RCS wide range system pressure reaches 600 psig. This event models the failure of the valves to close at this set point. Again, two cases are modeled, reflecting the configuration controls in place before and after the March overpressurization event. If the pressure transmitters are isolated, the RHR system suction MOVs will not receive an autoclosure signal, so the failure probability of event MOV-FTC becomes 1.0. In the second case, where the transmitters are not isolated, a demand failure probability of 3.0×10^{-3} is taken from [B-1]. As there are two valves in series, both valves must fail to close to prevent isolation of the RHR system. To model the possible common-cause failure of both valves to close, a beta factor of 0.08 is assumed [B-2]. The failure probability of event MOV-FTC then becomes $(3.0 \times 10^{-3}) * (0.08) = 2.4 \times 10^{-4}/d$.

RV-FT0-1. If the PORVs fail to open, overpressure protection may still be provided by the RHR system suction relief valve, which is set to relieve at least 900 gpm at 450 psig to the pressurizer relief tank (PRT), which is located inside containment. This event models the failure of the relief valve to open. A value of $3.0 \times 10^{-3}/d$ is used from [B-1].

RV-FT0-2. The discharge piping of the RHR system is protected by a relief valve that is set to open at 600 psig. This valve relieves ~400 gpm at full lift to the recycle holdup tank (RHT) in the auxiliary building. This event models failure of this valve to open. As above, a value of $3.0 \times 10^{-3}/d$ is used from [B-1].

RV-FTC. This event models a stuck-open relief valve. If the RHR system suction relief fails to close, water will be discharged from the RCS into containment. Thus, the end result is a LOCA inside containment. However, if the discharge relief fails to close, RCS water will be discharged into the RHT in the auxiliary building, an ISLOCA. The probability of a relief valve failing to close is taken at $3.0 \times 10^{-3}/d$ from [B-1].

OP-FTD. This event models operator failure to detect overpressurization of the RHR system. It is quantified for three different paths through the event

tree. In the first, the RHR system suction relief valve is open (possibly stuck open), so there will be indications in the control room of rising temperature, pressure, and level in the PRT. The human error probability (HEP) for this path is estimated to be 9.8×10^{-5} (see Appendix C for the details of the HEP calculations). In the second, the RHR system suction relief fails to open, but the RHR system discharge relief does open, so the operators must rely on other indicators, such as level and temperature in the RHT, and RHR system pump discharge pressure. The HEP for this path is estimated at 9.8×10^{-5} , also. In the third path, neither relief valve opens, so the operators must rely on indicators such as RHR system pump discharge pressure. The HEP for this path is estimated to be 2.2×10^{-5} .

OP-FTC-1. This event models failure of the operators to isolate the RHR system by closing the suction isolation valves, given that the operators are aware of the RHR system overpressurization. If the operators did not detect the overpressurization, then the assumption is made that they will not isolate the RHR system at this point in the sequence. The HEP for this event is estimated to be 2.4×10^{-3} if a relief valve is open, and 2.2×10^{-2} if no relief valves are open.

ND-RUPT. This event represents a rupture of the RHR system outside containment, given failure of the operator to isolate the RHR system from the RCS. A value of 1.0 is assigned to this event.

FTD-LOCA. This event represents operator failure to detect the LOCA outside containment in time to take action to prevent core uncover. The HEP for this event is estimated to be 5.95×10^{-2} .

DIAG-LOCA. This event models failure of the operator to correctly diagnose that an ISLOCA has occurred, given that he/she has successfully detected the LOCA. The HEP is estimated to be 6.7×10^{-2} .

OP-FTC-2. Given that the ISLOCA is detected and correctly diagnosed, this event represents failure to isolate the break in time to prevent core uncover. The HEP is estimated to be 1.95×10^{-2} .

REL-MIT. Given that core damage occurs, the release of fission products to the environment may still be mitigated by removal mechanisms, such as scrubbing through an overlying water pool or by auxiliary building fire protection sprays. Walkdowns of the auxiliary building and discussions with plant personnel indicate that the break is not likely to be submerged and no general area fire sprays are present in the auxiliary building. Therefore, this event was judged to have a failure probability of 1.0 for all sequences.

B.2 Residual Heat Removal System/RCS Interfacing LOCA During Startup (Bubble) - SEQ-1B

Figure B.3 shows the event tree for an interfacing LOCA between the RCS and RHR system during startup (Mode 4) with a steam bubble in the pressurizer. In this scenario, pressurizer heaters are being used to raise RCS pressure and temperature, under the assumption that RCS temperature is above 285° F, so that LTOP is no longer required.

M4-SUB. The initiating event is a plant startup from Mode 4 in which there is a steam bubble in the pressurizer. The RHR system suction isolation valves from the RCS are open initially, because the RHR system is being used to remove decay heat from the reactor. Such a startup is estimated to occur once per year, so the initiating event frequency is 1.0/y.

ND-OPEN. This event models the possibility that RCS pressure will be increased above 385 psig with the RHR system suction isolation valves open. The HEP is estimated at 8.2×10^{-3} .

MOV-FTC. As described above for the solid scenario, the RHR system suction isolation valves receive an autoclosure signal when RCS wide range system pressure reaches 600 psig. This event models the failure of the valves to close at this set point. The failure probability of $2.4 \times 10^{-4}/d$ was calculated as in the solid scenario.

RV-FTO-1. Overpressure protection may be provided by the RHR system suction relief valve, which is set to relieve at least 900 gpm at 450 psig. The

valves (one on each RHR system train) relieve to the PRT inside containment. This event models the failure of the relief valve to open. Note that with a bubble in the pressurizer, the pressurizer PORVs will be in the normal mode of operation, so they will not open at the LTOP set point of 400 psig. The demand failure probability of 3.0×10^{-3} is taken from [B-1].

RV-FTO-2. Same as in the solid scenario.

RV-FTC. This event again models a stuck-open relief valve. If the RHR system suction relief fails to close, water will be discharged from the RCS into containment. Thus, the end result is a LOCA inside containment. However, if the discharge relief fails to close, RCS water will be discharged into the RHT in the auxiliary building, an ISLOCA. The probability of a relief valve failing to close is taken at $3.0 \times 10^{-3}/d$ from [B-1].

OP-FTD. This event models operator failure to detect overpressurization of the RHR system. It is quantified for three different paths through the event tree. In the first, the RHR system suction relief valve is open (possibly stuck open), so there will be indications in the control room of rising temperature, pressure, and level in the PRT. The HEP for this path is estimated to be 5.0×10^{-6} . In the second, the RHR system suction relief fails to open, but the RHR system discharge relief does open, so the operators must rely on other indicators, such as level and temperature in the RHT, and RHR system pump discharge pressure. The HEP for this path is estimated at 5.0×10^{-6} , also. In the third path, neither relief valve opens, so the operators must rely on indicators such as RHR system pump discharge pressure. The HEP for this path is estimated to be 2.2×10^{-5} .

The remaining events are the same as for the solid scenario.

Figure B.3 RHR/RCS Interfacing LOCA During Startup (Bubble)

B.3 Residual Heat Removal System/RCS Interfacing LOCA During Shutdown - SEQ-2

An important scenario at the Babcock and Wilcox (B&W) plant involved premature entry into shutdown cooling, with RCS pressure and temperature above the open permissive set point of the DHR suction isolation valves. The HRA performed for the Westinghouse plant did not reveal any credible sequence of events that could lead to an analogous scenario. Therefore, this scenario was not developed further.

B.4 Residual Heat Removal System/RCS LOCA At Cold Leg Interface - SEQ-3

This scenario involves a failure of the two pressure isolation check valves in one of the low pressure injection lines from the RHR system to the RCS cold legs. Each RHR system loop connects to the safety injection system through a normally open MOV. The lines then divide to each RCS cold leg (4). Each of these injection lines has two check valves in series, as shown in Figure B.1. Figure B.4 shows the event tree used to quantify this scenario.

CV-L. The initiating event is internal rupture (with the reactor at power) of a pair of check valves in one of the four injection lines to the RCS. The frequency of this event is calculated under the assumption that the reactor will be at power for the entire year.

The check valves are leak-tested in accordance with the plant Technical Specifications prior to startup, so they are assumed to be properly seated, eliminating failure-to-close events from consideration. The failure mode of interest is the time-dependent probability that the valves will allow significant (>200 gpm) backleakage. The assumption is made that either valve in a pair can fail at any time during the year, regardless of whether the other valve has failed.

The valves are further assumed to each have a constant failure rate λ . Because of the environment and service the injection check valves see, they are assumed to experience a higher failure rate than other check valves ($8.7 \times$

$10^{-8}/h$ compared to $1.8 \times 10^{-8}/h$). The probability that one of the valves in the pair fails in a time τ is given by

$$\begin{aligned} P(T \leq \tau) &= \int_0^{\tau} \lambda e^{-\lambda t} dt \\ &= 1 - e^{-\lambda \tau} \end{aligned}$$

For $\lambda \tau \ll 1$, this can be approximated by $\lambda \tau$.

Because the check valves are identical and a separate failure rate is not available for each individual valve, the failure rates of each of the valves are assumed to be correlated. Therefore, for a pair of valves in series, the probability that both valves fail, denoted by Q , cannot be written simply as the product of the individual failure probabilities, q . Instead, the expected or mean value of Q is given by

$$E(Q) = E(Q^2) = \tau^2 E(\lambda^2)$$

and $E(\lambda^2)$ is given by

$$E(\lambda^2) = E^2(\lambda) + \text{Var}(\lambda).$$

The failure rate, λ , is assumed to be distributed lognormally, with a mean value of $8.7 \times 10^{-8}/h$ and an error factor (EF) of 10. Therefore, $E(Q)$, which is the mean probability that two check valves in series will fail, is given by

$$\begin{aligned} E(Q) &= (8760 h)^2 [(8.7 \times 10^{-8}/h)^2 + 4.6 \times 10^{-14}/h^2] \\ &= 4.11 \times 10^{-6}. \end{aligned}$$

Figure B.4 RHR/RCS LOCA At Cold Leg Injection Interface

In this expression, τ has been replaced with 8760 h, the number of calendar hours in a year. If correlation among the failure rates were not accounted for, the failure probability of two check valves in series would be given by

$$\begin{aligned} E(Q) &= [E(q)]^2 = \tau^2 [E(\lambda)]^2 \\ &= (8760 h)^2 (8.7 \times 10^{-8} / h)^2 \\ &= 5.81 \times 10^{-7}. \end{aligned}$$

This is almost one order of magnitude less than the value calculated assuming correlated failure rates.

Because there are four pairs of check valves, the value for one pair, 4.11×10^{-6} , is multiplied by four to give a frequency for event CV-L of 1.64×10^{-5} .

RV-FTO. This event models failure of the RHR system discharge relief valves to open. There is one valve on each RHR system loop, and the loops are normally cross-connected on the discharge side when the plant is at power. Each relief valve has a capacity of ~400 gpm at 600 psig, and relieves to the RHT in the auxiliary building. The failure probability is taken to be $2.1 \times 10^{-4}/d$, calculated by assuming an independent demand failure probability of 3.0×10^{-3} [B-1] and a beta factor of 0.07 [B-2].

OP-FTD. This event models operator failure to detect overpressurization of the RHR system. The relief valves do not mitigate the pressure rise in the RHR system piping significantly (see Appendix D). Therefore, the operators will have essentially no time to detect the overpressurization before failures occur in the safety injection system (small leaks of < 200 gpm have not been included because the operators should have time to detect and isolate leaks of this size). Thus, event OP-FTD has a failure probability of 1.0.

ND-RUPT. This event models breaks in the RHR system, given the failure of the injection check valves. The details of the rupture probability analysis

calculations are given in Appendices F and I. The result of this analysis is that small breaks and flange leaks are highly unlikely. Given the failure of the injection check valves, the break in the RHR system is almost certain to be large, with the most likely location for the break being the tube-side cylinder of the RHR heat exchanger. The average probability of a large break in the RHR system was calculated to be 0.98.

The remaining events were quantified as for the above startup scenarios.

B.5 Safety Injection System/RCS LOCA At Cold Leg Interface - SEQ-4

Figures B.5 and B.6 show a simplified flow diagram of the safety injection system. This scenario involves a failure of the two pressure isolation check valves in one of the high pressure injection lines from the safety injection system to the RCS cold legs. Figure B.6 shows the event tree used to quantify this scenario. There are two safety injection system loops, with the loops cross-connected to the cold leg injection lines via normally open MOVs. The common cold leg injection line passes through another normally open MOV before splitting into four separate lines, with one line injecting into each RCS cold leg. Each of the four injection lines has two pressure isolation check valves in series.

CV-L. This event models the failure of one of the four pairs of injection check valves. The calculation is identical to that for the analogous event in the RHR system scenario in Section B.3, giving a frequency of $1.64 \times 10^{-5}/y$.

Figure B.6 Simplified Flow Diagram of Safety Injection System (cont.)

Figure B.6 Safety Injection/RCS LOCA AT Cold Leg Injection Interface

DCV-L. The discharge piping of the safety injection system is rated at 1765 psia, and is hydrostatically tested to 1.5 times this value, so overpressure failure of the safety injection system discharge piping was not modeled. The suction piping to the safety injection pumps is rated at 235 psia, so this piping could be threatened by overpressurization. However, in order for the suction piping to be overpressurized, one of the safety injection pump discharge check valves (NI-116 or 148) must fail to close. Because the safety injection system discharge piping is cross-connected during normal operation, Event DCV-L models the failure on demand of either of these check valves. Because the safety injection pumps are flow-tested quarterly, the discharge check valves periodically see flow in the normal direction, but are normally in the "free" state. That is, most of the time there is no flow and no differential pressure across the valves. Therefore, in a situation that exposes the valves to reverse flow, they are demanded to close and isolate the safety injection pumps from the RCS. The failure probability is simply the estimated probability that a check valve fails to close on demand of 1.0×10^{-3} [B-1] multiplied by 2.

RV-FT0. Three relief valves protect the safety injection system from overpressurization, two on the discharge headers and one on the suction header. The capacity of these relief valves is probably not sufficient to protect the safety injection system suction piping from overpressurization should the injection check valves fail at normal RCS operating pressure; however, these valves relieve to the RHT in the auxiliary building, so their opening may aid operators in detecting overpressurization. The probability that none of the three valves opens is taken to be

$$(3.0 \times 10^{-3}/d) * (0.07) * (0.2) = 4.2 \times 10^{-5}/d.$$

The remaining events on the tree are the same as described for the scenarios above. Screening values have been retained (1.0 for rupture of the safety injection system and 0.5 for all HEPs), because the core damage frequency contributed by this scenario is already very low.

B.6 Safety Injection System/RCS LOCA At Hot Leg Interface - SEQ-5

This scenario involves failure of the safety injection system/RCS pressure boundary at the high pressure injection interface with the RCS. As shown in Figure B.5, there are four injection lines, with two check valves in series in each line. There is a code class break at normally closed MOVs NI-121A and 152B, with a change in pressure rating from 2500 psia on the RCS side to 1765 psia on the safety injection system side. As before, because the safety injection system discharge piping is hydrostatically tested to a pressure greater than rated RCS pressure, this code class break is not expected to be of concern in terms of an ISLOCA. The transition of concern occurs at the safety injection pumps, where the discharge-side piping is rated to 1765 psia, but the suction side is rated to only 235 psia. Figure B.7 shows the event tree used to model this scenario.

M1. Event M1 represents plant operation at power (Mode 1). The assumption is made that the plant will be at power for the entire year (four quarters). This is multiplied by two because there are two safety injection system trains.

CV-L. Event CV-L represents internal rupture of one of the pairs of injection check valves. Because the initiating event is quantified on a quarterly basis, the exposure time, τ , in the calculation of the check valve failure probability is taken to be 2190 h. The probability that a pair of check valves fails is, using the same data as before, 2.57×10^{-7} (per quarter). This is multiplied by two to get the failure probability of event CV-L, since there are two pairs of check valves in each injection line (the fact that there are four injection lines was taken into account in the calculation of the initiating event frequency).

MOV-OPEN. This event models the probability that NI-121A, which is normally closed, is open. Because the valve is stroke-tested quarterly, the conditional probability of this event is 1.0. Again, the fact that there are two stroke tests per quarter was accounted for in the calculation of the initiating event frequency.

Figure B.7 Safety Injection/RCS LOCA At Hot Leg Interface

The remaining events on the tree are the same as described for the scenarios above. Screening values have been retained (1.0 for rupture of the safety injection system and 0.5 for all HEPs), because the core damage frequency contributed by this scenario is already very low.

B.7 Residual Heat Removal System/RCS LOCA At Hot Leg Interface - SEQ-6

This scenario involves a failure of the RCS/RHR system pressure boundary in the hot leg recirculation portion of the RHR system. As Figure B.1 shows, there are two hot leg injection lines, with two check valves in series in each line. The code class break (high/low pressure) occurs on the RHR system side of normally closed MOV NI-183B. Figure B.8 shows the event tree used to model this scenario.

CV-L. The initiating event is failure of a pair of injection check valves with the plant at power. The frequency is calculated with the same data as before, giving a frequency of $8.22 \times 10^{-6}/y$, because there are two injection lines.

MOV-FAIL. This event models the internal failure of normally closed MOV NI-183B, allowing the RHR system to be overpressurized. The valve is stroke-tested under the IST program during cold shutdown only, so the probability that the valve will be open with the reactor at power is considered to be negligible. This failure is modeled as an unavailability, using an internal rupture failure rate of $1.7 \times 10^{-8}/h$, and an exposure time of 8760 h (1 y). This failure rate is taken from the EG&G LER compilation, assuming 10% of the observed internal leaks are large. Inadvertent transfers to the open position are not modeled, because power is removed from the valve in accordance with the plant Technical Specifications. Similarly, the probability that the valve is open while indicating closed is assumed to be negligible, because Technical Specifications require that the valve be verified to be closed once every 12 hours during power operations. Under these assumptions, the failure probability becomes

$$P(failure) = \frac{1}{\tau} \int_0^{\tau} (1 - e^{-\lambda t}) dt$$

$$\approx \frac{\lambda \tau}{2}.$$

Inserting the values listed above gives a failure probability of 7.45×10^{-5} .

RV-FT0. This event models failure of relief valves ND-31 and 64 to open. These are the RHR system discharge relief valves. They relieve to the RHT in the auxiliary building, and have a capacity of 400 gpm each.

The remaining events on the tree are the same as described for the scenarios above. Screening values have been retained (1.0 for rupture of the safety injection system and 0.5 for all HEPs), because the core damage frequency contributed by this scenario is already very low.

Figure B.8 RHR/RCS LOCA At Hot Leg Injection Interface

References

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Appendix C

Human Reliability Analysis

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ISLOCA Human Reliability Analysis

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

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

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

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

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

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

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

- mechanical and electrical system descriptions (e.g., the reactor coolant, residual heat removal, safety injection, and chemical and volume control systems),
- a sourcebook of plant systems and schematic drawings (NRC-03-87-029, FIN D-1763 [C-5]),
- the plant's Final Safety Analysis Report (FSAR),
- the plant's Technical Specifications [C-6],
- plant procedures (operating, abnormal, emergency, maintenance, administrative, etc.), station directives, and operational practices,
- piping and instrumentation diagrams (P&IDs),
- the types, capacities, and locations of check valves/motor-operated valves identified as being pressure isolation valves,
- training materials such as flow charts, lesson plans, etc.,
- crew composition (for control room and auxiliary building operators) and level of training/experience,
- significant precursor information from general ISLOCA-related LERs, as well as a plant-specific ISLOCA-related LER (NRC Inspection Report 50-412/90-10 and 50-414/90-10 [C-7]).

This training/familiarization process for the plant's systems was enhanced by a two-week visit to the plant.

Step #3 required that significant human actions and interactions be incorporated into the ISLOCA PRA analysis. This was accomplished through an extensive data collection process during the plant visit. As part of the data collection, the utility provided written procedures, training materials, and P&ID drawings. This data was supplemented by interviews and detailed task

analyses with both licensed and non-licensed nuclear operators in the plant. Observations of control room personnel, the use of the utility's simulator, and system walkdowns with licensed and non-licensed operators supplied additional information.

The initial plant models were constructed in the fourth step. Using the plant-specific data gathered in Step #3, the HRA analysts worked with the PRA analyst and systems engineering personnel to specify human actions related to the postulated ISLOCA scenarios. As a consequence of the plant's ISLOCA-related LER and several findings from the earlier ISLOCA PRA of the B&W plant, significant attention was given to latent, or precursor, human errors during normal operations which could lead to inoperable equipment or misaligned valves. Examples of these precursor actions included: jumpering of valves to defeat protective interlocks, in-service testing practices, maintenance procedures, and the operation of pressure isolation valves (PIVs) from the auxiliary shutdown panels and standby shutdown facility.

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

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

The output from the preceding step (i.e., Step #5) was a group of important human actions, for specific ISLOCA scenarios, which were described in generic, functional terms (e.g., operators recover system). In the sixth step, the analysts expanded the description of each of these key human actions from a functional description into specific operator tasks and subtasks (e.g., operator opens valve ND-2A, or operator closes valve ND-36B). By breaking down the human actions into specific tasks and subtasks associated with individual equipment and procedures, the analysts began to identify specific failure modes, root causes, and failure effects. The description of each task/subtask was enhanced by referencing significant performance shaping factors (PSFs) which affected a given task. These PSFs were derived from the task analyses, time line analyses, evaluation of the human-machine interface, and direct observations of operator performance. Examples of PSFs included:

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

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

- 1 - "the human-machine interface ... minimized the probability of an operator error initiating an ISLOCA event ... because most of the pressure isolation valves (PIVs) were check valves which could not be manipulated by the operators. Additionally, for those PIVs which the operators could manipulate at power, there were interlocks and administrative controls to prevent inadvertent operation of these valves at power." [C-8]
- 2 - "The control room ... appeared to be well designed in that it had a good control display integration, effective panel layouts, and proper labeling." [C-8]
- 3 - There were ISLOCA-related training programs and additional training modules were being considered for development.

Examples of negative PSFs were:

- 1 - "Based on past events, the operators did not readily recognize the significance of interfacing systems configurations which could potentially initiate an ISLOCA event." [C-8]
- 2 - "In the control room, there was only one alarm (the residual heat removal High Discharge Pressure Annunciator) that would warn the operators of a potential ISLOCA condition and this alarm did not function during a recent [RHR] overpressurization event. Although the inspection did identify other instrumentation in the control room which could be used by operators to detect a potential ISLOCA event, their use, without the aid of annunciators or alarms could easily delay the operator detection of and response to an ISLOCA event." [C-8]
- 3 - "Observations concerning the relevance of procedures to human performance during detection and mitigation of an ISLOCA included: various procedures did not attribute high pressures in the interfacing systems, or high levels in tanks which receive flow from relief valves found in these interfacing systems to a potential ISLOCA event; ...The annunciator response procedure (ARP) for high PRT pressure, temperature, and level did not attribute the cause of the high or increasing PRT level to leaking

pressure isolation valves. Instead, the ARP for high PRT level identified the pressurizer PORV as the only probable source of flow into the PRT." [C-8]

- 4 - Operators' ISLOCA diagnostic abilities were centered around the presence of EMF 41 (auxiliary building vent high radiation alarm), which has the undesirable effect of increasing operator stress due to initiating preparation for a site evacuation, and the sealing and positive pressurization of the control room.

For the purposes of this HRA, stress and dependence were modeled as two of the more significant PSFs. From a human performance perspective, high levels of stress lead to higher probabilities of human error. Generally, a person's short-term, or working, memory (STM) can retain from five to nine items of information for brief periods. However, as stress increases, this capacity shrinks to levels where STM can only hold three to five items. This well documented finding interacts with a phenomenon called cognitive tunnel vision wherein high levels of stress cause an operator's visual and perceptual abilities to begin shrinking into a limited focus so that only one or two salient aspects of his environment are featured. Also, as stress continues to increase, the operator begins to retreat from current conditions, relying on previously learned (perhaps incorrect) patterns of behavior. In this HRA, stress levels were generally modeled as optimal with the exception of those actions occurring after an ISLOCA. In Sequences 1A, 1B, and 3, for human action FTD-LOCA (fail to detect LOCA), stress levels were modeled as moderately high due to required procedural responses during a reactor trip and/or safety injection. When EMF 41, the high radiation vent alarm, is activated (event DIAG-LOCA, diagnose ISLOCA), stress levels were increased to threat stress because the presence of EMF 41 requires the operators to begin initiating a site evacuation along with the sealing and positive pressurization of the control room.

In several of the ISLOCA scenarios, low (LD) to moderate (MD) levels of dependence were assigned between the control room supervisor (CRS) or shift supervisor (SS) and the licensed reactor operator (RO). As used in THERP, dependence refers to the level of interaction between two or more workers.

Dependence is usually modeled on a scale which ranges from complete dependence (where a second worker fails on a given task because of the failure of a primary worker on the same task) to complete independence (zero dependence or ZD).

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

The output from the preceding step (#6) is an extensive list of operator tasks and subtasks (with their associated PSFs) for each human action in the dominant PRA sequences. These detailed tasks are the required input for the seventh step, where appropriate HRA techniques for modeling the significant human actions were selected and applied. For each human action, the analysts selected an appropriate technique for task modeling and quantification. Because most of the human actions in this HRA entailed the use of various written procedures, THERP-type HRA event trees were used in modeling a majority of the human actions in the detailed analysis. However, not all ISLOCA scenarios were best represented by THERP event trees alone. In those cases, HRA fault trees were used in conjunction with the THERP event trees. The fault trees and THERP event trees were used in a detailed analysis to estimate the probability of human error for each of the dominant human actions. Quantification techniques included THERP and Human Cognitive Reliability (HCR) [C-9]. For each human failure, basic HEPs were calculated using THERP or HCR and modified using performance shaping factors (PSFs) to realistically describe the work processes at the utility.

Prior to the estimation of human error probabilities, the PRA and HRA analysts reviewed and evaluated the significant human actions, and their

Sequence ID _____ Task ID _____ Subtask ID _____

Crew size & composition _____

Who does task/subtask? _____

Crew experience Low_____ Optimal_____ Moderate_____ High_____

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

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

Median response time for whole task_____ Std Dev _____

Plant/system time available _____

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

and type of alarms competing for attention _____

Quality of plant interface Excellent___ Good___ Fair___ Poor___ Very Poor

Operators Stress Low___ Optimal___ Moderate___ High___

Type of instrument/control _____

HF notes on controls _____

Consequence of improper performance High___ Medium___ Low___

Explain _____

Feedback/system response to operator action _____

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

Proc Req'd Yes or No Proc covers case Yes or No

Proc well written Yes or No Proc understood Yes or No

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

Cognitive Behavior Skill_____ Rule_____ Knowledge_____

Tagging Yes or No Describe _____

Recovery Actions Checklists_____ Inspections_____ 2nd Person_____

Feedback from Annunciators_____ Alarms_____ Displays_____

Figure 1: ISLOCA Data Collection Form, page 1

Local or Remote operation? Explain _____

Type of clothing during action _____

Tasks or subtasks done step-by-step_____ or Dynamic_____

Dependence Is the order of the tasks critical Yes or No

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

Yes or No Explain _____

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

Is the job done with rest stops_____ or continuous performance_____?

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

If yes, what dosage?_____ mrem

HF comments of plant-specific PSF s _____

Additional Comments/observations _____

Figure 2: ISLOCA Data Form, page 2

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

Assigning HEP estimates to each of the subtasks was the major activity in Step #9 - Quantification. Traditionally, HRA analysts model human performance through the use of an event tree like Figure 3, which represents event ND-OPEN, - a failure on the part of the operators to close the residual heat removal system suction isolation valves during startup activities. Operator error was generally placed along the descending right branches of the event tree. Successful operator actions were sequenced on the left side of the tree. For example, on the top left, event "a" was the success path, i.e., RO goes to procedure OP/1/A/6200/04, step 2.26 of enclosure 4.1. Failure to accomplish this task was modeled on the branch on the right side as event "A" - RO fails to go to OP.... When a second operator is involved, such as in events "F" and "H", the actions were generally modeled in succession, as shown in Figure 3. For example, in task "E" the RO has failed to close valve ND-1B or ND-2A. The CRS, being in the same room, had an opportunity to detect the RO's error, then correct it (i.e., event "f"). If the CRS was successful in correcting the RO's error, this became a recovery action which returned the model to the success path via the dotted lines for event "f."

Basic HEPs for each individual failure were calculated using THERP or HCR. These basic HEPs were then modified using PSFs to realistically describe the work process. Each PSF either increased or reduced the likelihood of a given human error action. For the event tree in Figure 3, the following PSFs were used to modify the basic HEPs:

Figure 3: HRA Event Tree for ND-OPEN (Sequences 1A and 1B)

Sequence 1A & 1B -
ND-OPEN

- Even though the start-up procedure (OP/1/A/6100/01) is controlling the RO's actions, he/she may miss a significant step directing entry into a new procedure (OP/1/A/6200/04) which would eventually lead to the closure of the residual heat removal isolation valves.
- Stress levels were modeled as being optimal.
- The crew was judged to be experienced.
- Low Dependency (LD) was used to model the relationship between the RO and CRS, since the procedure calls for independent verification of PIV closure (ND-1B, ND-2A, ND-36B, and ND-37A).

Individual error paths were identified and failure probabilities were estimated using the HEPs and tables from THERP or estimates from HCR. (The probability values in the THERP tables are to be considered as median values from a lognormal distribution. The estimates from HCR are assumed to be point estimates). For example, path "A" in Figure 3 leads to failure by the RO to go to procedure OP/6200/04 (the first branch on the right side of the tree). This particular failure had a basic HEP of 0.003 (median) and an error factor of 3. This information came from THERP Table 20-7, item #2, and was not modified by a PSF. Upper (95%) and lower (5%) confidence bounds were obtained by multiplying or dividing the basic HEP of 0.003 by the error factor of 3 (upper bound = 0.009, lower bound = 0.001).

Each event tree contains several unique error paths. For example, event "A" and event "B" together constitute an error path wherein the RO fails to enter procedure OP/1/A/6200/04 on two occasions: step 2.26 of OP/1/A/6100/01 and later, step 2.34 of the same procedure. In a similar manner, failure path "A-b-C" models a sequence in which the RO fails to enter procedure OP/1/A/6100/01 from step 2.26, then recovers (e.g., correctly goes to OP/1/A/6200/04) from his first failure (event "b"), only to fail at selecting the proper enclosure (4.3 - event "C"). Probabilities for each unique error path were calculated by multiplying each HEP on a given error path by other HEPs on the same error path. For example, the error rate for path "A-B" would be calculated by multiplying the HEP for failure "A" (0.003) by the HEP for failure "B" (0.003) resulting in a nominal HEP for that path equaling 0.000009 ($0.003 \times 0.003 = 0.000009$). Other examples of error paths for this event tree

include: "A-b-c-D", "a-c-d-e-G-H", and "A-b-c-d-E-F". The failure probabilities for individual error paths were summed to give the total failure probability for that event tree.

Table 1-A lists the error paths and nominal HEPs for the event tree depicted in Figure 3 (i.e., ND-OPEN, Sequences 1A and 1B). This table enumerates the basic human actions/errors, the basic or unmodified HEP and its source from the table and item number in THERP, whether the action was modeled as being performed in a step-by-step mode or dynamically, PSF modifier values and the related THERP source, level of dependency, and finally, the nominal, or modified, HEP with its error factor in parentheses.

Table 1-B provides a listing of the individual failure paths for Figure 3, event ND-OPEN, and the resulting failure probabilities for each path, including how these probabilities were calculated. Failure probabilities of "0" in the tables signify error probabilities less than 10^{-6} ; such error probabilities were treated as being negligible. Table 1-B also lists a total failure probability for each event tree, which is simply the summation of the failure probabilities from the individual failure paths. As indicated in Table 1-B, the total failure probability for the ND-OPEN event tree in Figure 3 is estimated to be 0.0063. As a point estimate, given the PSFs discussed earlier, an RO, during start-up, can be expected to leave the ND PIVs open about 6 times out of 1000.

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

"Although we would like to have data clearly showing the distributions of human performance for various NPP (nuclear power plant) tasks, there

is ample evidence that the outcomes of HRAs are relatively insensitive to assumptions about such distributions...."

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

"The assumption of normal, lognormal, or other similar distributions will make no material difference in the results of HRA analyses for NPP operations. In some cases, this insensitivity may result from a well designed system that has so many recovery factors that the effect of any one human error on the system is not substantial.... For computational convenience, one might wish to assume the same distribution for probabilities of human failure as the one used for probabilities of equipment failure, as was used in WASH-1400."

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

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

This HRA adopted THERP'S recommendation to treat each HEP as a median value from a lognormal distribution. Detailed analyses were conducted for each of the significant scenarios identified in this ISLOCA PRA. Tables 1A and 1B summarize the results of these analyses using the nominal HEPs and Tables 1C and 1D provide the results for event ND-OPEN, after converting the median HEPs to mean HEPs using the following formulas:

Table 1-A: Median HEPs for Sequence 1A & 1B, ND-OPEN

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to go to OP/1/A/6200/04, to secure ND system	0.003	20-7 #2	SBS	1	20-16 #2a	ZD	0.003 (3)
B RO fails to go to OP/1/A/6200/04, to secure ND system (step 2.34) (recovery)	0.003	20-7 #2	SBS	1	20-16 #2a	ZD	0.003 (3)
C RO fails to select proper enclosure (4.3)	0.003	20-7 #2	SBS	1	20-16 #2a	ZD	0.003 (3)
D RO omits step 2.4 of OP/1/A/6200/04, enclosure 4.3 (omission)	0.003	20-7 #2	SBS	1	20-16 #2a	ZD	0.003 (3)
E RO fails to close 1ND-1B or 1ND-2A (commission)	0.001	20-12 #3	SBS	1	20-16 #2a	ZD	0.001 (3)
F CRS fails to verify RO closed 1ND-1B or 1ND-2A (recovery)	0.1	20-22 #1	SBS	1	20-16 #2a	LD	0.145 (5)
G RO fails to close 1ND-36B or 1ND-37A (commission)	0.001	20-12 #3	SBS	1	20-16 #2a	ZD	0.001 (3)
H CRS fails to verify RO closed 1ND-36B or 1ND-37A (recovery)	0.1	20-22 #1	SBS	1	20-16 #2a	LD	0.145 (5)

Table 1-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A & 1B, ND-OPEN (Median HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.003×0.003			
2 .AbC	0.003×0.003	0.000009		
3 .AbcD	0.003×0.003	0.000009		
4 .AbcdEF	$0.003 \times 0.001 \times 0.145$	0.000009		
5 .AbcdEfGH	$0.003 \times 0.001 \times 0.001 \times 0.145$	0.		
6 .AbcdeGH	$0.003 \times 0.001 \times 0.145$	0.		
7 .aC	0.003	0.003		
8 .acD	0.003	0.003		
9 .acdEF	0.001×0.145	0.000145		
10 .acdEfGH	$0.001 \times 0.001 \times 0.145$	0.		
11 .acdeGH	0.001×0.145	0.000145		
			0.006318	0.993681

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

where $\hat{x}$ = the Median HEP;

$$\mu = \ln \hat{x}; \text{ and,}$$

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

Converting median HEPs (from an assumed lognormal distribution) to mean HEPs allowed uncertainties in human error to be included in calculations of the uncertainty in core damage frequency. The actual conversions to mean HEPs were accomplished by inserting the nominal, median HEPs in each event tree (which had already been modified by PSFs) into the equations above. The resulting mean HEPs were then used in the appropriate error branch on specific event trees to calculate new error path and total failure probabilities for each event tree (see Tables 1C and 1D). Note that the modifications to the HEPs to account for PSFs were made in the calculations using the median values (seen in Tables 1A & 1B). Mean HEP values (in Tables 1C & 1D) were not modified a second time for PSFs such as stress and dependence.

This example focused on the event tree for event ND-OPEN; a similar process was followed for each of the remaining human actions analyzed in the HRA. Specific details, including event trees and HEP tables, are provided in the following section.

In the final two steps (#9 and #10) of the HRA process, the analysts reviewed the results of the HRA and documented all of the information needed to provide an audit trail. As final HRA failure probabilities were generated for each ISLOCA sequence, the HRA analysts consulted with the PRA analyst and a systems engineer regarding the validity, completeness, and relevance of the results. During these reviews, several questions arose which necessitated telephone calls to operations personnel at the plant. One such question

concerned the probability of an NLO (non-licensed operator) detecting an RHR pump high discharge pressure alarm in the auxiliary building. The utility answered the question by pointing out that the particular alarm was in a locked, high radiation area, which was only briefly entered once per 12-hour shift. This limited the likelihood of an NLO noticing this ISLOCA indication in a timely manner.

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

Table 1-C: Mean HEPs for Sequence 1A, ND-OPEN

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to go to OP/1/A/6200/04, to secure ND system	0.00375	Mean Convers.	SBS	1	20-16	ZD	0.00375 (3)
B RO fails to go to OP/1/A/6200/04, to secure ND system (step 2.34)	0.00375	Mean Convers.	SBS	1	20-16	ZD	0.00375 (3)
C RO fails to select proper enclosure (4.3)	0.00375	Mean Convers.	SBS	1	20-16	ZD	0.00375 (3)
D RO omits step 2.4 of OP/1/A/6200/04, enclosure 4.3 (omission)	0.00375	Mean Convers.	SBS	1	20-16	ZD	0.00375 (3)
E RO fails to close 1ND-1B or 1ND-2A (commission)	0.00125	Mean Convers.	SBS	1	20-16	ZD	0.00125 (3)
F CRS fails to verify RO closed 1ND-1B or 1ND-2A (recovery)	0.234	Mean Convers.	SBS	1	20-16	ZD	0.234 (5)
G RO fails to close 1ND-36B or 1ND-37A (commission)	0.00125	Mean Convers.	SBS	1	20-16	ZD	0.00125 (3)
H CRS fails to verify RO closed 1ND-36B or 1ND-37A (commission)	0.234	Mean Convers.	SBS	1	20-16	ZD	0.234 (5)

Table 1-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A, ND-OPEN (Mean HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.00375×0.00375	0.000014		
2 .AbC	0.00375×0.00375	0.000014		
3 .AbcD	0.00375×0.00375	0.000014		
4 .AbcdEF	$0.00375 \times 0.00125 \times 0.234$	0.000001		
5 .AbcdEFGH	$0.00375 \times 0.00125 \times 0.00125 \times 0.234$	0.		
6 .AbcdeGH	$0.00375 \times 0.00125 \times 0.234$	0.000001		
7 .aC	0.00375	0.00375		
8 .acD	0.00375	0.00375		
9 .acdEF	0.00125×0.234	0.000292		
10 .acdEFGH	$0.00125 \times 0.00125 \times 0.234$	0.		
11 .acdeGH	0.00125×0.234	0.000292		
			0.008129	0.99187

Scenarios and Human Actions for the ISLOCA HRA

This section describes the scenarios and summarizes the human actions analyzed in this ISLOCA HRA. Human actions for the sequences were initially identified in a cooperative effort by PRA and HRA analysts based on plant-specific information. The sequences were selected for analysis by using screening HEPs in the PRA modeling to determine likely scenarios in terms of ISLOCA risk (i.e., core damage frequencies greater than $10^{-8}/y$). A screening HEP of 0.5 was used, except in some cases where a screening HEP of 1.0 was judged appropriate due to a very limited time available for operator action. Following are brief descriptions of the selected scenarios from an HRA perspective and specific tables of the human actions relevant to the scenarios.

Scenarios 1A and 1B: RHR/RCS Interfacing LOCA During Startup

In these scenarios the reactor is shut down and the RHR system is in service to remove decay heat while the reactor coolant system (RCS) is being pressurized during startup. RHR system overpressurization occurs as a result of operator failure to close the RHR suction isolation valves to the RCS as reactor coolant pressure increases above 385 psig, the pressure prior to which the RHR system should be isolated and taken out of service.

In scenario 1A the RCS and pressurizer are filled with water (solid) and operators are using charging pumps to raise reactor coolant pressure slowly by collapsing air trapped in the steam generator U-tubes. The pressurizer power-operated relief valves (PORVs) are in the low temperature overpressure protection (LTOP) mode (i.e., two of the three PORVs have setpoints lowered to 400 psig). The RHR suction isolation valves should receive an autoclosure signal when reactor coolant wide range system pressure reaches 600 psig. The PORVs fail to open, the RHR suction isolation autoclosure function fails, and the RHR suction relief valves fail to open.

In scenario 1B there is a steam bubble in the pressurizer, reactor coolant temperature is assumed to be above 285° F, and operators are using

pressurizer heaters to raise RCS pressure and temperature. Scenario 1B proceeds just as scenario 1A except the PORVs are not in the LTOP mode and are thus not available to provide RHR overpressure protection.

The PRA event trees for both scenarios contain two pathways (denoted path 2 and path 3) of significance for the HRA. In path 2 the RHR discharge reliefs stick open and RHR pressure reaches approximately 600 psig, thus opening a drainage path to the recycle holdup tank (RHT) in the auxiliary building. Failure to detect, diagnose, and isolate this flow path could result in core damage. In path 3 the RHR discharge reliefs fail to open. Failure to detect RHR overpressurization could result in a break outside of containment and failure to detect, diagnose, and isolate this break could result in core damage.

Scenario 2: RHR/RCS Interfacing LOCA During Shutdown

In scenario 2 the plant is in the process of being shutdown. Operators initiate shutdown cooling with reactor coolant pressure and temperature still above allowable limits for the RHR system. This could only occur if the operators opened the RHR suction isolation valves, thereby exposing the RHR system to overpressurization. To open the valves, operators would have to override interlock permissives, disregard administrative barriers, and take actions beyond those specified by operating procedures. Further human actions for this scenario were not analyzed in the HRA, as premature opening of the RHR suction isolation (the scenario initiator) was estimated by the HRA to be an incredible event having a negligible probability.

Scenario 3: RHR/RCS Interfacing LOCA At Cold Leg Interface

Scenario 3 involves the failure of two PIV check valves (installed in series) in one of the four low pressure injection lines from the RHR system to the RCS cold legs. The reactor is assumed to be at power when a pair of check valves fails in one of the four injection lines. The RHR system overpressurizes and a break occurs outside of containment; this RHR pressurization occurs too quickly to allow operator detection, diagnosis, or isolation prior

to the break. Operator failure to detect, diagnosis, and isolate following the rupture will lead to core damage.

Human Actions for this ISLOCA HRA

The following table lists the PRA identifier and a brief description of each relevant human action identified for analysis in this HRA.

IDENTIFIER	DESCRIPTION
ND-OPEN	Operators fail to close RHR isolation MOVs during plant startup (scenarios 1A & 1B), or open RHR isolation MOVs too soon during plant shutdown (scenario 2)
OP-FTD-1	Operators fail to detect RHR discharge reliefs stuck open (scenarios 1A & 1B, path 2), or fail to detect RHR overpressurization (scenario 3)
OP-FTC-1	Operators fail to isolate RHR system by closing MOVs
FTD-LOCA	Operators fail to detect LOCA indications following a rupture
DIAG-LOCA	Operators fail to diagnose interfacing system LOCA outside of containment
OP-FTC-2	Operators fail to isolate ISLOCA by closing isolation MOVs

Human Error Descriptions for the ISLOCA HRA

Modeling Of Human Actions And Estimated Human Error Probabilities

This section presents the HRA event trees and the HEP estimates for the human actions identified as significant for this ISLOCA HRA. An HRA event tree (with any associated fault trees), subtask HEP tables documenting HEP estimation for each subtask branch on the tree, and tables providing failure path calculations and total failure probability estimates (for both median and mean HEPs) are presented for each human action. Each set of trees and tables for a human action is preceded by a brief discussion relevant to the modeling and HEP estimation for that action.

ND-OPEN: Scenario 1A and 1B

Event ND-OPEN for scenarios 1A and 1B is operator failure to close at least one RHR suction isolation MOV in each RHR train prior to reactor coolant pressure exceeding 385 psig during plant startup. The critical subtasks for ND-OPEN are modeled according to procedures that would be in effect. Applicable procedures are OP/1/A/6100/01, Controlling Procedure For Unit Startup, and OP/1/A/6200/04, Residual Heat Removal System. HRA modeling includes critical procedural steps and transitions, omission of the procedural step instructing the closure of the isolation MOVs, and commission errors for closing at least one MOV in each RHR train. Recovery paths are also modeled. The HRA event tree, subtask quantifications, and total failure probabilities are presented in Figure 3 and Tables 1A through 1D. As listed in Table 1-B, the total failure probability (using medians) of ND-OPEN, for this sequence is 0.006318.

ND-OPEN: Scenario 2

Event ND-OPEN for scenario 2 represents operator opening of RHR isolation MOVs too soon during plant shutdown. Due to operator training, appropriate operating procedures, well controlled interlocks, and the presence of administrative barriers, the probability of opening the RHR isolation MOVs during plant shutdown while reactor coolant pressure is still above the

allowable limit for the RHR system is assessed as negligible. An HRA tree and associated tables were therefore not developed for ND-OPEN in scenario 2.

OP-FTD-1: Scenario 1A (Path 2: ND discharge relief valves stuck open)

Event OP-FTD-1 for path 2, scenario 1A, represents operator failure to detect the RHR discharge relief valves stuck open after RHR overpressurization. HRA modeling includes relevant control room and auxiliary building indications available to operators. The subtask branch on the HRA event tree representing auxiliary building indications is further represented by a fault tree. High HEPs for NLO detection of auxiliary building indications reflect the fact that the indications appear on a panel not normally manned except for possible periodic (less than hourly) checks. No credit was given for a local RHR pump discharge pressure indicator as it is located in a locked high radiation area visited only once per shift. Dependence between specific operator actions was assessed and modeled. The HRA event tree (with associated fault tree), subtask quantifications, and total median failure probability (0.000063 for this scenario) are presented in Figure 4 and Tables 2-A through 2-D, respectively.

Figure 4: HRA Event Tree for OP-FTD-1, Path 2 - Relief Valves Stuck Open

Sequence 1A

OP-FTD-1 (Path 2: Relief Valves Stuck Open)

Table 2-A: Median HEPs for Sequence 1A OP-FTD-1 (Path 2: Relief Valves Open)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to detect decreasing PZR level	0.001	20-9 #3	SBS	1	20-16 #2a	ZD	0.001 (3)
B RO fails to detect PZR low level alarm (recovery)	0.0001	20-23 #1	SBS	1	20-16 #2a	LD	0.050095 (3)
C RO fails to detect ND pump high discharge pressure alarm	0.0001	20-23 #1	SBS	1	20-16 #2a	ZD	0.0001 (3)
D RO fails to detect ND isolation valve status lights(valves open) (recovery)	0.003	20-9 #4	SBS	1	20-16 #2a	MD	0.145428 (3)
E NLO fails to detect Aux. Bldg. indication (recovery)	0.903	fault tree	SBS	1	20-16 #2a	ZD	0.903 (5)

Table 2-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A OP-FTD-1 (Path 2: Relief Valves Open - Median HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.001×0.050095	0.00005		
2 .AbCDE	$0.001 \times 0.0001 \times 0.145428 \times 0.903$	0.		
3 .aCDE	$0.0001 \times 0.145428 \times 0.903$	0.000013		
			0.000063	0.999936

Table 2-C: Mean HEPs for Sequence 1A OP-FTD-1 (Path 2: Relief Valves Open)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to detect decreasing PZR level	0.00125	Mean Convers	SBS	1	20-16 #2	ZD	0.00125 (3)
B RO fails to detect PZR low level alarm (recovery)	0.0626	Mean Convers.	SBS	1	20-16 #2	ZD	0.0626 (3)
C RO fails to detect ND pump high discharge pressure alarm	0.000125	Mean Convers.	SBS	1	20-16 #2	ZD	0.000125 (3)
D RO fails to detect ND isolation valve status lights(valves open)(recovery)	0.182	Mean Convers.	SBS	1	20-16 #2	ZD	0.182 (3)
E NLO fails to detect Aux. Bldg. indication (recovery)	0.903	Mean Convers.	SBS	1	20-16 #2	ZD	0.903 (1)

Table 2-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A OP-FTD-1 (Path 2: Relief Valves Open - Mean HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.00125×0.0626			
2 .AbCDE	$0.00125 \times 0.000125 \times 0.182 \times 0.903$	0.000078		
3 .aCDE	$0.000125 \times 0.182 \times 0.903$	0. 0.00002		
			0.000098	0.999901

OP-FTD-1: Scenario 1B (Path 2: ND discharge relief valves stuck open)

Event OP-FTD-1 for path 2 scenario 1B is modeled as it was for scenario 1A except scenario 1B provides an additional control room indication related to the pressurizer. This is the case because in scenario 1B there is a bubble in the pressurizer, compared to Sequence 1A where the RCS is "solid."

Figure 5 shows the event tree for this scenario and is followed by Tables 3-A through 3-D, which document calculations for individual HEPs, failure path HEPs, and total failure probabilities (both mean and median). As seen in Table 3-B, the estimated median total failure probability for this scenario is 0.000002.

Figure 5: HRA Event Tree for OP-FTD-1, Sequence 1B, Path 2-Relief Valves Open

Sequence 1B

OP-FTD-1 (Path 2 Relief Valves Stuck Open)

Table 3-A: Median HEPs for Sequence 1B OP-FTD-1 (Path 2: Relief Valves Open)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to detect PZR pressure decreasing	0.001	20-9 #3	SBS	1	20-16 #2a	ZD	0.001 (3)
B RO fails to detect PZR level decreasing (recovery)	0.001	20-9 #3	SBS	1	20-16 #2a	LD	0.05095 (3)
C RO fails to detect PZR low level alarm (recovery)	0.0001	20-23 #1	SBS	1	20-16 #2a	LD	0.050095 (3)
D RO fails to detect ND pump discharge high pressure alarm	0.0001	20-23 #1	SBS	1	20-16 #2a	ZD	0.0001 (3)
E RO fails to detect ND isolation valve status lights (valves open)(recovery)	0.003	20-9 #4	SBS	1	20-16 #2a	ZD	0.003 (3)
F NLO fails to detect Aux Bldg indications (recovery)	0.903	fault tree	SBS	1	20-16 #2a	ZD	0.903 (5)

Table 3-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1B OP-FTD-1 (Path 2: Relief Valves Open - Median HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .ABC	$0.001 \times 0.05095 \times 0.050095$	0.000002		
2 .ABcDEF	$0.001 \times 0.05095 \times 0.0001 \times 0.003 \times 0.903$	0.		
3 .AbDEF	$0.001 \times 0.0001 \times 0.003 \times 0.903$	0.		
4 .aDEF	$0.0001 \times 0.003 \times 0.903$	0.		
			0.000002	0.999997

Table 3-C: Mean HEPs for Sequence 1B OP-FTD-1 (Path 2: Relief Valves Open)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to detect PZR pressure decreasing	0.00125	Means Convers.	SBS	1	20-16 #2	ZD	0.00125 (3)
B RO fails to detect PZR level decreasing (recovery)	0.0626	Mean Convers.	SBS	1	20-16 #2	ZD	0.0626 (3)
C RO fails to detect PZR low level alarm (recovery)	0.0626	Mean Convers.	SBS	1	20-16 #2a	ZD	0.0626 (3)
D RO fails to detect ND pump discharge high pressure alarm	0.000125	Mean Convers.	SBS	1	20-16 #2a	ZD	0.000125 (3)
E RO fails to detect ND isolation valve status lights (valves open)(recovery)	0.00375	Mean Convers.	SBS	1	20-16 #2a	ZD	0.00375 (3)
F NLO fails to detect Aux Bldg indications (recovery)	0.903	Mean Convers.	SBS	1	20-16 #2a	ZD	0.903 (1)

Table 3-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1B OP-FTD-1 (Path 2: Relief Valves Open - Mean HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .ABC	$0.00125 \times 0.0626 \times 0.0626$	0.000004		
2 .ABcDEF	$0.00125 \times 0.0626 \times 0.000125 \times 0.00375 \times 0.903$	0.		
3 .AbDEF	$0.00125 \times 0.000125 \times 0.00375 \times 0.903$	0.		
4 .aDEF	$0.000125 \times 0.00375 \times 0.903$	0.		
			0.000005	0.999994

OP-FTD-1: Scenario 1A and 1B (Path 3: ND discharge reliefs fail to open)

Event OP-FTD-1 for path 3 of scenarios 1A and 1B represents operator failure to detect RHR overpressurization. An RHR pump discharge high pressure alarm and the control board status lights for the RHR isolation valves are the only relevant control room indications. The status lights are modeled as a recovery indication for missing the high pressure alarm. Moderate dependence between the actions was assessed, as failure to detect the alarm suggests influences which would similarly affect detection of the status light indication. The HRA event tree, subtask quantifications, and total failure probabilities are presented in Figure 6 and Tables 4-A through 4-D, respectively. The total median failure probability for this sequence is 0.000014, and is found in Table 4-B.

Figure 6: HRA Event Tree for FTD-LOCA, Sequence 1A & 1B, Path 3-Reliefs Not Open

Sequenc 1A & 1B

OP-FTD-1 (Path 3 Relief Valves Not Open)

Table 4-A: Median HEPs for Sequence 1A & 1B OP-FTD-1 (Path 3: Relief Valves Not Open)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to detect ND pump high discharge pressure alarm	0.0001	20-23 #1	SBS	1	20-16 #2a	ZD	0.0001 (3)
B RO fails to detect ND isolation valve status lights(valves open)(recovery)	0.003	20-9 #4	SBS	1	20-16 #2a	MD	0.145428 (3)

Table 4-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A & 1B OP-FTD-1 (Path 3: Relief Valves Not Open - Median HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.0001×0.145428	0.000014	0.000014	0.999985

Table 4-C: Mean HEPs for Sequence 1A & 1B OP-FTD-1 (Path 3: Relief Valves Not Open)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to detect ND pump high discharge pressure alarm	0.000125	Mean Convers.	SBS	1	20-16 #2	ZD	0.0001 (3)
B RO fails to detect ND isolation valve status lights(valves open)(recovery)	0.182	Mean Convers.	SBS	1	20-16 #2	ZD	0.145428 (3)

Table 4-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A&1B OP-FTD-1(Path 3: Reliefs Not Open - Mean HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.000125×0.182	0.000022	0.000022	0.999977

OP-FTC-1: Scenario 1A and 1B (Path 2: ND discharge reliefs stuck open)

Event OP-FTC-1 represents operator failure to isolate the RHR system by closing at least one RHR suction isolation MOV in each RHR train, following detection of stuck-open RHR discharge relief valves. HRA modeling includes diagnosis of the need to isolate the RHR system, and omission and commission errors relevant to closing the isolation MOVs. Recovery credit is given to the CRS with a moderate level of dependence assessed between the RO and the CRS. Omission and commission branches for closing the MOVs required for RHR isolation are further modeled and quantified with a fault tree (Figure 8). The HCR technique is used to quantify the diagnosis subtasks with the assumptions of a knowledge-based action in a 3.5-hour time frame. The HRA event tree, subtask quantifications, and total failure probabilities are presented in Figure 7 and Tables 5-A through 5-D respectively. Table 5-B also lists the total median failure probability for this event tree as 0.000929. Note that HEP values for events A and B are negligible, i.e., less than 10^{-6} , and are only listed numerically because of the mathematical approach used in THERP to quantify event trees and failure paths. The same situation applies to event E, which has an HEP of 0.000002. It is considered to be negligible since the hardware failure rate is estimated to be between 10^{-4} and 10^{-3} .

Figure 7: HRA Event Tree for OP-FTC-1, Sequence 1A & 1B, Path 2-Reliefs Lift

Sequence 1A & 1B

OP-FTC-1 (Path 2, reliefs lift)

Figure 8: HRA Fault Tree for OP-FTC-1, Sequence 1A & 1B, Path 2-Reliefs Lift

Fault Tree #1

Sequence 1A & 1B

Paths 2 & 3

Table 5-A: Median HEPs for Sequence 1A & 1B, OP-FTC-1 (Path 2, Reliefs Lift)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to diagnose need to isolate ND (3.5 hrs time)	0.000001	hcr knowledge	SBS	1	20-16 #2a	ZD	0.000001 (3)
B CRS fails to diagnose need to isolate ND (recovery)	0.000001	HCR knowledge	SBS	1	20-16 #2a	MD	0.142858 (3)
C RO fails to isolate ND (Omission)	0.005	fault tree	SBS	1	20-16 #2a	ZD	0.005 (5)
D CRS fails to verify RO isolates ND (recovery)	0.05	20-22 #3	SBS	1	20-16 #2a	MD	0.185714 (5)
E RO fails to isolate ND (Commission)	0.000002	fault tree	SBS	1	20-16 #2a	ZD	0.000002 (3)
F CRS fails to isolate ND (recovery)	0.05	20-22 #3	SBS	1	20-16 #2a	MD	0.185714 (5)

Table 5-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 1A & 1B, OP-FTC-1 (Path 2, Reliefs Lift - Median HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.000001×0.142858	0.		
2 .AbCD	$0.000001 \times 0.005 \times 0.185714$	0.		
3 .AbCdEF	$0.000001 \times 0.005 \times 0.000002 \times 0.185714$	0.		
4 .AbcEF	$0.000001 \times 0.000002 \times 0.185714$	0.		
5 .aCD	0.005×0.185714	0.000928		
6 .aCdEF	$0.005 \times 0.000002 \times 0.185714$	0.		
7 .acEF	0.000002×0.185714	0.		
			0.000929	0.99907

Table 5-C: HEPs for Sequence 1A & 1B, OP-FTC-1 (Path 2, Reliefs Lift)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to diagnose need to isolate ND (3.5 hrs time)	0.000001	Mean Convers.	SBS	1	20-16 #2a	ZD	0.000001 (1)
B CRS fails to diagnose need to isolate ND (recovery)	0.179	Mean Convers.	SBS	1	20-16 #2a	ZD	0.179 (1)
C RO fails to isolate ND (Omission)	0.00807	Mean Convers.	SBS	1	20-16 #2a	ZD	0.00807 (5)
D CRS fails to verify RO isolates ND (recovery)	0.3	20-22 #3	SBS	1	20-16 #2a	ZD	0.3 (5)
E RO fails to isolate ND (Commission)	0.000002	Mean Convers.	SBS	1	20-16 #2a	ZD	0.000002 (3)
F CRS fails to isolate ND (recovery)	0.3	20-22 #3	SBS	1	20-16 #5	ZD	0.3 (5)

Table 5-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A & 1B, OP-FTC-1 (Path 2, Reliefs Lift - Mean HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.000001×0.179	0.		
2 .AbCD	$0.000001 \times 0.00807 \times 0.3$	0.		
3 .AbCdEF	$0.000001 \times 0.00807 \times 0.000002 \times 0.3$	0.		
4 .AbcEF	$0.000001 \times 0.000002 \times 0.3$	0.		
5 .aCD	0.00807×0.3	0.002421		
6 .aCdEF	$0.00807 \times 0.000002 \times 0.3$	0.		
7 .acEF	0.000002×0.3	0.		
			0.002421	0.997578

OP-FTC-1: Scenario 1A and 1B (Path 3: ND discharge reliefs fail to open)

Event OP-FTC-1 represents operator failure to isolate the RHR system by closing at least one RHR suction isolation MOV in each RHR train, following detection of RHR overpressurization. HRA modeling includes diagnosis of the need to isolate RHR, and omission and commission errors relevant to closing the isolation MOVs. Recovery credit is given to the CRS, with a moderate level of dependence assessed between the RO and the CRS. Omission and commission branches for closing the MOVs required for RHR isolation are further modeled and quantified with a fault tree (see Figure 10). The HCR technique is used to quantify the diagnosis subtasks with the assumption of knowledge-based action in a 0.5-hour time frame. Unlike path 2, the total failure probability for OP-FTC-1 in path 3 is dominated by the high diagnosis HEP. This high diagnosis HEP results from the limited time available to operators before RHR system rupture, given that the RHR discharge relief valves fail to open. The HRA event tree, subtask quantifications and total failure probabilities (0.019926 - median) are presented in Figure 9 and Tables 6-A through 6-D, respectively. Note that the HEP for event E is 0.000002. It is considered negligible since the hardware failure rate for these valves is once again between 10^{-4} and 10^{-3} .

Figure 9: HRA Event Tree for OP-FTC-1, Sequence 1A & 1B, Path 3-Reliefs Do Not Lift

Sequence 1A & 1B

OP-FTC-1 (Path 3, reliefs do not lift)

Figure 10: HRA Fault Tree for OP-FTC-1, Sequence 1A & 1B, Path 3-Reliefs Do Not Lift

Table 6-A: Median HEPs for Sequence 1A & 1B, OP-FTC-1 path 3(reliefs do not lift)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to diagnose need to isolate ND	0.087	HCR knowledge	SBS	1	20-16 #2a	ZD	0.087 (10)
B CRS fails to diagnose need to isolate ND (recovery)	0.087	HCR knowledge	SBS	1	20-16 #2a	MD	0.217428 (10)
C RO fails to isolate ND (omission)	0.005	20-7 #5	SBS	1	20-16 #2a	ZD	0.005 (5)
D CRS fails to verify RO isolates ND (recovery)	0.05	20-22#3	SBS	1	20-16 #2a	MD	0.185714 (5)
E RO fails to isolate ND (commission)	0.000002	20-12 #3	SBS	1	20-16 #2a	ZD	0.000002 (3)
F CRS fails to isolate ND (recovery)	0.05	20-22 #3	SBS	1	20-16 #2a	MD	0.185714 (5)

Table 6-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 1A & 1B, OP-FTC-1 path 3 (reliefs do not lift - Median HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.087×0.217428	0.018916		
2 .AbCD	$0.087 \times 0.005 \times 0.185714$	0.00008		
3 .AbCdEF	$0.087 \times 0.005 \times 0.000002 \times 0.185714$	0.		
4 .AbcEF	$0.087 \times 0.000002 \times 0.185714$	0.		
5 .aCD	0.005×0.185714	0.000928		
6 .aCdEF	$0.005 \times 0.000002 \times 0.185714$	0.		
7 .acEF	0.000002×0.185714	0.		
			0.019926	0.980073

Table 6-C: HEPs for Sequence 1A, OP-FTC-1 path 3(reliefs do not lift)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to diagnose need to isolate ND	0.087	Mean Convers.	SBS	1	20-16 #5	ZD	0.087 (1)
B CRS fails to diagnose need to isolate ND (recovery)	0.217	Mean Convers.	SBS	1	20-16#5	ZD	0.217 (1)
C RO fails to isolate ND (omission)	0.00807	20-7 #5	SBS	1	20-16 #5	ZD	0.00807 (5)
D CRS fails to verify RO isolates ND (recovery)	0.3	20-22#3	SBS	1	20-16 #5	ZD	0.3 (5)
E RO fails to isolate ND (commission)	0.000002	20-12 #3	SBS	1	20-16 #5	ZD	0.000002 (3)
F CRS fails to isolate ND (recovery)	0.3	20-22 #3	SBS	1	20-16 #5	ZD	0.3 (5)

Table 6-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 1A, OP-FTC-1 path 3(reliefs do not lift - Mean HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.087×0.217	0.018879		
2 .AbCD	$0.087 \times 0.00807 \times 0.3$	0.00021		
3 .AbCdEF	$0.087 \times 0.00807 \times 0.000002 \times 0.3$	0.		
4 .AbcEF	$0.087 \times 0.000002 \times 0.3$	0.		
5 .aCD	0.00807×0.3	0.002421		
6 .aCdEF	$0.00807 \times 0.000002 \times 0.3$	0.		
7 .acEF	0.000002×0.3	0.		
			0.021511	0.978488

FTD-LOCA: Scenario 1A and 1B (Path 3: ND discharge reliefs fail to open)

Event FTD-LOCA represents operator failure to detect LOCA indications following a rupture in the RHR system. Two paths in the PRA event tree are applicable: Path 3A, where operators have detected RHR overpressurization but have failed to isolate the RHR system prior to rupture, and path 3B, where operators have not detected RHR overpressurization prior to rupture. The same modeling of FTD-LOCA applies for both paths as the relevant indications are identical following the rupture. The critical subtasks for FTD-LOCA are modeled according to available control room indications and procedures that should be in effect. The modeling of this task reflects a strong reliance on the applicable procedures since the detailed task analyses and operator interviews during the plant inspection repeatedly indicated that the operators would follow procedural guidance closely in this situation. The applicable procedures are AP/1/A/5500/10, Reactor Coolant Leak, and EP/2/A/5000/01, Reactor Trip Or Safety Injection. HRA modeling includes detection of relevant control room indications, entering AP/1/A/5500/10 based on these symptoms, and transitioning to EP/2/A/5000/01 from subsequent actions step 1b3 (Response Not Obtained) of the AP based on detection of continued decrease in pressurizer level. The PSF for stress is modeled as increasing from optimal to moderately high upon entry into the abnormal procedure (i.e., the point at which an off-normal plant condition is acknowledged by the crew). The HRA tree, subtask quantifications, and total failure probabilities (estimated as 0.025 - median) are presented in Figure 11 and Tables 7-A through 7-D, respectively. The event tree models the necessary symptoms for entry into AP/1/A/5500/10 but it does not include any recovery credit because the CRS, STA, and other operators would be involved with other significant tasks.

Figure 11: HRA Event Tree for FTD-LOCA, Sequence 1A & 1B, Paths 3A & 3B

Sequence 1A & 1B

FTD-LOCA (Paths 3A & 3B)

Table 7-A: Median HEPs for Sequence 1A & 1B, FTD-LOCA (paths 3A and 3B)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to detect charging level in- creasing	0.001	20-9 #3	SBS	1	20-16 #2a	ZD	0.001 (3)
B RO fails to detect PZR level low or decreasing initially	0.001	20-9 #3	SBS	1	20-16 #2a	ZD	0.001 (3)
C RO fails to detect PZR pressure decreasing initially	0.001	20-9 #3	SBS	1	20-16 #2a	ZD	0.001 (3)
D RO fails to enter AP/1/A/5500/10	0.005	20-6 #4	SBS	2	20-16 #4a	ZD	0.01 (10)
E RO fails to detect continued decrease in PZR level	0.001	20-9 #3	SBS	2	20-16 #4a	ZD	0.002 (3)
F RO fails to enter EP/2/A/5000/01	0.005	20-6 #4	SBS	2	20-16 #4a	ZD	0.01 (10)

Table 7-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 1A & 1B, FTD-LOCA (paths 3A and 3B - Median HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0 001			
2 .aB	0 001	0.001		
3 .abC	0 001	0.001		
4 .abcD	0 01	0.001		
5 .abcdE	0 002	0.01		
6 .abcdeF	0 01	0.002		
		0.01		
			0.025	0.975

Table 7-C: Mean HEPs for Sequence 1A & 1B, FTD-LOCA (paths 3A and 3B)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to detect charging level increasing	0.00125	Mean Convers.	SBS	1	20-16 #2	ZD	0.00125 (3)
B RO fails to detect PZR level low or decreasing initially	0.00125	Mean Convers.	SBS	1	20-16 #5	ZD	0.00125 (3)
C RO fails to detect PZR pressure decreasing initially	0.00125	Mean Convers.	SBS	1	20-16 #2	ZD	0.00125 (3)
D RO fails to enter AP/1/A/5500/10	0.0266	Mean Convers.	SBS	1	20-16 #4	ZD	0.0266 (10)
E RO fails to detect continued decrease in PZR level	0.0025	Mean Convers.	SBS	1	20-16 #2	ZD	0.0025 (3)
F RO fails to enter EP/2/A/5000/01	0.0266	Mean Convers.	SBS	1	20-16 #4	ZD	0.0266 (10)

Table 7-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A & 1B, FTD-LOCA (paths 3A and 3B - Mean HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.00125			
2 .aB	0.00125	0.00125		
3 .abC	0.00125	0.00125		
4 .abcD	0.0266	0.00125		
5 .abcdE	0.0025	0.0266		
6 .abcdef	0.0266	0.0025		
		0.0266		
			0.05945	0.94055

FTD-LOCA: Scenario 3

Event FTD-LOCA represents operator failure to detect LOCA indications following a rupture in the RHR system. Recall that in scenario 3 the reactor is operating at power, and check valve failures at the RCS/RHR pressure isolation boundary expose the RHR system to high RCS pressures. The critical subtasks are modeled according to applicable procedures. These procedures are EP/1/A/5000/1A, Reactor Trip Response, and EP/2/A/5000/10, Reactor Trip Or Safety Injection. HRA modeling includes entering EP/1/A/5000/1A upon reactor trip, referencing the proper enclosure and procedural step, and properly transitioning to EP/2/A/5000/10. Stress is modeled as moderately high as the operators experience a reactor trip and enter the emergency procedures. The HRA tree, subtask quantifications, and total failure probabilities (estimated as 0.026 - median) are presented in Figure 12 and Tables 8-A through 8-D, respectively.

Figure 12: HRA Event Tree for FTD-LOCA, Sequence 3

Sequence 3
FTD-LOCA

Figure 12: HRA Event Tree for FTD-LOCA, Sequence 3

Sequence 3

FTD-LOCA

Table 8-A: Median HEPs for Sequence 3, FTD-LOCA

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A CRS fails to enter EP/1/A/5000/1A	0.005	20-6 #4	SBS	2	20-16 #4a	ZD	0.01 (10)
B CRS fails to read step A of Enclosure 1 for EP/1/A/5000/1A	0.003	20-7 #2	SBS	2	20-16 #4a	ZD	0.006 (3)
C CRS fails to enter EP/2/A/5000/01 as directed in Encl. 1 EP/1/A/5000/1A	0.005	20-6 #4	SBS	2	20-16 #4a	ZD	0.01 (10)

Table 8-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 3, FTD-LOCA (Median HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.01	0.01		
2 .aB	0.006	0.006		
3 .abC	0.01	0.01		
			0.026	0.974

Table 8-C: Mean HEPs for Sequence 3, FTD-LOCA

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A CRS fails to enter EP/1/A/5000/1A	0.0266	Mean Convers.	SBS	1	20-16 #4	ZD	0.0266 (10)
B CRS fails to read step A of Enclosure 1 for EP/1/A/5000/1A	0.0075	Mean Convers.	SBS	1	20-16 #4	ZD	0.0075 (3)
C CRS fails to enter EP/2/A/5000/01 as directed in Encl. 1 EP/1/A/5000/1A	0.0266	Mean Convers.	SBS	1	20-16 #4	ZD	0.0266 (10)

Table 8-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 3, FTD-LOCA (Mean HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.0266	0.0266		
2 .aB	0.0075	0.0075		
3 .abC	0.0266	0.0266		
			<hr/> 0.0607	<hr/> 0.9393

DIAG-LOCA: Scenario 1A, 1B and 3

Event DIAG-LOCA represents operator failure to diagnose an ISLOCA. The critical subtasks are modeled according to applicable procedures and available control room indications. HRA modeling includes detection of EMF 41 (an auxiliary building vent high radiation alarm) and entry into procedure EP/1/A/5000/1C6 (LOCA Outside Containment). Operators are directed to enter EP/1/A/5000/1C6 in an alarm response flow chart, or via a transition from step 28 (Response Not Obtained) of procedure EP/2/A/5000/01, Reactor Trip Or Safety Injection, based on the receipt of an alarm from EMF 41. Alarm response procedures for EMF 41 require preparations for a site evacuation, and locking down and initiating positive ventilation for the control room. Therefore, stress is modeled as increasing from a moderately high level to a threat level upon detection of alarm EMF 41. The HRA tree, subtask quantifications, and total failure probabilities (estimated as a median HEP of 0.025) are presented in Figure 13 and Tables 9-A through 9-D, respectively.

Figure 13: HRA Event Tree for DIAG-LOCA

Sequence 1A, 1B, & 3
DIAG-LOCA (Path 3A & 3B)

Table 9-A: Median HEPs for Sequence 1A, 1B, & 3, DIAG-LOCA (paths 3A and 3B)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	THERP T20-17 Dependency	Nominal HEP (Error Factor)
A RO fails to detect EMF 41 radiation stack alarm	0.0001	20-23 #1	SBS	2	20-16 #4a	ZD	0.0002 (10)
B RO fails to enter EP/1/A/5000/1C6 as indicated by presence of EMF41	0.005	20-6 #4	SBS	5	20-16 #6a	ZD	0.025 (10)

Table 9-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 1A, 1B, & 3, DIAG-LOCA (paths 3A and 3B - Median HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.0002			
2 .aB	0.025	0.0002 0.025		
			0.0252	0.9748

Table 9-C: Mean HEPs for Sequence 1A, 1B, & 3, DIAG-LOCA (paths 3A and 3B)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A RO fails to detect EMF 41 radiation stack alarm	0.000533	Mean Convers.	SBS	1	20-16 #4	ZD	0.000533 (10)
B RO fails to enter EP/1/A/5000/1C6 as indicated by presence of EMF41	0.0666	Mean Convers.	SBS	1	20-16 #6	ZD	0.0666 (10)

Table 9-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Sequence 1A, 1B, & 3, DIAG-LOCA (paths 3A and 3B - Mean HEPs)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.000533			
2 .aB	0.0666	0.000533 0.0666		
			0.067133	0.932867

OP-FTC-2 Scenario 1A, 1B, and 3

OP-FTC-2 represents operator failure to isolate the ISLOCA by closing at least one ND (residual heat removal system) suction isolation MOV in each ND system train (scenario 1A and 1B), or closing isolation MOVs NI173A and NI178B (scenario 3; NI= safety injection system) following diagnosis of ISLOCA. Subtasks are modeled as per step 1 (scenarios 1A and 1B), and step 3 (scenario 3), of EP/1/A/5000/1C6, LOCA Outside Of Containment. The modeling assumes that the CRS is functioning as procedure reader and the RO is performing the actions as is specified when an emergency procedure is in effect. Stress is assessed as threat stress for all subtasks since OP-FTC-2 immediately follows the detection of alarm EMF-41 with the subsequent initiations of the emergency response plan as described under DIAG-LOCA above. HRA modeling includes omission errors regarding CRS direction to the RO (i.e., omission of procedural step), commission errors by the RO, and credits recovery potential for CRS verification of RO actions. Moderate dependence is assessed between the CRS and the RO. No recovery credit is given for an RO reminding the CRS in event A, since it was assumed that all available control room personnel would actively involved in tasks directed by the CRS (the procedure reader). The HRA tree, subtask quantifications, and total failure probability (estimated to be a median value of 0.0179) are presented in Figure 14 and Tables 10-A through 10-D, respectively.

OP-FTC-2: Scenario 1A, 1B, and 3

Event OP-FTC-2 represents operator failure to isolate the ISLOCA by closing at least one RHR suction isolation MOV in each RHR train (scenario 1A and 1B), or closing safety injection isolation MOVs NI-173A and NI-178B (scenario 3) following diagnosis of ISLOCA. Subtasks are modeled as per step 1 (scenarios 1A and 1B), and step 3 (scenario 3), of procedure EP/1/A/5000/1C6, LOCA Outside Of Containment. The modeling assumes that the CRS is functioning as procedure reader and the RO is performing the actions as is specified when an emergency procedure is in effect. Threat stress is assessed for all subtasks since event OP-FTC-2 immediately follows the detection of alarm EMF-41 with the subsequent initiations of the emergency response plan as described under event DIAG-LOCA above. HRA modeling includes omission errors regarding CRS direction to the RO (i.e., omission of procedural step), commission errors by the RO, and credits recovery potential for CRS verification of RO actions. Moderate dependence is assessed between the CRS and the RO. No recovery credit is given for an RO reminding the CRS in event A, since it was assumed that all available control room personnel would be actively involved in tasks directed by the CRS (the procedure reader). The HRA tree, subtask quantifications, and total failure probability (estimated to be a median value of 0.0179) are presented in Figure 14 and Tables 10-A through 10-D, respectively.

Figure 14: HRA Event Tree for OP-FTC-2

Sequence 1A, 1B, & 3

OP-FTC-2 (Path 3A & 3B)

** NOTE Valves ND2A, ND1B, ND36B, & ND37A refer to Sequences 1A & 1B
Valves NI173A & NI178B refer to Sequence 3

Table 10-A: Median HEPs for Sequence 1A, 1B, & 3, OP-FTC-2 (Path 3A & 3B)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A CRS fails to direct RO to close ND1B, ND2A or NI173A (Omission)	0.003	20-7 #2	SBS	5	20-16 #6	ZD	0.015 (3)
B RO fails to close ND1B or ND2A (or NI173A - Commission)	0.001	20-12 #3	SBS	5	20-16 #6	ZD	0.005 (3)
C CRS fails to verify RO closes ND1B or ND2A (or NI173A - recovery)	0.1	20-22 #1	SBS	5	20-16 #6	MD	0.571428 (5)
D CRS fails direct RO to close ND36B or ND37A (or NI178B - Omission)	0.003	20-7 #2	SBS	5	20-16 #6	ZD	0.015 (3)
E RO fails to close ND36B or ND37A (or NI178B - Commission)	0.001	20-12 #3	SBS	5	20-16 #6	ZD	0.005 (3)
F CRS fails to verify RO closes ND36B or ND37A (or NI178B - recovery)	0.1	20-22 #1	SBS	5	20-16 #6	MD	0.571428 (5)

Table 10-B: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A, 1B, & 3, OP-FTC-2 (Path 3A & 3B - Median HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 A	0.015	0.015		
2 aBC	0.005×0.571428	0.002857		
3 aBcD	0.005×0.015	0.000075		
4 aBcdEF	$0.005 \times 0.005 \times 0.571428$	0.000014		
5 abD	0.015	0.015		
6 abdEF	0.005×0.571428	0.002857		
			0.035803	0.964196

Table 10-C: Mean HEPs for Sequence 1A, 1B, & 3, OP-FTC-2 (Path 3A & 3B)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A CRS fails to direct RO to close ND1B or ND2A (or NI173A - Omission)	0.0187	Mean Convers.	SBS	1	20-16 #6	ZD	0.0187 (3)
B RO fails to close ND1B or ND2A (or NI173A - Commission)	0.00625	Mean Convers.	SBS	1	20-16 #6	ZD	0.00625 (3)
C CRS fails to verify RO closes ND1B or ND2A (or NI173A - recovery)	0.0922	Mean Convers.	SBS	1	20-16 #6	ZD	0.0922 (5)
D CRS fails direct RO to close ND36B or ND37A (or NI178B - Omission)	0.0187	Mean Convers.	SBS	1	20-16 #6	ZD	0.0187 (3)
E RO fails to close ND36B or ND37A (or NI178B - Commission)	0.00625	Mean Convers.	SBS	1	20-16 #6	ZD	0.00625 (3)
F CRS fails to verify RO closes ND36B or ND37A (or NI178B - recovery)	0.0922	Mean Convers.	SBS	1	20-16 #6	ZD	0.0922 (5)

Table 10-D: FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES**Sequence 1A, 1B, & 3, OP-FTC-2 (Path 3A & 3B - Mean HEPs)**

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.0187			
2 .aBC	0.00625×0.0922	0.0187		
3 .aBcD	0.00625×0.0187	0.000576		
4 .aBcdEF	$0.00625 \times 0.00625 \times 0.0922$	0.000116		
5 .abD	0.0187	0.000003		
6 .abdEF	0.00625×0.0922	0.0187		
		0.000576		
			0.038672	0.961327

Summary of HRA Results

Detailed HRA analyses were conducted for each of the significant scenarios identified in this ISLOCA PRA. Extensive documentation for each analysis has been provided above (e.g., descriptions of the scenarios, human errors, and both median and mean HEPs). Tables 11 through 14 summarize the HRA results.

Table 11: Summary of HRA Results for Scenario 1A

Scenario 1A: Overpressurization of ND System During Startup - Pressurizer Solid

IDENTIFIER	HUMAN ERROR	NOMINAL HEP	MEAN HEP
ND-OPEN	Operator fails to close ND isolation valves ND1B & ND2A/ND36B & ND37A	0.0063	0.0082
OP-FTD-1	Op. fails to detect problem: PATH 2=reliefs stuck open; PATH 3=reliefs fail to open	0.000063 0.000014	0.000098 0.000022
OP-FTC-1	Op. fails to isolate ND PATH 2=reliefs stuck open PATH 3=reliefs fail to open	0.00093 0.0199	0.0024 0.0844
FTD-LOCA	Op. fails to detect LOCA	0.025	0.0595
DIAG-LOCA	Op. fails to diagnose ISLOCA	0.025	0.067
OP-FTC-2	Op. fails to isolate ISLOCA	0.0179	0.019

Table 12: Summary of HRA Results for Scenario 1B

Scenario 1B: Overpressurization of ND System During Startup -
Bubble in Pressurizer

IDENTIFIER	HUMAN ERROR	NOMINAL HEP	MEAN HEP
ND-OPEN	Operator fails to close ND isolation valves ND1B & ND2A/ND36B & ND37A	0.0063	0.0082
OP-FTD-1	Op. fails to detect problem: PATH 2=reliefs stuck open; PATH 3=reliefs fail to open	0.000002 0.000014	0.000005 0.000022
OP-FTC-1	Op. fails to isolate ND PATH 2=reliefs stuck open PATH 3=reliefs fail to open	0.00093 0.0199	0.0024 0.0844
FTD-LOCA	Op. fails to detect LOCA	0.025	0.0595
DIAG-LOCA	Op. fails to diagnose ISLOCA	0.025	0.067
OP-FTC-2	Op. fails to isolate ISLOCA	0.0179	0.019

Table 13: Summary of HRA Results for Scenario 2

Scenario 2: Premature Entry Into Shutdown Cooling

IDENTIFIER	HUMAN ERROR	NOMINAL HEP	MEAN HEP
ND-OPEN	Op. opens ND suction MOVs too soon. (The remaining human errors for this scenario were not analyzed, since the HEP for the initiator, ND-OPEN, was estimated as negligible).	Negligible	Negligible

Table 14: Summary of HRA Results for Scenario 3

Scenario 3: Failure of Check Valves at NC/ND Pressure Isolation Boundary

IDENTIFIER	HUMAN ERROR	NOMINAL HEP	MEAN HEP
FTD-LOCA	Op. fails to detect LOCA	0.026	0.0607
DIAG-LOCA	Op. fails to diagnose ISLOCA	0.025	0.067
OP-FTC-2	Op. fails to isolate ISLOCA	0.0179	0.019

Inspection of these tables reveals that HEPs are largest for actions which occur following a break outside containment, actions such as detection of the LOCA, diagnosis of the ISLOCA, and isolation of the break. These relatively high error rates reflect the following circumstances. Procedures do not always effectively lead operators to the control room indications relevant for detection of a LOCA outside containment, and may not provide definitive guidance to applicable procedures or procedure sections. Procedures and training are focused on a single indicator for detection and diagnosis of ISLOCA. This indicator is annunciator EMF-41, an auxiliary building high radiation stack alarm. Procedures may not alert the operator to verify the presence or absence of this alarm in a timely manner. Furthermore, the detection of EMF-41 in an alarmed state is identified by alarm response procedures as a cue to initiate preparations for a site evacuation while ensuring isolation and positive pressurization of the control room. Therefore, operator workload is increased and significant (threat) stress is likely to be experienced by the operators at the time when ISLOCA isolation actions are required.

Conclusions

The HRA found that operator error could contribute to ISLOCA initiation. However, risk-significant human error initiators were judged to be unlikely during normal operations involving interfacing systems. This is due, in part, to the existence of vital administrative procedures and related operator training, as well as to the presence of well controlled interlocks which prevent the inadvertent operation of pressure isolation MOVs. The HRA found a significantly higher probability of operator error for detection, diagnosis, and isolation actions following a break outside containment. This resulted from an interaction of the following variables:

- 1 - A limited number of clear control room indications for ISLOCA,
- 2 - Procedures which may occupy operators for some time with steps related to the diagnosis of leaks or ruptures inside containment before directing them to indications of ISLOCA,
- 3 - Significant reliance on a single alarm, EMF 41, which is not referenced until the last page of the appropriate procedure and which limits the operators' ability to detect and diagnose ISLOCA in a timely manner,
- 4 - Limited amount of time in some scenarios for the detection, diagnosis, and isolation of ISLOCA before core damage,
- 5 - Operator exposure to high workload and threat stress at the time when actions to isolate an ISLOCA are needed and directed by procedure.

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Appendix D

Thermal Hydraulic Analysis

Appendix D
Thermal Hydraulic Analysis

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INTRODUCTION

The following calculations develop pressure distribution data within the residual heat removal (RHR) system. This information is required for the ISLOCA analysis. Initially, two questions must be answered. First, if the system high pressure isolation boundary fails, will the installed relief capacity be sufficient to protect the system from overpressurization? Second, if the system is overpressurized, what are the peak pressure distributions through the system?

The two ways for the RHR system high pressure isolation boundary to fail involve the premature opening of the shutdown cooling suction isolation valves, or failure of the cold leg injection check valves. Preliminary human factors analysis shows the first possibility to be extremely unlikely. Therefore, while the model described in this calculation package will contain the components needed to model both possibilities, the analysis will be centered on the second. The calculations are based on conservative assumptions; however, it was judged that these assumptions will not lead to excessively unrealistic estimates of the RHR system pressure distributions.

The approach used in this analysis is to develop a detailed RELAP5 model of a single train of the RHR system (including the discharge cross-connect and the discharge relief valve on the opposite train). "Detailed" implies sufficient resolution to determine the local pressures at components likely to fail should an overpressure event occur. The hot leg isolation failure scenario will use the entire single train model, the cold leg isolation failure scenario will use just that portion of the model between the cold leg isolation check valves and the discharge reliefs.

MODEL DEVELOPMENT

Figure 1 provides an overview of the RHR system as it is to be modeled for this analysis. The hot leg isolation failure model starts at the last train A upstream isolation valve (IND002) and continues the entire length of

train A to the cold leg injection isolation check valves. On the downstream end of the model, the normally open cross-connect valves and associated piping are included up to the train B discharge relief valve. This will provide three relief paths for the system, (1) the upstream relief valve IND003, (2) the discharge relief IND031 on train A, and (3) the discharge relief IND064 on train B. The train B upstream relief valve is not included because the pump discharge check valve in train B will prevent pressurization of the train B suction piping. The cold leg isolation failure model will contain only the flow path from the cold leg isolation check valves to the discharge reliefs on both trains.

Figure 2 provides a RELAP component diagram of the complete single train model. The following describes the RELAP input requirements for the model in Figure 2.

Figure 1. RHR system overview.

Figure 2. RELAP nodalization diagram for the RHR system model.

Miscellaneous Control Cards

card images:

```
= RHR SYSTEM DETAILED PIPING MODEL
*
*   MISCELLANEOUS CONTROL CARDS
*
100      new      transnt
101      run
102      british british
*
105      1.0      2.0      ALLOC
                        180.0
```

Time Step Control Cards

card images:

```
*
*   TIME STEP CONTROL CARDS
*
201      1.      1.0e-6 0.001 00002 10      5000 5000
202      2.      1.0e-6 0.01 00002 10      5000 5000
203      15.     1.0e-3 0.02 00002 5       5000 5000
```

Minor Edits

card images:

```
*
*   MINOR EDITS
*
301      p      110010000      * volume pressure
302      p      120220000      * flange      "
303      p      120230000      * flange      "
304      p      120240000      * pump 1A inlet pressure
305      p      130010000      * pump 1A outlet (flange) pressure
306      p      130720000      * FE 5040
307      p      130830000      * hx 1A inlet pressure
308      p      130840000      * hx 1A outlet pressure
309      p      140010000      * volume pressure
310      p      140130000      * FE 5190
311      p      200380000      * FE 5180
312      p      150010000      * volume pressure
313      p      160010000      * "      "
314      p      170010000      * "      "
315      p      190010000      * "      "
316      p      200010000      * "      "
```

Trips

card images:

```

*
*   TRIPS
*
*   501   Starts transient by opening motor operated valve
*   502   Keeps valve open for a long time
*   503   Opens relief valve 1ND003
*   504   Closes " " "
*   505   Opens relief valve 1ND031
*   506   Closes " " "
*   507   Opens relief valve 1ND064
*   508   Closes " " "
*
501   time    0      ge      null    0      0.0      n
502   time    0      ge      null    0      1.0e+6   n
503   p  190100000 ge      null    0      450.     n
504   p  190100000 le      null    0      427.5    n
505   p  150010000 ge      null    0      600.     n
506   p  150010000 le      null    0      570.     n
507   p  200440000 ge      null    0      600.     n
508   p  200440000 le      null    0      570.     n

```

Hydrodynamic Components

Assumptions:

1. Elevation changes are negligible within the model because of the large pressure differences involved. The maximum elevation change in the RHR system is about 50 ft. This gives a maximum elevation related pressure difference of about 20 psi. This is insignificant compared to the >2000 psi pressure difference driving the transient, and can be conservatively accounted for by just adding 20 psi to the pressure at the RCS boundary.
2. All bends can be adequately modeled using form loss coefficients (K factors). Modeling piping orientation (vector) changes is therefore not necessary since reaction forces are not considered in this calculation.
3. Dead leg piping will have no effect on steady state pressure distributions throughout the model. When fluid pressures within a dead leg volume are required, they can be obtained by using the pressure at the location where the dead leg connects with the main flow path.

4. Choking occurs only at the relief valves. Flow elements (orifices) within the system can be adequately modelled using form loss coefficients referenced to the pipe, not the bore, diameter.
5. There is no stored energy anywhere in the system being modeled.
6. The peak RHR system pressures will occur at steady state, and blowing down the primary system at RHR system relief valve flow rates will not depressurize the primary before steady state is reached in the RHR system.
7. Valve opening times will not affect the steady state solution and will be sufficiently long not to cause significant transient effects. Relief valve opening times of 1 second, and MOV opening times of 10 seconds are used because actual values are unknown. A failed check valve is assumed to open to reverse flow in 10 seconds. This assumption is based on a review of LER data performed for the Babcock and Wilcox ISLOCA study. This review turned up no instantaneous check valve failures. If instantaneous check valve failure were to be considered, transient effects would dominate the analysis, leading to transient pressures much higher than the steady state pressures considered in the current analysis.
8. Turbulent flow is assumed when calculating friction factors. This assumption is based on preliminary estimates of the Reynolds number (about 10^5) based on rated relief valve flows and known piping diameters.
9. Initial conditions within the RHR system are 50 psia pressure and 100° F temperature.

Friction data (applied throughout the model):

absolute roughness = 0.00015 ft

(Ref D-1, A-23)

Time Dependent Volume 100. Upstream boundary when modeling hot leg isolation failure. As stated in the introduction, hot leg isolation failure as a result of operator error is assumed to occur at hot standby conditions. Reactor coolant system temperature at hot standby is 557 °F and nominal system pressure is 2280 psia.

area = 1.0e+3 ft²

length = 1.0e+5

flags = 003

initial conditions:

pressure = 2280. psia

temperature = 557 degF

card images:

```
*
*   Time-dependent volume - upstream boundary
*
1000000 'HOTLEG'          tmdpvol
*   AREA   LENGTH  VOL    AZIM   INCL   ELEV   ROUGH   HYDR   FLAGS
1000101 1.0e+3  1.0e+5  0.00   0.00   0.00   0.00   0.00   0.00   00010
*   FLAGS
1000200 003
*
          PRESS   TEMP
1000201 0.00     2280.  557.
```

Valve Component 105. 12-inch motor-operated gate valve IND002A, the hot leg isolation valve. This component also accounts for the schedule 140-to-standard wall transition that occurs just downstream of IND002A. The piping between the hot leg and the isolation valve is neglected because it is not expected to cause a significant pressure drop at the flow velocities associated with rated relief valve capacities. This was verified with a hand calculation of the pressure drop caused by the seventy feet of piping and five ninety-degree bends between IDN002A and IND001B. Because of the large diameter of the piping the associated pressure drop at rated relief valve capacities is less than one psia. This estimate did not include the piping from IND001B to the hot leg. See Figure 3 for the summary sketch.

area = .7854 ft²

No area reduction modeled. Area is that of 12-inch standard wall piping.

flow resistance coefficients:

MOGV IND002A

$f_T = .013$

(Ref D-1, A-25)

$$K = 8 f_T = 0.104 \quad (\text{Ref D-1, A-27})$$

Transition piece (schedule 140 to standard wall)

theta = 45 deg (assumed)

$$\beta = d_1/d_2 = 10.5/12 = .875$$

$$K_2 = \frac{2.6 \sin(\theta/2)(1-\beta^2)^2}{\beta^4}$$

$$K_2 = .094 \quad (\text{Ref D-1, A-26, Formula 3})$$

flags = 000100

valve change rate = 0.1 s^{-1} Assume a valve stroke time of 10 seconds.

card images:

```

*
* Valve 1ND002A - last hot leg isolation valve
* 12-inch motor-operated gate valve
*
1050000 '1ND002A' valve
* FROM TO AREA FLOSS RLOSS FLAGS
1050101 100000000 110000000 0.7854 0.198 0.198 000100
1050201 1 0.00 0.00 0.00
1050300 mtrvly
1050301 501 502 0.10 0.00

```

Single Volume 110. This volume models the piping from upstream isolation valve 1ND002A to the tee leading to relief valve 1ND003 (see Figure 3).

Figure 3. Summary sketch of single volume 110

area = 0.7854 ft² 12-inch schedule 40S
(standard wall) (Ref D-1, B-17)

length = 3.0 ft

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100 degF

card images:

```
*
*   Single volume - downstream from 1ND002A
*   12-inch standard wall (schedule 40S)
*
1100000 'VOL-110'      snglvol
*   AREA  LENGTH  VOL  AZIM  INCL  ELEV  ROUGH  HYDR  FLAGS
1100101 0.7854  3.0   0.00  0.00  0.00  1.5e-4  0.00  00000
*   FLAGS  PRESS  TEMP
1100200 003    50.0  100.
```

Cross Flow Junction 114. Entrance to relief valve piping (for 1ND003). This is a 12-inch-to-4-inch tee junction. Flow is 90 deg to the 12-inch RHR pump suction line.

area = .0884 ft² Area is that of relief valve piping
(4-inch schedule 40).

flow resistance coefficients:

The following does not account for the reduction in flow area and may therefore be nonconservative. This is not expected to have a significant effect on the results.

$f_T = .017$ (Ref D-1, A-25)

$K = 60 f_T = 1.02$ (Ref D-1, A-29)

flags = 001002 No choking, smooth area change with
user-supplied loss coefficients, and cross
flow in the volume.

card images:

```
*
*   Cross flow junction - from RHR piping to relief valve piping
*
1140000 'TEE'          sngljun
*   FROM  TO  AREA  FLOSS  RLOSS  FLAGS
1140101 110000000 190000000 0.0884 1.02  1.02  001002
1140201 1      0.0    0.0    0.0
```

Single Junction 115. Connects Tee with remainder of RHR suction piping.

area = .7854 ft² Area is that of pump suction piping (12-inch standard wall).

flow resistance coefficients:

The flow resistance is that of a straight pass through the 12-inch run of the 12-inch-to-4-inch tee.

$f_T = .013$ (Ref D-1, A-25)

$K = 20 f_T = 0.26$ (Ref D-1, A-29)

flags = 001000 No choking.

card images:

```
*
*   Single junction - after tee, includes loss for straight pass through tee
*
1150000 'Jun-115'      sngljun
*      FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1150101 110010000 120000000 0.785  0.26  0.26  001000
1150201 1          0.0      0.0      0.0
```

Pipe Component 120. This component includes the piping between the tee to the upstream relief valve, IND003, and the suction to RHR pump 1A. See Figure 4 for a summary of the reference isometric drawing.

area = 0.7854 ft² 12-inch schedule 40S (standard wall) for the first 12 pipe volumes. (Ref D-1, B-17)

= 0.9575 ft² 14-inch schedule 30 (standard wall) for the next 12 volumes. (Ref D-1, B-17)

length = 94.8 ft Divide into 24 volumes, each 3.95 ft long.

flow resistance coefficients:

The flow resistances in the first 12 volumes include one 45-deg bend and three 90-deg bends. The second 12 volumes (in the 14-inch pipe) include the 12-inch-to-14-inch expansion, four 90-deg bends, and one tee. Will assume $r/d = 1.5$ for 90-deg bends everywhere in this component based on detail B on print CN-1492-ND001.

45-deg bend in 12-inch piping

$f_T = .013$ (Ref D-1, A-25)

$K = 15 f_T = 0.195$ (Ref D-1, A-29)

90-deg bend in 12- and 14-inch piping (they have the same f_T)

$f_T = .013$ (Ref D-1, A-25)

$$K = 14 f_T = 0.182 \quad (\text{Ref D-1, A-29})$$

12-inch-to-14-inch expansion. Use Formula 3 on p. A-26 of Reference D-1. Evaluate in terms of 12-inch diameter.

$$K = 2.6 \sin(\theta/2)(1-\beta^2)^2$$

where

$$\beta = d_1/d_2 = 1.000/1.104 = .906$$

$$\theta = 45 \text{ deg (assumed)}$$

$$K = 0.032$$

Tee in 14-inch piping (flow through branch)

$$f_T = .013 \quad (\text{Ref D-1, A-25})$$

$$K = 60 f_T = 0.780 \quad (\text{Ref D-1, A-29})$$

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

```

*
*   Pipe - from tee to RHR pump 1A suction
*   PS-601.2 - 12 and 14-inch standard wall
*
1200000 'P120'          pipe
1200001 24              * number of volumes
1200101 0.7854 12       * volume flow area
1200102 0.9575 13       * volume flow area
1200103 0.9575 24       * volume flow area
1200301 3.95 24         * " length
1200601 0.0 24         * " vertical orientation
1200801 1.5e-4 0.0 24   * " friction data
1200901 0.195 0.195 1   * pipe junction loss coefficients
1200902 0.182 0.182 2   * 90 deg bend
1200903 0.0 0.0 3       *
1200904 0.0 0.0 4       *
1200905 0.0 0.0 5       *
1200906 0.0 0.0 6       *
1200907 0.0 0.0 7       *
1200908 0.0 0.0 8       *
1200909 0.182 0.182 9   * 90 deg bend
1200910 0.0 0.0 10      *
1200911 0.0 0.0 11      *
1200912 0.214 0.215 12   * 90 deg bend + expansion
1200913 0.0 0.0 13       *
1200914 0.0 0.0 14       *
1200915 0.0 0.0 15       *
1200916 0.182 0.182 16   * 90 deg bend
1200917 0.0 0.0 17       *
1200918 0.182 0.182 18   * 90 deg bend
1200919 0.0 0.0 19       *
1200920 0.0 0.0 20       *
1200921 0.780 0.780 21   * Tee - flow through branch
1200922 0.182 0.182 22   * 90 deg bend
1200923 0.182 0.182 23   * 90 deg bend

```

1201001	00	24				* volume control flags
1201101	01000	23				* junction control flags
1201201	003	50.0	100.0	0.	0.	0. 0. 24
1201301	0.0	0.0	0.0	23

Figure 4. Summary sketch of pipe component 120

Single Junction 125. RHR pump 1A. This analysis does not require a detailed pump model since the pump will be turned off during system pressurization. The 1A RHR pump is modeled as a single junction for convenience, and to allow for later upgrading if it should become desirable. The pump volume is neglected.

area = .3601 ft² Area is that of downstream piping (8 inch-schedule 20).

flow resistance coefficients:

The flow resistance is that of the 1A RHR pump. By engineering judgement,

K = 10.0

flags = 001000 No choking, smooth area change with user input loss coefficients.

card images:

```
*
*   Single junction - RHR pump 1A
*
1250000 'ND1A'          sngljun
*      FROM      TO      AREA   FLOSS   RLOSS   FLAGS
1250101 120010000 130000000 0.3601 10.0   10.0   001000
1250201 1         0.0     0.0     0.0
```

Pipe Component 130. This component lumps together the piping between the discharge of the 1A RHR pump and the cross-connect to the 1B injection piping. See Figures 5, 6, 7, and 8 for a summary of the reference isometric drawing. Note that this will be a dead leg volume in the cold leg isolation failure case because check valve 1ND010 on the pump discharge will prevent back flow to the suction relief valve.

Figure 5. Summary sketch of pipe component 130

Figure 6. Summary sketch of pipe component 130 (cont.)

Figure 7. Summary sketch of pipe component 130 (cont.)

area = 0.3601 ft² 8-inch schedule 20 (Ref D-1, B-17)

length = 339.2 ft Divide into 85 volumes, each 4 ft long.

flow resistance coefficients:

90-deg bend

$f_T = .014$ (Ref D-1, A-25)

$K = 14 f_T = 0.196$ (Ref D-1, A-29)

8-inch swing check valve (ND010)

$f_T = .014$ (Ref D-1, A-25)

$K = 100 f_T = 1.40$ (Ref D-1, A-29)

8-inch motor operated gate valves 1ND067, 1ND019, and 1ND026

$f_T = .014$ (Ref D-1, A-25)

$K = 8 f_T = .112$ (Ref D-1, A-29)

Orifice, flow element FE 5040

$A_f = \pi * D^2 / 4$ The flow area reduction will not be
used unless results suggest the
possibility of choking anywhere
besides the relief valves.

$A_f = 3.14(5.138/12)^2/4 = .1440 \text{ ft}^2$

$\beta = 5.138/8.125 = .632$

$C = .66$ (Ref D-1, A-20)

$k = \frac{1 - \beta^2}{C^2 * \beta^4} = 8.62$

Tee, flow through branch

$f_T = .014$ (Ref D-1, A-25)

$K = 60 f_T = .840$ (Ref D-1, A-29)

Heat exchanger

by engineering judgment

$K = 3.0$

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

*

```

*   Pipe - RHR pump 1A discharge to cross connect to 1B discharge
*   PS-601.2 - 8 inch schedule 20
*
1300000 'VOL-130'      pipe
1300001 85              * number of volumes
1300101 0.3601 85       * volume flow area
1300201 0.3601 1        * junction flow areas
1300202 0.3601 71       *
1300203 0.3601 72       * FE 5040
1300204 0.3601 73       *
1300205 0.3601 84       *
1300301 4.00 85         * volume length
1300601 0.0 85          * " vertical orientation
1300801 1.5e-4 0.0 85   * " friction data
1300901 0.196 0.196 1   * pipe junction loss coefficients
1300902 1.708 1.708 2   * 90 deg bend, check valve, MOGV
1300903 0.0 0.0 3       *
1300904 0.196 0.196 4   * 90 deg bend
1300905 0.0 0.0 5       *
1300906 0.0 0.0 16      *
1300907 0.196 0.196 17  * 90 deg bend
1300908 0.0 0.0 18      *
1300909 0.196 0.196 19  * 90 deg bend
1300910 0.0 0.0 20      *
1300911 0.196 0.196 28  * 90 deg bend
1300912 0.0 0.0 29      * 90 deg bend + expansion
1300913 0.196 0.196 30  *
1300914 0.0 0.0 31      *
1300915 0.0 0.0 32      *
1300916 0.196 0.196 33  * 90 deg bend
1300917 0.0 0.0 34      *
1300918 0.0 0.0 37      *
1300919 0.196 0.196 38  * 90 deg bend
1300920 0.0 0.0 39      *
1300921 0.196 0.196 40  * 90 deg bend
1300922 0.0 0.0 41      *
1300923 0.0 0.0 47      *
1300924 0.196 0.196 48  * 90 deg bend
1300925 0.0 0.0 49      *
1300926 0.0 0.0 57      *
1300927 0.196 0.196 58  * 90 deg bend
1300928 0.0 0.0 59      *
1300929 0.0 0.0 64      *
1300930 0.196 0.196 65  * 90 deg bend
1300931 0.0 0.0 66      *
1300932 0.196 0.196 67  * 45 deg bend
1300933 0.196 0.196 68  * 90 deg bend
1300934 0.0 0.0 69      *
1300935 0.0 0.0 71      *
1300936 8.62 8.62 72    * FE 5040
1300937 0.0 0.0 73      *
1300938 0.0 0.0 74      *
1300939 0.196 0.196 75  * 90 deg bend
1300940 0.196 0.196 76  * 90 deg bend
1300941 0.0 0.0 77      *
1300942 0.0 0.0 80      *
1300943 0.308 0.308 81  * MOGV 1ND019, 90 deg bend
1300944 0.196 0.196 82  * 90 deg bend
1300945 3.0 3.0 83      * 1A heat exchanger
1300946 1.148 1.148 84  * 90 deg bend, Tee, MOGV
1301001 00 85          * volume control flags
1301101 01000 84       * junction control flags
1301201 003 50.0 100.0 0. 0. 85
1301301 0.0 0.0 0.0 84

```

Single Junction 135. Tee, splits flow from train 1A pump discharge piping to

train 1A and 1B injection piping. This junction represents that part of the flow path leading to the 1A injection piping (and hence relief valve 1ND031). This junction is not used in the cold leg isolation failure case because pipe volume 130 will be dead legged in this case. See Figure 8 for the summary sketch.

area = .180 ft² Area is half that of 8-inch schedule 20 pipe.

flow resistance coefficients:

Standard tee, flow through branch

$f_T = .014$ (Ref D-1, A-25)

$K = 60 f_T = 0.840$ (Ref D-1, A-29)

flags = 001000 No choking, smooth area change with user input loss coefficients.

card images:

```
*
*   Single junction - Tee to downstream piping
*
1350000 'J135'          sngljun
*   FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1350101 130010000 140000000 0.180 0.840 0.840 001000
1350201 1          0.0      0.0      0.0
```

Single Junction 136. Tee, splits flow between train 1A discharge piping and train 1A and 1B injection piping. This junction represents that part of the flow path leading to the 1B injection piping (and hence relief valve 1ND064). Like junction 135, this junction is not used in the cold leg isolation valve failure case.

area = .3601/2 = .180 ft² Flow area is half of 8-inch schedule 20 pipe area.

flow resistance coefficients:

flow through branch

$f_T = .014$ (Ref D-1, A-25)

$K = 60 f_T = 0.840$ (Ref D-1, A-29)

flags = 001000 No choking, smooth area change with user input loss coefficients.

card images:

```
*
*   Single junction - Tee, flow through branch
*
1360000 'JUN-136'      sngljun
```

*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS
1360101	130010000	200000000	0.180	0.84	0.84	001000
1360201	1	0.0	0.0			

Single Junction 137. Tee, splits flow between train 1A discharge piping and the 1A and 1B injection piping. This junction represents that part of the flow path connecting the 1A injection piping to the 1B injection piping. In the cold leg isolation failure case this junction will include the entire pipe flow area instead of half as in the hot leg isolation failure case.

area = $.360/2 = .180 \text{ ft}^2$ Half of the 8-inch schedule 20 pipe area.

flow resistance coefficients:

flow through branch

$f_T = .014$ (Ref D-1, A-25)

$K = 60 f_T = 0.840$ (Ref D-1, A-29)

flags = 001000

card images:

```

*
*   Single junction - Tee, flow through branch
*
1370000 'JUN-137'      sngljun
*   FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1370101 140000000 200000000 0.180 0.84 0.84 001000
1370201 1        0.0    0.0    0.0

```

Pipe Component 140. This component includes the train 1A injection piping from the cross connect to the 1B train on to the junction with relief valve 1ND031. See Figure 9 for a summary of the reference isometric drawing.

Figure 9. Summary sketch of pipe components 140 and 160

area = 0.3601 ft² 8-inch schedule 20 (Ref D-1, B-17)
length = 123.1 ft Divide into 30 volumes, each 4.1 ft long.

flow resistance coefficients:

90-deg bend

$f_T = .014$ (Ref D-1, A-25)

$K = 14 f_T = 0.196$ (Ref D-1, A-29)

Orifice, FE 5190

The flow area reduction will not be used
unless results suggest choking anywhere
besides the relief valves.

$A_f = 3.142 (6.33/12)^2/4 = .219 \text{ ft}^2$

$\beta = 6.33/8.125 = .78$

$C = .76$ (Ref D-1, A-20)

$k = \frac{1 - \beta^2}{C^2 * \beta^4} = 1.83$

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

```
*
*   Pipe - from branch at cold leg injection piping to relief valve Tee
*
1400000 'P140'           pipe
1400001 30               * number of volumes
1400101 0.3601 30        * volume flow area
1400201 0.3601 1         * junction area
1400202 0.3601 12        *
1400203 0.3601 13        * FE 5190
1400204 0.3601 14        *
1400205 0.3601 29        *
1400301 4.1 30           * volume length
1400601 0.0 30           * vertical orientation
1400801 1.5e-4 0.0 30    * friction data
1400901 0.196 0.196 1    * 90 deg bend
1400902 0.196 0.196 2    * 90 deg bend
1400903 0.0 0.0 3        *
1400904 0.0 0.0 4        *
1400905 0.196 0.196 5    * 90 deg bend
1400906 0.196 0.196 6    * 90 deg bend
1400907 0.196 0.196 7    * 90 deg bend
1400908 0.0 0.0 8        *
1400909 0.0 0.0 12       *
1400910 1.83 1.83 13     * FE 5190
1400911 0.0 0.0 14       *
1400912 0.0 0.0 19       *
1400913 0.196 0.196 20   * 90 deg bend
```

1400914	0.196	0.196	21						* 90 deg bend
1400915	0.196	0.196	22						* 90 deg bend
1400916	0.0	0.0	23						*
1400917	0.196	0.196	24						* 90 deg bend
1400918	0.196	0.196	25						* 90 deg bend
1400919	0.0	0.0	26						*
1400920	0.196	0.196	27						* 90 deg bend
1400921	0.0	0.0	28						*
1400922	0.0	0.0	29						*
1401001	00	30							*
1401101	001000	29							* junction control flags
1401201	003	50.0	100.0	0.	0.	0.	0.	30	
1401301	0.0	0.0	0.0	29					

Single Junction 145. Joins piping to single volume at entrance to 1ND031.

area = .3601 ft² 8-inch schedule 20.

flow resistance coefficients:

none

flags = 001000 No choking, smooth area change with user
input loss coefficients.

card images:

```

*
*   Single junction - Piping to single volume
*
1450000 'JUN-145'      sngljun
*      FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1450101 140010000 150000000 0.3601 0.0      0.0      001000
1450201 1          0.0      0.0      0.0

```

Single Volume 150. This volume is located at the tee splitting flow to 1ND031 and the 1A injection piping (see Figure 9).

area = 0.3601 ft² 8-inch schedule 20 (Ref D-1, B-17)

length = 4.0 ft

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

```

*
*   Single volume - at junction to 1ND031 piping
*
1500000 'VOL-150'      snglvol
*      AREA  LENGTH  VOL      AZIM      INCL      ELEV      ROUGH      HYDR      FLAGS
1500101 0.3601 4.0      0.00      0.00      0.00      0.00      1.5e-4 0.00      00
*      FLAGS  PRESS   TEMP
1500200 003      50.      100.

```

Valve Component 154. Relief valve 1ND031. The short run of piping between

the injection piping and the valve is neglected (two feet of 8-inch and 4-inch piping). Valve 1ND031 has the following characteristics (Nuclear Nameplate Data from Order Control Dwg. No. 3NC-051, sheet 2A):

Set pressure = 600 psi, Accumulation = 60 psi, Blowdown = 30 psi

Rated flow = 92000 lb/hr (25.6 lb/sec) at 400° F

A problem encountered here is that the nameplate data and the tag plate data do not agree. The above nameplate data indicate a valve capacity of about 214 gpm (25.6 lb/sec) at 600 psi and 400° F while tag plate data indicate a capacity of 585 gpm (80.9 lbm/sec) at 720 psi and 100° F. The difference in the pressures and temperatures specified does not explain the flow difference. The transients will be run both ways to see if this discrepancy has a significant effect.

The RELAP modeling parameters follow:

area = 1.846E-3 ft² From stand-alone RELAP calculation (based on 25.6 lb/sec at 600 psi and 400° F)

area = 3.783E-3 ft² From critical flow table for 720 psia and 100° F (based on 80.9 lb/sec)

flow resistance coefficients:

none

flags = 000100 Choked flow.

valve change rate = 1 sec⁻¹

card images:

```

*
*   Relief valve 1ND031
*   size = 4 in      set pressure = 600 psi   flow = 92000 lb/hr
*   bore = 1.99 in   temperature  = 400 F     = 25.6 lb/sec
*   accu = 60 psi     blowdown = 30 psi
*
1540000 '1ND031'      valve
*   FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1540101 150010003 998000000 1.85e-3 0.00  0.00  000100
1540201 1          0.00  0.00  0.00
1540300 mtrvlv
1540301 505      506      1.00  0.00

```

Single Junction 155. Tee, flow through branch to injection piping.

area = .2532 ft² 8-inch schedule 160. Use downstream piping diameter.

flow resistance coefficients:

Tee, flow through branch

$$f_T = .014 \quad (\text{Ref D-1, A-25})$$

$$K = 60 f_T = 0.840 \quad (\text{Ref D-1, A-29})$$

MOGV 1NI173A

$$f_T = .014 \quad (\text{Ref D-1, A-25})$$

$$K = 8 f_T = 0.112 \quad (\text{Ref D-1, A-29})$$

Schedule change (8-inch schedule 20 to 160)

$$\theta = 45 \text{ deg (assumed)}$$

$$\beta = d_1/d_2 = 7.187/8.125 = .885$$

$$K_1 = 0.8 \sin(\theta/2)(1-\beta^2)$$

$$K_1 = .066 \quad (\text{Ref D-1, A-26, Formula 1})$$

flags = 001000

No choking, smooth area change with user
input loss coefficients.

card images:

```
*
*   Single junction - Tee to downstream piping
*
1550000 'JUN-155'      sngljun
*   FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1550101 150010000 160000000 0.2532 1.018 1.018 001000
1550201 1      0.0    0.0    0.0
```

Pipe Component 160. This component includes the train 1A injection piping from relief valve 1ND031 to split before the cold leg isolation check valves. See Figures 9 and 10 for a summary of the reference isometric drawing.

Figure 10. Summary sketch of pipe component 160

area = 0.2532 ft² 8-inch schedule 160 (Ref D-1, B-17)
length = 26.4 ft Divide into 6 volumes, each 4.4 ft long.

flow resistance coefficients:

90-deg bend

$f_T = .015$ (Ref D-1, A-25)

$K = 14 f_T = 0.210$ (Ref D-1, A-29)

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

```
*
*   Pipe - from relief valve tee to branch at cold leg injection piping
*
1600000 'JUN-160'      pipe
1600001 6               * number of volumes
1600101 0.2532 6       * volume flow area
1600301 4.4 6          * length
1600601 0.0 6          * vertical orientation
1600801 1.5e-4 0.0 6   * friction data
1600901 0.0 0.0 1     * pipe junction loss coefficients
1600902 0.210 0.210 2 * 90 deg bend
1600903 0.0 0.0 3     *
1600904 0.0 0.0 4     *
1600905 0.210 0.210 5 *
1601001 00 6          * volume control flags
1601101 001000 5      * junction control flags
1601201 003 50.0 100.0 0. 0. 0. 6
1601301 0.0 0.0 0.0 5
```

Single Junction 165. Tee with schedule change. Piping splits to cold leg isolation check valves and changes from 8-inch schedule 160 to 6-inch schedule 160.

area = .1469 ft² 6-inch schedule 160.

flow resistance coefficients:

Tee, flow through branch

$f_T = .015$ (Ref D-1, A-25)

$K = 60 f_T = 0.900$ (Ref D-1, A-29)

Pipe spec. change (8-inch to 6-inch schedule 160)

theta = 45

beta = $d_1/d_2 = 5.187/6.813 = .761$

$$K1 = 0.8 \sin(\theta/2)(1-\beta^2) = .129$$

(Ref D-1, A-26, Formula 1)

flags = 001000

No choking, smooth area change with user-input loss coefficients.

card images:

```
*
*   Single junction - Tee to downstream piping
*
1650000 'JUN-165'      sngljun
*   FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1650101 160010000 1700000000 0.1469 1.03  1.03  001000
1650201 1      0.0    0.0    0.0
```

Single Junction 166. Not required at this time

Pipe Component 170. This component includes the train 1A injection piping from the tee after 1NI173 to the first isolation check valve 1NI175. The piping from 1NI175 to the cold leg is not included because it is not expected to cause a significant pressure drop. Flow at the rated capacity of the relief valves will cause less than one psi pressure drop through the 40 feet of piping between 1NI175 and 1NI082, according to a simple hand calculation. See Figure 10 for a summary of the reference isometric drawing.

area = 0.1469 ft² 6-inch schedule 160 (Ref D-1, B-17)
length = 26.0 ft Divide into 6 volumes, each 4.3 ft long.

flow resistance coefficients:

90-deg bend

$$f_T = .015 \quad (\text{Ref D-1, A-25})$$

$$K = 14 f_T = 0.210 \quad (\text{Ref D-1, A-29})$$

Orifice, FE 5890 (Area reduction not used)

$$A_f = (3.142/4)(3.056/12)^2 = 0.051 \text{ ft}^2$$

$$\beta = 3.056/5.187 = 0.589$$

$$C = 0.65 \quad (\text{Ref D-1, A-20})$$

$$k = \frac{1 - \beta^2}{C^2 * \beta^4} = 12.8$$

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

```
*
*   Pipe - from branch to cold leg D isolation check valve
*
1700000 'P170'          pipe
1700001 6                * number of volumes
1700101 0.1469 6         * volume flow area
1700201 0.1469 1         * junction flow area
1700202 0.1469 4         *
1700203 0.1469 5         *
1700301 4.3 6            * volume length
1700601 0.0 6            * vertical orientation
1700801 1.5e-4 0.0 6     * friction data
1700901 0.0 0.0 1        * pipe junction loss coefficients
1700902 0.420 0.420 2    * 2 90 deg bends
1700903 0.210 0.210 3    * 90 deg bend
1700904 0.210 0.210 4    * 90 deg bend
1700905 12.8 12.8 5      * FE 5890
1701001 00 6             * volume control flags
1701101 01000 5          * junction control flags
1701201 003 50. 100.0 0. 0. 0. 6
1701301 0.0 0.0 0.0 5
```

Check Valve Component 175. First cold leg boundary isolation check valve and model boundary. The piping between the cold leg and the first isolation check valve is neglected. This is high pressure-rated piping and is therefore not of interest in the ISLOCA analysis. The total pressure drop in this piping is expected to be insignificant based on a simple hand calculation. In the hot leg isolation failure case, this is a check valve junction. In the cold leg isolation failure case, this is a motor-operated valve with an opening time of 10 seconds. See Figure 10.

area = .1469 ft² No area reduction modeled. Area is that of 6-inch schedule 160.

flow resistance coefficients:

swing check valve

$f_T = .015$ (Ref D-1, A-25)

$K = 100 f_T = 1.5$ (Ref D-1, A-27)

flags = 001000 No choking, smooth area change with user input loss coefficients.

card images:

```
*
*   Valve 1N1175 - first cold leg C isolation check valve
*
1750000 '1N1175'        valve
*   FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1750101 1700000000 9990000000 0.1469 1.50 1.50 001000
1750201 1 0.0 0.0 0.0
```

1750300 chkvlv
 1750301 1 1 1.0 0.0

Pipe Component 180. This component is not needed. It is included only because it may be necessary to develop the model for the check valve failure sequences at a later date.

Check Valve Component 185. This component is not needed at present (see component 180 above).

Pipe Component 190. Piping from RHR pump 1A suction line to relief valve IND003 (See Figure 3). Includes 10.4 ft of 4-inch schedule 40 piping. There are three 90-deg bends in this piping run. The ratio of bend radius (r) to pipe diameter (d) is unknown; therefore assume it is 1.0.

area = 0.0884 ft² Piping is PS-601.2 -- 4-inch
 schedule 40. (Ref D-1, B-17)

length = 10.9 ft

flags = 003

flow resistance coefficients:

90-deg bends:

$f_T = .017$ (Ref D-1, A-25)

$K = 20 f_T = 0.34$ (Ref D-1, A-29)

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

```
*
* Pipe - from tee to RV IND003
*
1900000 'RV-PIPE'      pipe
1900001 10
1900101 0.0884 10      * number of volumes
1900301 1.09 10        * volume flow area
1900601 0.0 10         * length
1900801 1.5e-4 0.0 10  * vertical orientation
1900901 0.34 0.34 1    * friction data
1900902 0.0 0.0 2      * 90 deg bend
1900903 0.34 0.34 3    * 90 deg bend
1900904 0.0 0.0 7      *
1900905 0.34 0.34 8    * 90 deg bend
```

1900906	0.0	0.0	9	*			
1901001	00	10		*	volume control flags		
1901101	01000	9		*	junction control flags		
1901201	003	50.	100.0	0.	0.	0.	10
1901301	0.0	0.0	0.0	9			

Valve Component 195. Relief valve IND003. Modeled as a motor controlled valve strictly for convenience. The flow area is determined with a stand-alone RELAP job not included in this package. The area is calculated to produce rated flows at rated conditions. The following valve data was obtained from Order Control Dwg. No. 3NC-051:

size = 4 in, bore = 2.97 in

set pressure = 450 psi, accumulation = 45 psi, blowdown = 22.5

flow = 154935 lb/hr (43. lb/s) at 400 degF

RELAP input:

area = $3.94\text{E-}3 \text{ ft}^2$

Area is that required to produce rated flow under rated conditions, given choked flow in the valve.

flow resistance coefficients:

none

flags = 00100

valve change rate = 1 sec^{-1} Assume a valve stroke time of 1 second.

card images:

```

*
*   Valve IND003 - RHR relief valve
*   size = 4 in      set pressure = 450 psi   flow = 154935 lb/hr
*   bore = 2.97 in   temperature = 400 F     = 43 lb/sec
*
1950000 'IND003'      valve
*   FROM      TO      AREA  FLOSS  RLOSS  FLAGS
1950101 190010000 998000000 3.94e-3 0.00  0.00  00100
1950201 1          0.00  0.00  0.00
1950300 mtrvLv
1950301 503      504      1.00  0.00

```

Pipe Component 200. This component includes piping from the cross-connect to the 1B train on to relief valve IND064. See Figure 8 for a summary of the reference isometric drawing.

area = 0.3601 ft^2

8-inch schedule 20 (Ref D-1, B-17)

length = 175.4 ft

Divide into 44 volumes, each 3.99 ft long.

flow resistance coefficients:

MOGV 1ND032A

$$f_T = .014 \quad (\text{Ref D-1, A-25})$$

$$K = 8 f_T = 0.112 \quad (\text{Ref D-1, A-27})$$

90-deg bends

$$f_T = .014 \quad (\text{Ref D-1, A-25})$$

$$K = 14 f_T = 0.196 \quad (\text{Ref D-1, A-29})$$

Tee, flow through branch

$$f_T = .014 \quad (\text{Ref D-1, A-25})$$

$$K = 60 f_T = 0.840 \quad (\text{Ref D-1, A-29})$$

Expansion

$$\theta = 45$$

$$\beta = d_1/d_2 = 8.125/12.25 = .663$$

$$K_1 = 2.6 \sin(\theta/2)(1-\beta^2)^2 \quad (\text{Ref D-1, A-26, Formula 3})$$

$$K_1 = .313$$

Contraction

$$\theta = 45$$

$$\beta = d_1/d_2 = 8.125/12.25 = .663$$

$$K_1 = 0.8 \sin(\theta/2)(1-\beta^2) \quad (\text{Ref D-1, A-26, Formula 1})$$

$$K_1 = .172$$

Orifice, FE 5180 (Area reduction not used)

$$A_f = (3.142/4)(6.335/12)^2 = 0.219 \text{ ft}^2$$

$$\beta = 6.335/8.125 = 0.78$$

$$C = 0.76 \quad (\text{Ref D-1, A-20})$$

$$k = \frac{1 - \beta^2}{C^2 * \beta^4} = 1.83$$

flags = 003

initial conditions:

pressure = 50 psia

temperature = 100° F

card images:

```
*
*   Pipe - from cross-connect to 1ND064
*
2000000 'P200'      pipe
2000001 44
2000101 0.3601  44
2000201 0.3601  1
```

* number of volumes
* volume flow area
* junction flow area

2000202	0.3601	37			*
2000203	0.3601	38			*
2000204	0.3601	39			*
2000205	0.3601	43			*
2000301	3.99	44			* volume length
2000601	0.0	44			* vertical orientation
2000801	1.5e-4	0.0	44		* friction data
2000901	0.112	0.112	1		* MOGV
2000902	0.0	0.0	2		*
2000903	0.196	0.196	3		* 90 deg bend
2000904	0.196	0.196	4		* 90 deg bend
2000905	0.0	0.0	5		*
2000906	0.196	0.196	6		* 90 deg bend
2000907	0.0	0.0	7		*
2000908	0.84	0.84	8		* Tee
2000909	0.596	0.596	9		* Exp, Contr, MOGV
2000910	0.0	0.0	10		*
2000911	1.68	1.68	11		* 2 Tees
2000912	0.0	0.0	12		*
2000913	0.0	0.0	14		*
2000914	0.392	0.392	15		* 2 90 deg bends
2000915	0.0	0.0	16		*
2000916	0.0	0.0	17		*
2000917	0.196	0.196	18		* 90 deg bend
2000918	0.0	0.0	19		*
2000919	0.0	0.0	20		*
2000920	0.196	0.196	21		* 90 deg bend
2000921	0.0	0.0	22		*
2000922	0.196	0.196	23		* 90 deg bend
2000923	0.0	0.0	24		*
2000924	0.0	0.0	25		*
2000925	0.196	0.196	26		* 90 deg bend
2000926	0.0	0.0	27		*
2000927	0.196	0.196	28		* 90 deg bend
2000928	0.196	0.196	29		* 90 deg bend
2000929	0.0	0.0	30		*
2000930	0.0	0.0	37		*
2000931	1.83	1.83	38		* FE 5180
2000932	0.0	0.0	39		* 90 deg bend
2000933	0.392	0.392	40		* 2 45 deg bends
2000934	0.0	0.0	41		*
2000935	0.0	0.0	42		*
2000936	0.196	0.196	43		* 90 deg bend
2001001	00	44			* volume control flags
2001101	001000	43			* junction control flags
2001201	003	50.	100.0	0.	0. 0. 44
2001301	0.0	0.0	0.0	43	

Valve Component 205. Relief valve IND064. Same as valve IND031 (see component 154).

$$\text{area} = 1.85\text{E-}3 \text{ ft}^2$$

Area is determined to achieve rated flow at rated conditions using a critical flow model (see component 154).

$$\text{area} = 3.783\text{E-}3 \text{ ft}^2$$

From critical flow table.

flow resistance coefficients:

none

$$\text{flags} = 001$$

This junction is not modeled cross-flow.

$$\text{valve change rate} = 1. \text{ s}^{-1}$$

Assume a valve stroke time of 1 second.

card images:

```
*
* Valve 1ND064 - RHR relief valve
* size = 4 in      set pressure = 600 psi    flow = 92000 lb/hr
* bore = 1.99 in   temperature = 400 F      = 25.6 lb/s
*
2050000 '1ND064'      valve
* FROM      TO      AREA  FLOSS  RLOSS  FLAGS
2050101 200010000 998000000 1.85e-3 0.00 0.00 00100
2050201 1      0.00 0.00 0.00
2050300 mtrvlv
2050301 507      508      1.00 0.00
```

Time Dependent Volume 998. Atmospheric boundary condition. Relief valve blowdown is to a tank containing liquid water; however, the relief valve discharge piping is possibly air- and vapor-filled. Force a vapor boundary condition by using a higher temperature.

area = 1.0e+3

length = 1.0e+5

flags = 003

initial conditions:

pressure = 14.7 psia

temperature = 215° F

card images:

```
*
* Time-dependent volume - atmosphere boundary
*
9980000 'ATMOS'      tmdpvol
* AREA  LENGTH  VOL  AZIM  INCL  ELEV  ROUGH  HYDR  FLAGS
9980101 1.0e+3 1.0e+5 0.0 0.0 0.0 0.0 0.0 10
9980200 003
9980201 0.0      14.7 215.
```

Time Dependent Volume 999. Downstream boundary (cold leg) used for the cold leg isolation failure scenario. This scenario could occur with the plant at 100% power. Cold leg temperature is 562° F at 100% power. Cold leg pressure is the hot leg pressure of 2280 psia plus the core pressure drop of 46 psi. To assure that the elevation head in the RHR system is conservatively accounted for without modeling each elevation change, the total elevation head within the system can be added to the boundary pressure. The maximum elevation change within the system is 54 ft. This gives a maximum pressure drop of 23 psi.

area = 1.0e+3

length = 1.0e+5

flags = 003

initial conditions:

pressure = 2280 + 46 + 23 = 2349 psia

temperature = 562° F

card images:

```
*
* Time-dependent volume - cold leg boundary
*
9990000 'COLDLEG'      tmdpvol
*      AREA  LENGTH VOL    AZIM    INCL    ELEV    ROUGH    HYDR    FLAGS
9990101 1.0e+3 1.0e+5 0.0    0.0    0.0    0.0    0.0    0.0    10
9990200 003
9990201 0.0      2349.  562.
```

RESULTS

The following describes the RELAP5 results for the RHR system cold leg isolation failure sequence. The edited RELAP5 output for this sequence is provided in Appendix 1. It contains the input deck card images, selected minor edit results for the approach to steady state, and a major edit at final steady state conditions. The results show that the system reaches steady state pressure within a matter of seconds. This is well before cold water in the system is flushed out through the relief valves. However, running the problem until steady state temperature distributions were obtained did not significantly affect the results. Pressures within the system vary only slightly because of the low flow rates allowed by the discharge relief valves, and a single pressure of 2306 psia is adequate to describe the pressure distribution within the RHR system. Further refinement of this value is possible, but would produce only minor changes in the calculation of system overpressure failure probability.

One important observation must be made about these results. The maximum pressure seen by the RHR system is sensitive to the time required for a check valve to open to reverse flow. With opening times on the order of 10 seconds, transient effects are not important, and the steady state pressure is the maximum pressure seen by the system. With opening times on the order of 1 second, transient effects cause pressure pulses within the system 500 psi greater than the final steady state pressures. Since previous work for a Babcock and Wilcox plant has shown instantaneous check valve failure to be an incredible event, the transient effects were not thoroughly explored.

Another consideration involves the difference between nuclear nameplate data and tag plate data for relief valve capacity. A sensitivity case was run using both tag plate and name plate data. Using tag plate data results in the pressure distributions provided in Appendix 1. Using name plate data results in pressures about 30 psi higher. Since the tag plate data is consistent with the information provided on system P&IDs, it is considered to be the appropriate data for use in this analysis. The results from the nameplate-based data are not included in Appendix 1.

REFERENCES

- D-1. Flow of Fluids Through Valves, Fittings, and Pipe, Crane Technical Paper 410, Crane Co., 300 Park Avenue, New York, N.Y. 10022, 1982.

HOT LEG ISOLATION FAILURE INPUT DECK LISTING

= RHR SYSTEM DETAILED PIPING MODEL

*

* MISCELLANEOUS CONTROL CARDS

*

100	new	transnt	
101	run		
102	british	british	
*			ALLOC
105	1.0	2.0	180.0

*

* TIME STEP CONTROL CARDS

*

201	1.	1.0e-6	0.001	00002	10	5000	5000
202	2.	1.0e-6	0.01	00002	10	5000	5000
203	15.	1.0e-3	0.02	00002	5	5000	5000

*

* MINOR EDITS

*

pressure	301	p	110010000	* volume pressure
	302	p	120220000	* flange "
	303	p	120230000	* flange "
	304	p	120240000	* pump 1A inlet pressure
	305	p	130010000	* pump 1A outlet (flange)
	306	p	130720000	* FE 5040
	307	p	130830000	* hx 1A inlet pressure
	308	p	130840000	* hx 1A outlet pressure
	309	p	140010000	* volume pressure
	310	p	140130000	* FE 5190
	311	p	200380000	* FE 5180
	312	p	150010000	* volume pressure
	313	p	160010000	* " "
	314	p	170010000	* " "
	315	p	190010000	* " "
	316	p	200010000	* " "

*

* TRIPS

*

*	501	Starts transient by opening motor operated valve
*	502	Keeps valve open for a long time
*	503	Opens relief valve 1ND003
*	504	Closes " " "
*	505	Opens relief valve 1ND031
*	506	Closes " " "
*	507	Opens relief valve 1ND064
*	508	Closes " " "

*

501	time	0	ge	null	0	0.0	n
502	time	0	ge	null	0	1.0e+6	n
503	p	190100000	ge	null	0	450.	n
504	p	190100000	le	null	0	427.5	n
505	p	150010000	ge	null	0	600.	n
506	p	150010000	le	null	0	570.	n
507	p	200440000	ge	null	0	600.	n
508	p	200440000	le	null	0	570.	n

*
* Time-dependent volume - upstream boundary
*

		'HOTLEG'		tmdpvol				
		AREA	LENGTH	VOL	AZIM	INCL	ELEV	ROUGH
HYDR	FLAGS							
	1000101	1.0e+3	1.0e+5	0.00	0.00	0.00	0.00	0.00
0.00	00010							
		* FLAGS						
	1000200	003						
			PRESS	TEMP				
	1000201	0.00	2280.	557.				

*
* Valve 1ND002A - last hot leg isolation valve
* 12-inch motor-operated gate valve
*

		'1ND002A'		valve				
		FROM	TO	AREA	FLOSS	RLOSS	FLAGS	
	1050101	100000000	110000000	0.7854	0.198	0.198	000100	
	1050201	1	0.00	0.00				
	1050300	mtrv1v						
	1050301	501	502	0.10	0.00			

*
* Single volume - downstream from 1ND002A
* 12-inch standard wall (schedule 40S)
*

		'VOL-110'		snglvol				
		AREA	LENGTH	VOL	AZIM	INCL	ELEV	ROUGH
HYDR	FLAGS							
	1100101	0.7854	3.0	0.00	0.00	0.00	0.00	1.5e-4
0.00	00000							
		* FLAGS		PRESS	TEMP			
	1100200	003	50.0	100.				

*
* Cross flow junction - from RHR piping to relief valve piping
*

		'TEE'		sngljun				
		FROM	TO	AREA	FLOSS	RLOSS	FLAGS	
	1140101	110000000	190000000	0.0884	1.02	1.02	001002	
	1140201	1	0.0	0.0	0.0			


```

*
*   Single junction - after tee, includes loss for straight pass
through tee
*
1150000 'Jun-115'          sngljun
*      FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1150101 110010000 120000000 0.785 0.26 0.26 001000
1150201 1          0.0      0.0      0.0

*
*   Pipe - from tee to RHR pump 1A suction
*   PS-601.2 - 12 and 14-inch standard wall
*
1200000 'P120'            pipe
1200001 24                * number of volumes
1200101 0.7854 12          * volume flow area
1200102 0.9575 13          * volume flow area
1200103 0.9575 24          * volume flow area
1200301 3.95 24            * " length
1200601 0.0 24             * " vertical

orientation
1200801 1.5e-4 0.0 24      * " friction data
1200901 0.195 0.195 1      * pipe junction loss

coefficients
1200902 0.182 0.182 2      * 90 deg bend
1200903 0.0 0.0 3          *
1200904 0.0 0.0 4          *
1200905 0.0 0.0 5          *
1200906 0.0 0.0 6          *
1200907 0.0 0.0 7          *
1200908 0.0 0.0 8          *
1200909 0.182 0.182 9      * 90 deg bend
1200910 0.0 0.0 10         *
1200911 0.0 0.0 11         *
1200912 0.214 0.215 12     * 90 deg bend + expansion
1200913 0.0 0.0 13         *
1200914 0.0 0.0 14         *
1200915 0.0 0.0 15         *
1200916 0.182 0.182 16     * 90 deg bend
1200917 0.0 0.0 17         *
1200918 0.182 0.182 18     * 90 deg bend
1200919 0.0 0.0 19         *
1200920 0.0 0.0 20         *
1200921 0.780 0.780 21     * Tee - flow through

branch
1200922 0.182 0.182 22     * 90 deg bend
1200923 0.182 0.182 23     * 90 deg bend
1201001 00 24              * volume control flags
1201101 01000 23           * junction control flags
1201201 003 50.0 100.0 0. 0. 0. 24
1201301 0.0 0.0 0.0 23

```

```

*
*   Single junction - RHR pump 1A
*
1250000 'ND1A'          sngljun
*      FROM          TO      AREA      FLOSS      RLOSS      FLAGS
1250101 120010000 130000000 0.3601 10.0      10.0      001000
1250201 1          0.0      0.0      0.0

*
*   Pipe - RHR pump 1A discharge to cross connect to 1B discharge
*   PS-601.2 - 8 inch schedule 20
*
1300000 'VOL-130'      pipe
1300001 85              * number of volumes
1300101 0.3601 85        * volume flow area
1300201 0.3601 1         * junction flow areas
1300202 0.3601 71        *
1300203 0.3601 72        * FE 5040
1300204 0.3601 73        *
1300205 0.3601 84        *
1300301 4.00 85          * volume length
1300601 0.0 85           * "      vertical

orientation
1300801 1.5e-4 0.0 85    * "      friction data
1300901 0.196 0.196 1    * pipe junction loss

coefficients
1300902 1.708 1.708 2    * 90 deg bend, check

valve, MOGV
1300903 0.0 0.0 3        *
1300904 0.196 0.196 4    * 90 deg bend
1300905 0.0 0.0 5        *
1300906 0.0 0.0 16       *
1300907 0.196 0.196 17    * 90 deg bend
1300908 0.0 0.0 18       *
1300909 0.196 0.196 19    * 90 deg bend
1300910 0.0 0.0 20       *
1300911 0.196 0.196 28    * 90 deg bend
1300912 0.0 0.0 29       * 90 deg bend + expansion
1300913 0.196 0.196 30    *
1300914 0.0 0.0 31       *
1300915 0.0 0.0 32       *
1300916 0.196 0.196 33    * 90 deg bend
1300917 0.0 0.0 34       *
1300918 0.0 0.0 37       *
1300919 0.196 0.196 38    * 90 deg bend
1300920 0.0 0.0 39       *
1300921 0.196 0.196 40    * 90 deg bend
1300922 0.0 0.0 41       *
1300923 0.0 0.0 47       *
1300924 0.196 0.196 48    * 90 deg bend
1300925 0.0 0.0 49       *
1300926 0.0 0.0 57       *

```

1300927	0.196	0.196	58							* 90 deg bend
1300928	0.0	0.0	59							*
1300929	0.0	0.0	64							*
1300930	0.196	0.196	65							* 90 deg bend
1300931	0.0	0.0	66							*
1300932	0.196	0.196	67							* 45 deg bend
1300933	0.196	0.196	68							* 90 deg bend
1300934	0.0	0.0	69							*
1300935	0.0	0.0	71							*
1300936	8.62	8.62	72							* FE 5040
1300937	0.0	0.0	73							*
1300938	0.0	0.0	74							*
1300939	0.196	0.196	75							* 90 deg bend
1300940	0.196	0.196	76							* 90 deg bend
1300941	0.0	0.0	77							*
1300942	0.0	0.0	80							*
1300943	0.308	0.308	81							* MOGV 1ND019, 90 deg bend
1300944	0.196	0.196	82							* 90 deg bend
1300945	3.0	3.0	83							* 1A heat exchanger
1300946	1.148	1.148	84							* 90 deg bend, Tee, MOGV
1301001	00	85								* volume control flags
1301101	01000	84								* junction control flags
1301201	003	50.0	100.0	0.	0.	0.	0.	0.	85	
1301301	0.0	0.0	0.0	84						

*

* Single junction - Tee to downstream piping

*

1350000	'J135'		sngljun						
*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS			
1350101	130010000	140000000	0.180	0.840	0.840	001000			
1350201	1	0.0	0.0	0.0					

*

* Single junction - Tee, flow through branch

*

1360000	'JUN-136'		sngljun						
*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS			
1360101	130010000	200000000	0.180	0.84	0.84	001000			
1360201	1	0.0	0.0	0.0					

*

* Single junction - Tee, flow through branch

*

1370000	'JUN-137'		sngljun						
*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS			
1370101	140000000	200000000	0.180	0.84	0.84	001000			
1370201	1	0.0	0.0	0.0					

*

* Pipe - from branch at cold leg injection piping to relief

valve Tee

```

*
1400000 'P140'          pipe
1400001 30              * number of volumes
1400101 0.3601 30      * volume flow area
1400201 0.3601 1       * junction area
1400202 0.3601 12      *
1400203 0.3601 13      * FE 5190
1400204 0.3601 14      *
1400205 0.3601 29      *
1400301 4.1 30         * volume length
1400601 0.0 30         *          vertical

orientation
1400801 1.5e-4 0.0 30  *          friction data
1400901 0.196 0.196 1  * 90 deg bend
1400902 0.196 0.196 2  * 90 deg bend
1400903 0.0 0.0 3      *
1400904 0.0 0.0 4      *
1400905 0.196 0.196 5  * 90 deg bend
1400906 0.196 0.196 6  * 90 deg bend
1400907 0.196 0.196 7  * 90 deg bend
1400908 0.0 0.0 8      *
1400909 0.0 0.0 12     *
1400910 1.83 1.83 13   * FE 5190
1400911 0.0 0.0 14     *
1400912 0.0 0.0 19     *
1400913 0.196 0.196 20 * 90 deg bend
1400914 0.196 0.196 21 * 90 deg bend
1400915 0.196 0.196 22 * 90 deg bend
1400916 0.0 0.0 23     *
1400917 0.196 0.196 24 * 90 deg bend
1400918 0.196 0.196 25 * 90 deg bend
1400919 0.0 0.0 26     *
1400920 0.196 0.196 27 * 90 deg bend
1400921 0.0 0.0 28     *
1400922 0.0 0.0 29     *
1401001 00 30          *
1401101 001000 29      * junction control flags
1401201 003 50.0 100.0 0. 0. 0. 30
1401301 0.0 0.0 0.0 29

```

```

*
*   Single junction - Piping to single volume
*
1450000 'JUN-145'      sngljun
*      FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1450101 140010000 150000000 0.3601 0.0 0.0 001000
1450201 1 0.0 0.0 0.0
*
*   Single volume - at junction to 1ND031 piping
*
1500000 'VOL-150'      snglvol

```

	HYDR	0.00	00	1500101	0.3601	4.0	0.00	0.00	0.00	0.00	1.5e-4
				1500200	003	50.	100.				
				* Relief valve 1ND031							
1b/hr				* size = 4 in set pressure = 600 psi flow = 92000							
1b/sec				* bore = 1.99 in temperature = 400 F = 25.6							
				* accu = 60 psi blowdown = 30 psi							
				* 1540000 '1ND031' valve							
				* FROM TO AREA FLOSS RLOSS FLAGS							
				1540101	150010003	998000000	1.85e-3	0.00	0.00	000100	
				1540201	1	0.00	0.00	0.00			
				1540300	mtrv1v						
				1540301	505	506	1.00	0.00			
				* Single junction - Tee to downstream piping							
				* 1550000 'JUN-155' sngljun							
				* FROM TO AREA FLOSS RLOSS FLAGS							
				1550101	150010000	160000000	0.2532	1.018	1.018	001000	
				1550201	1	0.0	0.0	0.0			
				* Pipe - from relief valve tee to branch at cold leg injection							
piping				* 1600000 'JUN-160' pipe							
				1600001	6						* number of volumes
				1600101	0.2532	6					* volume flow area
				1600301	4.4	6					* length
				1600601	0.0	6					* vertical
orientation				1600801	1.5e-4	0.0	6				* friction data
				1600901	0.0	0.0	1				* pipe junction loss
coefficients				1600902	0.210	0.210	2				* 90 deg bend
				1600903	0.0	0.0	3				*
				1600904	0.0	0.0	4				*
				1600905	0.210	0.210	5				*
				1601001	00	6					* volume control flags
				1601101	001000	5					* junction control flags
				1601201	003	50.0	100.0	0.	0.	0.	6
				1601301	0.0	0.0	0.0	5			

```

*
*   Single junction - Tee to downstream piping
*
1650000 'JUN-165'          sngljun
*      FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1650101 160010000 170000000 0.1469 1.03      1.03      001000
1650201 1          0.0      0.0      0.0

```

```

*
*   Pipe - from branch to cold leg D isolation check valve
*

```

```

1700000 'P170'          pipe
1700001 6                * number of volumes
1700101 0.1469 6          * volume flow area
1700201 0.1469 1          * junction flow area
1700202 0.1469 4          *
1700203 0.1469 5          *
1700301 4.3      6          * volume length
1700601 0.0      6          *          vertical

orientation
1700801 1.5e-4 0.0      6          *          friction data
1700901 0.0      0.0      1          * pipe junction loss

coefficients
1700902 0.420    0.420    2          * 2 90 deg bends
1700903 0.210    0.210    3          * 90 deg bend
1700904 0.210    0.210    4          * 90 deg bend
1700905 12.8     12.8     5          * FE 5890
1701001 00       6          * volume control flags
1701101 01000    5          * junction control flags
1701201 003      50.      100.0  0.    0.      0.      6
1701301 0.0      0.0      0.0      5

```

```

*
*   Valve 1NI175 - first cold leg C isolation check valve
*

```

```

1750000 '1NI175'          valve
*      FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1750101 170000000 999000000 0.1469 1.50      1.50      001000
1750201 1          0.0      0.0      0.0
1750300 chkvlv
1750301 1          1          1.0      0.0

```

```

*
*   Pipe - from tee to RV 1ND003
*

```

```

1900000 'RV-PIPE'          pipe
1900001 10                * number of volumes
1900101 0.0884 10          * volume flow area
1900301 1.09      10          *          length
1900601 0.0      10          *          vertical

orientation

```

	1900801	1.5e-4	0.0	10	*	friction data	
	1900901	0.34	0.34	1	*	90 deg bend	
	1900902	0.0	0.0	2	*		
	1900903	0.34	0.34	3	*	90 deg bend	
	1900904	0.0	0.0	7	*		
	1900905	0.34	0.34	8	*	90 deg bend	
	1900906	0.0	0.0	9	*		
	1901001	00	10		*	volume control flags	
	1901101	01000	9		*	junction control flags	
	1901201	003	50.	100.0	0.	0. 0. 10	
	1901301	0.0	0.0	0.0	9		
	*						
	* Valve IND003 - RHR relief valve						
1b/hr	*	size = 4 in		set pressure = 450 psi	flow = 154935		
1b/sec	*	bore = 2.97 in		temperature = 400 F	= 43		
	*						
	1950000	'IND003'		valve			
	*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS
	1950101	190010000	998000000	3.94e-3	0.00	0.00	00100
	1950201	1	0.00	0.00	0.00		
	1950300	mtrv1v					
	1950301	503	504	1.00	0.00		
	*						
	* Pipe - from cross-connect to IND064						
	*						
	2000000	'P200'		pipe			
	2000001	44		* number of volumes			
	2000101	0.3601	44	* volume flow area			
	2000201	0.3601	1	* junction flow area			
	2000202	0.3601	37	*			
	2000203	0.3601	38	*			
	2000204	0.3601	39	*			
	2000205	0.3601	43	*			
	2000301	3.99	44	* volume length			
orientation	2000601	0.0	44	* vertical			
	2000801	1.5e-4	0.0	44	* friction data		
	2000901	0.112	0.112	1	* MOGV		
	2000902	0.0	0.0	2	*		
	2000903	0.196	0.196	3	* 90 deg bend		
	2000904	0.196	0.196	4	* 90 deg bend		
	2000905	0.0	0.0	5	*		
	2000906	0.196	0.196	6	* 90 deg bend		
	2000907	0.0	0.0	7	*		
	2000908	0.84	0.84	8	* Tee		
	2000909	0.596	0.596	9	* Exp, Contr, MOGV		
	2000910	0.0	0.0	10	*		
	2000911	1.68	1.68	11	* 2 Tees		

2000912	0.0	0.0	12	*
2000913	0.0	0.0	14	*
2000914	0.392	0.392	15	* 2 90 deg bends
2000915	0.0	0.0	16	*
2000916	0.0	0.0	17	*
2000917	0.196	0.196	18	* 90 deg bend
2000918	0.0	0.0	19	*
2000919	0.0	0.0	20	*
2000920	0.196	0.196	21	* 90 deg bend
2000921	0.0	0.0	22	*
2000922	0.196	0.196	23	* 90 deg bend
2000923	0.0	0.0	24	*
2000924	0.0	0.0	25	*
2000925	0.196	0.196	26	* 90 deg bend
2000926	0.0	0.0	27	*
2000927	0.196	0.196	28	* 90 deg bend
2000928	0.196	0.196	29	* 90 deg bend
2000929	0.0	0.0	30	*
2000930	0.0	0.0	37	*
2000931	1.83	1.83	38	* FE 5180
2000932	0.0	0.0	39	* 90 deg bend
2000933	0.392	0.392	40	* 2 45 deg bends
2000934	0.0	0.0	41	*
2000935	0.0	0.0	42	*
2000936	0.196	0.196	43	* 90 deg bend
2001001	00	44		* volume control flags
2001101	001000	43		* junction control flags
2001201	003	50.	100.0	0. 0. 44
2001301	0.0	0.0	0.0	43

*
 * Valve IND064 - RHR relief valve
 * size = 4 in set pressure = 600 psi flow = 92000
 * bore = 1.99 in temperature = 400 F = 25.6

lb/hr
 lb/s

*
 2050000 'IND064' valve
 * FROM TO AREA FLOSS RLOSS FLAGS
 2050101 200010000 998000000 1.85e-3 0.00 0.00 00100
 2050201 1 0.00 0.00 0.00
 2050300 mtrv1v
 2050301 507 508 1.00 0.00

*
 * Time-dependent volume - atmosphere boundary
 *

9980000	'ATMOS'	tmdpvol					
*	AREA	LENGTH	VOL	AZIM	INCL	ELEV	ROUGH
9980101	1.0e+3	1.0e+5	0.0	0.0	0.0	0.0	0.0
9980200	003						

HYDR
 0.0

FLAGS
 10


```

          9980201 0.0      14.7    215.
          *
          *   Time-dependent volume - cold leg boundary
          *
          9990000 'COLDLEG'      tmdpvol
          *          AREA      LENGTH  VOL      AZIM      INCL      ELEV      ROUGH
HYDR      FLAGS
0.0      10      9990101 1.0e+3  1.0e+5  0.0      0.0      0.0      0.0      0.0
          9990200 003
          9990201 0.0      2349.    562.
          *
          .END

```

COLD LEG ISOLATION FAILURE INPUT DECK LISTING

= RHR SYSTEM PIPING MODEL

*

* MISCELLANEOUS CONTROL CARDS

*

100 new transnt

101 run

102 british british

* ALLOC

105 1.0 2.0 180.0

*

* TIME STEP CONTROL CARDS

*

201 1. 1.0e-6 0.001 00002 10 5000 5000

202 2. 1.0e-6 0.01 00002 10 5000 5000

203 15. 1.0e-3 0.02 00002 5 5000 5000

*

* MINOR EDITS

*

312 p 150010000 * volume pressure

313 p 160010000 * " "

314 p 170010000 * " "

316 p 200010000 * " "

*

* TRIPS

*

* 501 Starts transient by opening motor operated valve

* 502 Keeps valve open for a long time

* 503 Opens relief valve IND003

* 504 Closes " " "

* 505 Opens relief valve IND031

* 506 Closes " " "

* 507 Opens relief valve IND064

* 508 Closes " " "

*

501 time 0 ge null 0 0.0 n

502 time 0 ge null 0 1.0e+6 n

505 p 150010000 ge null 0 600. n

506 p 150010000 le null 0 570. n

507 p 200440000 ge null 0 600. n

508 p 200440000 le null 0 570. n

*

* Single junction - Tee, flow through branch

*

1370000 'JUN-137' sngljun

* FROM TO AREA FLOSS RLOSS FLAGS

1370101 140000000 200000000 0.3601 0.84 0.84 001000

1370201 1 0.0 0.0 0.0

*

* Pipe - from branch at cold leg injection piping to relief valve Tee

*

1400000	'P140'		pipe						
1400001	30								* number of volumes
1400101	0.3601	30							* volume flow area
1400201	0.3601	1							* junction area
1400202	0.3601	12							*
1400203	0.3601	13							* FE 5190
1400204	0.3601	14							*
1400205	0.3601	29							*
1400301	4.1	30							* volume length
1400601	0.0	30							* vertical orientation
1400801	1.5e-4	0.0	30						* friction data
1400901	0.196	0.196	1						* 90 deg bend
1400902	0.196	0.196	2						* 90 deg bend
1400903	0.0	0.0	3						*
1400904	0.0	0.0	4						*
1400905	0.196	0.196	5						* 90 deg bend
1400906	0.196	0.196	6						* 90 deg bend
1400907	0.196	0.196	7						* 90 deg bend
1400908	0.0	0.0	8						*
1400909	0.0	0.0	12						*
1400910	1.83	1.83	13						* FE 5190
1400911	0.0	0.0	14						*
1400912	0.0	0.0	19						*
1400913	0.196	0.196	20						* 90 deg bend
1400914	0.196	0.196	21						* 90 deg bend
1400915	0.196	0.196	22						* 90 deg bend
1400916	0.0	0.0	23						*
1400917	0.196	0.196	24						* 90 deg bend
1400918	0.196	0.196	25						* 90 deg bend
1400919	0.0	0.0	26						*
1400920	0.196	0.196	27						* 90 deg bend
1400921	0.0	0.0	28						*
1400922	0.0	0.0	29						*
1401001	00000	30							* volume flags - pvbfe
1401101	001000	29							* junction flags - fvcahs
1401201	003	50.0	100.0	0.	0.	0.	0.	30	
1401301	0.0	0.0	0.0	29					

*

* Single junction - Piping to single volume

*

1450000	'JUN-145'		sngljun						
*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS			
1450101	140010000	150000000	0.3601	0.0	0.0	001000			
1450201	1	0.0	0.0	0.0					

*

* Single volume - at junction to 1ND031 piping

*

1500000	'VOL-150'		snglvol						
*	AREA	LENGTH	VOL	AZIM	INCL	ELEV	ROUGH	HYDR	FLAGS
1500101	0.3601	4.0	0.00	0.00	0.00	0.00	1.5e-4	0.00	00000
*	FLAGS	PRESS	TEMP						
1500200	003	50.	100.						

```

*
*   Relief valve 1ND031
*       size = 4 in      set pressure = 600 psi      flow = 92000 lb/hr
*       bore = 1.99 in   temperature = 400 F        = 25.6 lb/sec
*       accu = 60 psi     blowdown = 30 psi
*
*
1540000 '1ND031'      valve
*       FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1540101 150010003  998000000  3.78e-3  0.00      0.00      000100
1540201 1          0.00      0.00      0.00
1540300 mtrvlv
1540301 505        506        1.00      0.00
*
*   Single junction - Tee to downstream piping
*
1550000 'JUN-155'      sngljun
*       FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1550101 150010000  160000000  0.2532  1.018     1.018     001000
1550201 1          0.0       0.0       0.0
*
*   Pipe - from relief valve Tee to branch at cold leg injection piping
*
1600000 'JUN-160'      pipe
1600001 6              * number of volumes
1600101 0.2532 6        * volume flow area
1600301 4.4 6          * length
1600601 0.0 6          * vertical orientation
1600801 1.5e-4 0.0 6    * friction data
1600901 0.0 0.0 1      * pipe junction loss coefficients
1600902 0.210 0.210 2  * 90 deg bend
1600903 0.0 0.0 3
1600904 0.0 0.0 4
1600905 0.210 0.210 5
1601001 00000 6        * volume flags - pvbfe
1601101 001000 5       * junction flags - fvcahs
1601201 003      50.0    100.0  0.    0.    0.    6
1601301 0.0      0.0     0.0     5
*
*   Single junction - Tee to downstream piping
*
1650000 'JUN-165'      sngljun
*       FROM      TO      AREA      FLOSS      RLOSS      FLAGS
1650101 160010000  170000000  0.1469  1.03      1.03      001000
1650201 1          0.0       0.0       0.0
*
*   Pipe - from branch to cold leg D isolation check valve
*
1700000 'P170'          pipe
1700001 6              * number of volumes
1700101 0.1469 6        * volume flow area
1700201 0.1469 1        * junction flow area

```

1700202	0.1469	4					*
1700203	0.1469	5					*
1700301	4.3	6					* volume length
1700601	0.0	6					* vertical orientation
1700801	1.5e-4	0.0	6				* friction data
1700901	0.0	0.0	1				* pipe junction loss coefficients
1700902	0.420	0.420	2				* 2 90 deg bends
1700903	0.210	0.210	3				* 90 deg bend
1700904	0.210	0.210	4				* 90 deg bend
1700905	12.8	12.8	5				* FE 5890
1701001	00000	6					* volume flags - pvbfe
1701101	001000	5					* junction flags - fvcchs
1701201	003	50.	100.0	0.	0.	0.	0. 0. 6
1701301	0.0	0.0	0.0	5			

*

* Valve INI175 - first coldleg C isolation check valve

*

1750000	'INI175'		valve				
*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS	
1750101	170010000	999000000	0.1469	0.00	0.00	001100	
1750201	1	0.0	0.0	0.0			
1750300	mtrv1v						
1750301	501	502	0.1	0.0			

*

* Pipe - from cross connect to IND064

*

2000000	'PIPE'		pipe				
2000001	44						* number of volumes
2000101	0.3601	44					* volume flow area
2000201	0.3601	1					* junction flow area
2000202	0.3601	37					*
2000203	0.3601	38					*
2000204	0.3601	39					*
2000205	0.3601	43					*
2000301	3.99	44					* volume length
2000601	0.0	44					* vertical orientation
2000801	1.5e-4	0.0	44				* friction data
2000901	0.112	0.112	1				* MOGV
2000902	0.0	0.0	2				*
2000903	0.196	0.196	3				* 90 deg bend
2000904	0.196	0.196	4				* 90 deg bend
2000905	0.0	0.0	5				*
2000906	0.196	0.196	6				* 90 deg bend
2000907	0.0	0.0	7				*
2000908	0.84	0.84	8				* Tee
2000909	0.596	0.596	9				* Exp, Contr, MOGV
2000910	0.0	0.0	10				*
2000911	1.68	1.68	11				* 2 Tees
2000912	0.0	0.0	12				*
2000913	0.0	0.0	14				*
2000914	0.392	0.392	15				* 2 90 deg bends
2000915	0.0	0.0	16				*

```

2000916 0.0      0.0      17      *
2000917 0.196    0.196    18      * 90 deg bend
2000918 0.0      0.0      19      *
2000919 0.0      0.0      20      *
2000920 0.196    0.196    21      * 90 deg bend
2000921 0.0      0.0      22      *
2000922 0.196    0.196    23      * 90 deg bend
2000923 0.0      0.0      24      *
2000924 0.0      0.0      25      *
2000925 0.196    0.196    26      * 90 deg bend
2000926 0.0      0.0      27      *
2000927 0.196    0.196    28      * 90 deg bend
2000928 0.196    0.196    29      * 90 deg bend
2000929 0.0      0.0      30      *
2000930 0.0      0.0      37      *
2000931 1.83     1.83     38      * FE 5180
2000932 0.0      0.0      39      * 90 deg bend
2000933 0.392    0.392    40      * 2 45 deg bends
2000934 0.0      0.0      41      *
2000935 0.0      0.0      42      *
2000936 0.196    0.196    43      * 90 deg bend
2001001 00000    44      * volume control flags - pvbfe
2001101 001000   43      * junction control flags - fvcchs
2001201 003      50.      100.0   0.      0.      0.      44
2001301 0.0      0.0      0.0      43
*
*   Valve IND064 - RHR relief valve
*   size = 4 in      set pressure = 600 psi      flow = 92000 lb/hr
*   bore = 1.99 in   temperature = 400 F         = 25.6 lb/s
*
2050000 'IND064'      valve
*   FROM      TO      AREA      FLOSS      RLOSS      FLAGS
2050101 200010000 997000000 3.78e-3 0.00      0.00      000100
2050201 1          0.00      0.00      0.00
2050300 mtrv1v
2050301 507      508      1.00      0.00
*
*   Time-dependent volume - atmosphere boundary
*
9970000 'ATMOS'      tmdpvol
*   AREA      LENGTH  VOL      AZIM      INCL      ELEV      ROUGH      HYDR      FLAGS
9970101 1.0e+3    1.0e+5    0.0      0.0      0.0      0.0      0.0      0.0      00010
9970200 003
9970201 0.0      14.7      215.
*
*   Time-dependent volume - atmosphere boundary
*
9980000 'ATMOS'      tmdpvol
*   AREA      LENGTH  VOL      AZIM      INCL      ELEV      ROUGH      HYDR      FLAGS
9980101 1.0e+3    1.0e+5    0.0      0.0      0.0      0.0      0.0      0.0      00010
9980200 003
9980201 0.0      14.7      215.

```

```

*
*   Time-dependent volume - cold leg boundary
*
9990000 'COLDLEG'      tmdpvol
*      AREA    LENGTH  VOL      AZIM    INCL    ELEV    ROUGH    HYDR    FLAGS
9990101 1.0e+3  1.0e+5  0.0     0.0     0.0     0.0     0.0     0.0     00010
9990200 003
9990201 0.0      2349.  562.
*
.END

```

APPENDIX 1

RELAP5 OUTPUT LISTINGS

RELAP5 PARTIAL OUTPUT LISTING

RHR System Cold Leg Isolation Failure Results

(Using 585 gpm as a basis for calculating IND031 and IND064 capacities)

1RBIC/MOD 3/ V5m5 RELAP5 Based Integrated Code

0Execute file: /usr/tmp/gls/rbic/5m5.x

0Effective command line parameters:

-f ftbl	-i nd
-o nd.out	-p plotfl
-r rstplt	-s stripf
-w /usr/local/selap/sth2xt	-d std2ot
-h sthyxt	-j jbinfo
-n null	-c cdffile

0Table of Update Define Values

athena	off	apollo	off	blkdta	off	bufr	on	cdc	off
cdccra	on	cos	off	cray	on	ctime	off	ctss	off
cb205	off	fourbyt	off	hp	off	ibm	off	in32	off
imprn	on	mass	off	mmfld	on	npa	on	plots	off
selap	on	timed	off	unicos	on	unix	on	vax	off
doe	on								

Define Warnings (if any) follow:

NONE

listing of input data for case 1

```

1  = RHR SYSTEM PIPING MODEL
2  *
3  *  MISCELLANEOUS CONTROL CARDS
4  *
5  100      new      transnt
6  101      run
7  102      british british
8  *
9  105      1.0      2.0      180.0
10 *
11 *  TIME STEP CONTROL CARDS
12 *
13 201      2.      1.0e-6  0.01  00003  10      5000  5000
14 202      5.      1.0e-3  0.05  00003  5       5000  5000
15 203      200.    1.0e-3  0.10  00003  5       5000  5000
16 *
17 *  MINOR EDITS
18 *
19 312      p      150010000      * volume pressure
20 313      p      160010000      * " "
21 314      p      170010000      * " "
```

```

22 316      p      200430000      *      "      "
23 *
24 *      TRIPS
25 *
26 *      501      Starts transient by opening motor operated valve
27 *      502      Keeps valve open for a long time
28 *      503      Opens relief valve 1ND003
29 *      504      Closes " " "
30 *      505      Opens relief valve 1ND031
31 *      506      Closes " " "
32 *      507      Opens relief valve 1ND064
33 *      508      Closes " " "
34 *
35 501      time    0      ge      null    0      0.0      n
36 502      time    0      ge      null    0      1.0e+6  n
37 505      p      150010000 ge      null    0      600      n
38 506      p      150010000 le      null    0      570.     n
39 507      p      200440000 ge      null    0      600.     n
40 508      p      200440000 le      null    0      570.     n
41 *
42 *      Single junction - Tee, flow through branch
43 *
44 1370000 'JUN-137'      sngljun
45 *      FROM      TO      AREA      FLOSS      RLOSS      FLAGS
46 1370101 1400000000 2000000000 0.3601 0.84 0.84 001000
47 1370201 1      0.0      0.0      0.0
48 *
49 *      Pipe - from branch at cold leg injection piping to relief valve Tee
50 *
51 1400000 'P140'      pipe
52 1400001 30
53 1400101 0 3601 30
54 1400201 0 3601 1
55 1400202 0 3601 12
56 1400203 0 3601 13
57 1400204 0 3601 14
58 1400205 0 3601 29
59 1400301 4 1 30
60 1400601 0 0 30
61 1400801 1 5e-4 0.0 30
62 1400901 0 196 0.196 1
63 1400902 0 196 0.196 2
64 1400903 0 0 0.0 3
65 1400904 0.0 0.0 4
66 1400905 0.196 0.196 5
67 1400906 0.196 0.196 6
68 1400907 0.196 0.196 7
69 1400908 0.0 0.0 8
70 1400909 0.0 0.0 12
71 1400910 1.83 1.83 13

```

* number of volumes

* volume flow area

* junction area

* FE 5190

* volume length

* vertical orientation

* friction data

* 90 deg bend

* 90 deg bend

* 90 deg bend

* 90 deg bend

* FE 5190

```

72 1400911 0.0 0.0 14 *
73 1400912 0.0 0.0 19 *
74 1400913 0.196 0.196 20 * 90 deg bend
75 1400914 0.196 0.196 21 * 90 deg bend
76 1400915 0.196 0.196 22 * 90 deg bend
77 1400916 0.0 0.0 23 *
78 1400917 0.196 0.196 24 * 90 deg bend
79 1400918 0.196 0.196 25 * 90 deg bend
80 1400919 0.0 0.0 26 *
81 1400920 0.196 0.196 27 * 90 deg bend
82 1400921 0.0 0.0 28 *
83 1400922 0.0 0.0 29 *
84 1401001 00000 30 * volume flags - pvbfe
85 1401101 001000 29 * junction flags - fvcchs
86 1401201 003 50.0 100.0 0. 0. 0. 30
87 1401301 0.0 0.0 0.0 29
88 *
89 * Single junction - Piping to single volume
90 *
91 1450000 'JUN-145' sngljun
92 * FROM TO AREA FLOSS RLOSS FLAGS
93 1450101 140010000 1500000000 0.3601 0.0 0.0 001000
94 1450201 1 0.0 0.0 0.0
95 *
96 * Single volume - at junction to 1ND031 piping
97 *
98 1500000 'VOL-150' snglvol
99 * AREA LENGTH VOL AZIM INCL ELEV ROUGH HYDR FLAGS
100 1500101 0.3601 4.0 0.00 0.00 0.00 0.00 1.5e-4 0.00 00000
101 * FLAGS PRESS TEMP
102 1500200 003 50. 100.
103 *
104 * Relief valve 1ND031
105 * size = 4 in set pressure = 600 psi flow = 92000 lb/hr
106 * bore = 1.99 in temperature = 400 F = 25.6 lb/sec
107 * accu = 60 psi blowdown = 30 psi
108 *
109 *
110 1540000 '1ND031' valve
111 * FROM TO AREA FLOSS RLOSS FLAGS
112 1540101 150010003 9980000000 3.78e-3 0.00 0.00 000100
113 1540201 1 0.00 0.00 0.00
114 1540300 mtrvly
115 1540301 505 506 1.00 0.00
116 *
117 * Single junction - Tee to downstream piping
118 *
119 1550000 'JUN-155' sngljun
120 * FROM TO AREA FLOSS RLOSS FLAGS
121 1550101 150010000 1600000000 0.2532 1.018 1.018 001000

```

```

122 1550201 1      0.0    0.0    0.0
123 *
124 *   Pipe - from relief valve Tee to branch at cold leg injection piping
125 *
126 1600000 'JUN-160'      pipe
127 1600001 6              * number of volumes
128 1600101 0.2532 6        * volume flow area
129 1600301 4.4    6        *      length
130 1600601 0.0    6        *      vertical orientation
131 1600801 1.5e-4 0.0    6  *      friction data
132 1600901 0.0    0.0    1  * pipe junction loss coefficients
133 1600902 0.210  0.210  2  * 90 deg bend
134 1600903 0.0    0.0    3  *
135 1600904 0.0    0.0    4  *
136 1600905 0.210  0.210  5  *
137 1601001 00000 6        * volume flags - pvbfe
138 1601101 001000 5       * junction flags - fvcchs
139 1601201 003    50.0   100.0  0.  0.  0.  6
140 1601301 0.0    0.0    0 0    5
141 *
142 *   Single junction - Tee to downstream piping
143 *
144 1650000 'JUN-165'      sngljun
145 *      FROM      TO      AREA      FLOSS      RLOSS      FLAGS
146 1650101 160010000 170000000 0.1469 1.03    1 03    001000
147 1650201 1      0 0    0.0    0.0
148 *
149 *   Pipe - from branch to cold leg D isolation check valve
150 *
151 1700000 'P170'      pipe
152 1700001 6              * number of volumes
153 1700101 0.1469 6        * volume flow area
154 1700201 0.1469 1        * junction flow area
155 1700202 0.1469 4        *
156 1700203 0.1469 5        *
157 1700301 4.3    6        * volume length
158 1700601 0.0    6        *      vertical orientation
159 1700801 1.5e-4 0 0    6  *      friction data
160 1700901 0.0    0.0    1  * pipe junction loss coefficients
161 1700902 0.420  0.420  2  * 2 90 deg bends
162 1700903 0.210  0.210  3  * 90 deg bend
163 1700904 0.210  0.210  4  * 90 deg bend
164 1700905 12.8   12.8   5  * FE 5890
165 1701001 00000 6        * volume flags - pvbfe
166 1701101 001000 5       * junction flags - fvcchs
167 1701201 003    50.    100.0  0.  0.  0.  6
168 1701301 0.0    0.0    0.0  5
169 *
170 *   Valve 1NI175 - first coldleg C isolation check valve
171 *

```

172	1750000	'INI175'	valve				
173	*	FROM	TO	AREA	FLOSS	RLOSS	FLAGS
174	1750101	170010000	999000000	0 1469	0 00	0 00	001100
175	1750201	1	0 0	0 0			
176	1750300	mtrv1v					
177	1750301	501	502	0 1	0 0		
178	*						
179	*	Pipe - from cross connect to 1ND064					
180	*						
181	2000000	'PIPE'	pipe				
182	2000001	44					* number of volumes
183	2000101	0 3601	44				* volume flow area
184	2000201	0 3601	1				* junction flow area
185	2000202	0 3601	37				*
186	2000203	0 3601	38				*
187	2000204	0 3601	39				*
188	2000205	0 3601	43				*
189	2000301	3 99	44				* volume length
190	2000601	0 0	44				* vertical orientation
191	2000801	1 5e-4	0 0	44			* friction data
192	2000901	0 112	0 112	1			* MOGV
193	2000902	0 0	0 0	2			*
194	2000903	0 196	0 196	3			* 90 deg bend
195	2000904	0 196	0 196	4			* 90 deg bend
196	2000905	0 0	0 0	5			*
197	2000906	0 196	0 196	6			* 90 deg bend
198	2000907	0 0	0 0	7			*
199	2000908	0 84	0 84	8			* Tee
200	2000909	0 596	0 596	9			* Exp, Contr, MOGV
201	2000910	0 0	0 0	10			*
202	2000911	1 68	1 68	11			* 2 Tees
203	2000912	0 0	0 0	12			*
204	2000913	0 0	0 0	14			*
205	2000914	0 392	0 392	15			* 2 90 deg bends
206	2000915	0 0	0 0	16			*
207	2000916	0 0	0 0	17			*
208	2000917	0 196	0 196	18			* 90 deg bend
209	2000918	0 0	0 0	19			*
210	2000919	0 0	0 0	20			*
211	2000920	0 196	0 196	21			* 90 deg bend
212	2000921	0 0	0 0	22			*
213	2000922	0 196	0 196	23			* 90 deg bend
214	2000923	0 0	0 0	24			*
215	2000924	0 0	0 0	25			*
216	2000925	0 196	0 196	26			* 90 deg bend
217	2000926	0 0	0 0	27			*
218	2000927	0 196	0 196	28			* 90 deg bend
219	2000928	0 196	0 196	29			* 90 deg bend
220	2000929	0 0	0 0	30			*
221	2000930	0 0	0 0	37			*

```

222 2000931 1.83 1.83 38 * FE 5180
223 2000932 0.0 0.0 39 * 90 deg bend
224 2000933 0.392 0.392 40 * 2 45 deg bends
225 2000934 0.0 0.0 41 *
226 2000935 0.0 0.0 42 *
227 2000936 0.196 0.196 43 * 90 deg bend
228 2001001 00000 44 * volume control flags - pybfe
229 2001101 001000 43 * junction control flags - fvcchs
230 2001201 003 50. 100.0 0. 0. 0. 44
231 2001301 0.0 0.0 0.0 43
232 *
233 * Valve 1ND064 - RHR relief valve
234 * size = 4 in set pressure = 600 psi flow = 92000 lb/hr
235 * bore = 1.99 in temperature = 400 F = 25.6 lb/s
236 *
237 2050000 '1ND064' valve
238 * FROM TO AREA FLOSS RLOSS FLAGS
239 2050101 200010000 997000000 3.78e-3 0.00 0.00 000100
240 2050201 1 0.00 0.00 0.00
241 2050300 mtrvlv
242 2050301 507 508 1 00 0.00
243 *
244 * Time-dependent volume - atmosphere boundary
245 *
246 9970000 'ATMOS' tmdpvol
247 * AREA LENGTH VOL AZIM INCL ELEV ROUGH HYDR FLAGS
248 9970101 1.0e+3 1.0e+5 0.0 0.0 0.0 0.0 0.0 0.0 00010
249 9970200 003
250 9970201 0.0 14.7 215.
251 *
252 * Time-dependent volume - atmosphere boundary
253 *
254 9980000 'ATMOS' tmdpvol
255 * AREA LENGTH VOL AZIM INCL ELEV ROUGH HYDR FLAGS
256 9980101 1.0e+3 1.0e+5 0.0 0.0 0.0 0.0 0.0 0.0 00010
257 9980200 003
258 9980201 0.0 14.7 215.
259 *
260 * Time-dependent volume - cold leg boundary
261 *
262 9990000 'COLDLEG' tmdpvol
263 * AREA LENGTH VOL AZIM INCL ELEV ROUGH HYDR FLAGS
264 9990101 .1469 1.0e+6 0.0 0.0 0.0 0.0 0.0 0.0 00010
265 9990200 003
266 9990201 0.0 2349. 562.
267 *
268 .END

```

Note: Lines deleted from output listing at this point

```

0 _ _ INFO ---restart no      323 written, block no      2---
0Number of elements in sparse matrix 1 original = 266, factored = 266   roundoff error = 1 000000E-12   ncount = 324
1 time
(sec)      p      p      p      p
           150010000 160010000 170010000 200430000
           (1bf/1n2) (1bf/1n2) (1bf/1n2) (1bf/1n2)

0 000000    50 000    50 000    50 000    50 000
0 100000    113 16    108 13    71 538    475 16
0 200000    417 83    418 08    424 17    545 72
0 300000    811 61    812 93    821 59    716 85
0 400000    1078 2    1078 7    1082 8    1062 2
0 500000    1346 1    1346 7    1350 4    1340 0
0 600000    1559 3    1559 9    1563 6    1546 8
0 700000    1694 3    1694 9    1699 0    1681 7
0 800000    1764 0    1764 8    1770 0    1748 5
0 900000    1788 6    1789 6    1796 0    1769 6
1 00000    1788 7    1789 8    1797 3    1767 2
1 10000    1778 6    1779 8    1788 1    1755 7
1 20000    1765 4    1766 7    1775 7    1741 8
1 30000    1875 8    1875 4    1871 3    1856 7
1 40000    1960 8    1961 3    1964 0    1953 4
1 50000    2014 8    2015 3    2018 2    2010 9
1 60000    2053 5    2054 0    2057 1    2050 9
1 70000    2084 3    2084 9    2087 9    2082 3
1 80000    2109 9    2110 4    2113 4    2108 1
1 90000    2131 4    2131 9    2134 9    2129 6
2 00000    2149 7    2150 2    2153 1    2147 9
2 25000    2185 1    2185 6    2188 5    2183 4
2 50000    2210 6    2211 2    2213 9    2208 8
2 75000    2229 4    2229 9    2232 6    2227 7
3 00000    2243 2    2243 8    2246 4    2241 5
3 25000    2253 6    2254 1    2256 8    2251 9
3 50000    2261 6    2262 1    2264 8    2259 8
3 75000    2268 1    2268 6    2271 2    2266 3
4 00000    2273 8    2274 3    2276 8    2272 1
4 25000    2279 3    2279 7    2282 1    2277 5
4 50000    2284 5    2284 9    2287 2    2282 7
4 75000    2289 4    2289 8    2291 9    2287 6
5 00000    2293 7    2294 0    2296 1    2291 9
5 50000    2299 8    2300 2    2302 1    2298 0
6 00000    2304 0    2304 4    2306 3    2302 2
6 50000    2306 5    2306 9    2308 9    2304 7
7 00000    2308 1    2308 4    2310 4    2306 3
7 50000    2309 1    2309 4    2311 4    2307 2
8 00000    2309 7    2310 1    2312 0    2307 9
8 50000    2310 1    2310 5    2312 5    2308 3
9 00000    2310 4    2310 8    2312 7    2308 6

```


32.5000	2311.1	2311.5	2313.4	2309.5
33.0000	2311.1	2311.5	2313.4	2309.5
33.5000	2311.1	2311.5	2313.4	2309.5
34.0000	2311.1	2311.5	2313.4	2309.5
34.5000	2311.1	2311.5	2313.4	2309.5
35.0000	2311.1	2311.5	2313.4	2309.5
35.5000	2311.1	2311.5	2313.4	2309.5
36.0000	2311.2	2311.5	2313.5	2309.5
36.5000	2311.2	2311.5	2313.5	2309.6
37.0000	2311.2	2311.5	2313.5	2309.6
37.5000	2311.2	2311.5	2313.5	2309.6
38.0000	2311.2	2311.5	2313.5	2309.6
38.5000	2311.2	2311.5	2313.5	2309.6
1 time (sec)	p 150010000 (1bf/1n2)	p 160010000 (1bf/1n2)	p 170010000 (1bf/1n2)	p 200430000 (1bf/1n2)
39.0000	2311.2	2311.5	2313.5	2309.6
39.5000	2311.2	2311.5	2313.5	2309.6
40.0000	2311.2	2311.5	2313.5	2309.6
40.5000	2311.2	2311.5	2313.5	2309.6
41.0000	2311.2	2311.5	2313.5	2309.6
41.5000	2311.2	2311.5	2313.5	2309.6
42.0000	2311.2	2311.5	2313.5	2309.6
42.5000	2311.2	2311.5	2313.5	2309.6
43.0000	2311.2	2311.5	2313.5	2309.6
43.5000	2311.2	2311.5	2313.5	2309.7
44.0000	2311.2	2311.5	2313.5	2309.7
44.5000	2311.2	2311.5	2313.5	2309.7
45.0000	2311.2	2311.6	2313.5	2309.7
45.5000	2311.2	2311.6	2313.5	2309.7
46.0000	2311.2	2311.6	2313.5	2309.7
46.5000	2311.2	2311.6	2313.5	2309.7
47.0000	2311.2	2311.6	2313.5	2309.7
47.5000	2311.2	2311.6	2313.5	2309.7
48.0000	2311.2	2311.6	2313.5	2309.7
48.5000	2311.2	2311.6	2313.5	2309.7
49.0000	2311.2	2311.6	2313.5	2309.7
49.5000	2311.2	2311.6	2313.5	2309.7
50.0000	2311.2	2311.6	2313.5	2309.7
50.5000	2311.2	2311.6	2313.5	2309.8
51.0000	2311.2	2311.6	2313.5	2309.8
51.5000	2311.2	2311.6	2313.5	2309.8
52.0000	2311.2	2311.6	2313.5	2309.8
52.5000	2311.2	2311.6	2313.5	2309.8
53.0000	2311.2	2311.6	2313.5	2309.8
53.5000	2311.2	2311.6	2313.5	2309.8
54.0000	2311.2	2311.6	2313.5	2309.8
54.5000	2311.2	2311.6	2313.5	2309.8
55.0000	2311.2	2311.6	2313.5	2309.8

55.5000	2311.2	2311.6	2313.5	2309.8
56.0000	2311.2	2311.6	2313.5	2309.8
56.5000	2311.2	2311.6	2313.5	2309.8
57.0000	2311.2	2311.6	2313.5	2309.8
57.5000	2311.3	2311.6	2313.5	2309.8
58.0000	2311.3	2311.6	2313.5	2309.8
58.5000	2311.3	2311.6	2313.5	2309.9
59.0000	2311.3	2311.6	2313.5	2309.9
59.5000	2311.3	2311.6	2313.5	2309.9
60.0000	2311.3	2311.6	2313.5	2309.9
60.5000	2311.3	2311.6	2313.5	2309.9
61.0000	2311.3	2311.6	2313.5	2309.9
61.5000	2311.3	2311.6	2313.5	2309.9
62.0000	2311.3	2311.6	2313.5	2309.9
62.5000	2311.3	2311.6	2313.5	2309.9
63.0000	2311.3	2311.6	2313.5	2309.9
63.5000	2311.3	2311.6	2313.5	2309.9
1 time (sec)	p 150010000 (lbf/in2)	p 160010000 (lbf/in2)	p 170010000 (lbf/in2)	p 200430000 (lbf/in2)
64.0000	2311.3	2311.6	2313.5	2309.9
64.5000	2311.3	2311.6	2313.5	2309.9
65.0000	2311.2	2311.6	2313.5	2309.9
65.5000	2311.2	2311.6	2313.5	2309.9
66.0000	2311.2	2311.6	2313.5	2309.9
66.5000	2311.2	2311.6	2313.5	2309.9
67.0000	2311.2	2311.6	2313.5	2309.9
67.5000	2311.2	2311.6	2313.5	2309.9
68.0000	2311.2	2311.6	2313.5	2309.9
68.5000	2311.2	2311.6	2313.5	2309.9
69.0000	2311.2	2311.5	2313.5	2309.8
69.5000	2311.2	2311.5	2313.5	2309.8
70.0000	2311.2	2311.5	2313.5	2309.8
70.5000	2311.2	2311.5	2313.5	2309.8
71.0000	2311.1	2311.5	2313.4	2309.8
71.5000	2311.1	2311.5	2313.4	2309.8
72.0000	2311.1	2311.5	2313.4	2309.8
72.5000	2311.1	2311.4	2313.4	2309.7
73.0000	2311.1	2311.4	2313.4	2309.7
73.5000	2311.0	2311.4	2313.3	2309.7
74.0000	2311.0	2311.3	2313.3	2309.6
74.5000	2311.0	2311.3	2313.3	2309.6
75.0000	2310.9	2311.3	2313.2	2309.6
75.5000	2310.9	2311.2	2313.2	2309.5
76.0000	2310.8	2311.2	2313.2	2309.5
76.5000	2310.8	2311.1	2313.1	2309.4
77.0000	2310.7	2311.1	2313.1	2309.4
77.5000	2310.7	2311.0	2313.0	2309.3
78.0000	2310.6	2311.0	2313.0	2309.2

78 5000	2310 6	2310 9	2312 9	2309 2
79 0000	2310 5	2310 8	2312 8	2309 1
79 5000	2310 4	2310 8	2312 8	2309 0
80 0000	2310 3	2310 7	2312 7	2308 9
80 5000	2310 3	2310 6	2312 6	2308 8
81 0000	2310 2	2310 5	2312 5	2308 7
81 5000	2310 1	2310 4	2312 5	2308 6
82 0000	2310 0	2310 3	2312 4	2308 5
82 5000	2309 9	2310 2	2312 3	2308 4
83 0000	2309 8	2310 1	2312 2	2308 3
83 5000	2309 7	2310 0	2312 1	2308 2
84 0000	2309 6	2309 9	2312 0	2308 1
84 5000	2309 5	2309 8	2311 9	2308 0
85 0000	2309 3	2309 7	2311 8	2307 8
85 5000	2309 2	2309 6	2311 7	2307 7
86 0000	2309 1	2309 5	2311 5	2307 6
86 5000	2309 0	2309 4	2311 4	2307 5
87 0000	2308 9	2309 2	2311 3	2307 3
87 5000	2308 7	2309 1	2311 2	2307 2
88 0000	2308 6	2309 0	2311 1	2307 0
88 5000	2308 5	2308 9	2311 0	2306 9
1 t ime (sec)	p 150010000 (lbf/in ²)	p 160010000 (lbf/in ²)	p 170010000 (lbf/in ²)	p 200430000 (lbf/in ²)
89 0000	2308 4	2308 7	2310 8	2306 8
89 5000	2308 2	2308 6	2310 7	2306 7
90 0000	2308 1	2308 5	2310 6	2306 5
90 5000	2308 0	2308 4	2310 5	2306 4
91 0000	2307 9	2308 3	2310 4	2306 3
91 5000	2307 8	2308 1	2310 3	2306 1
92 0000	2307 7	2308 0	2310 2	2306 0
92 5000	2307 6	2307 9	2310 1	2305 9
93 0000	2307 4	2307 8	2310 0	2305 8
93 5000	2307 4	2307 7	2309 9	2305 7
94 0000	2307 3	2307 7	2309 8	2305 6
94 5000	2307 2	2307 6	2309 7	2305 5
95 0000	2307 1	2307 5	2309 7	2305 4
95 5000	2307 0	2307 4	2309 6	2305 4
96 0000	2307 0	2307 3	2309 5	2305 3
96 5000	2306 9	2307 3	2309 5	2305 2
97 0000	2306 8	2307 2	2309 4	2305 2
97 5000	2306 8	2307 2	2309 4	2305 1
98 0000	2306 7	2307 1	2309 3	2305 1
98 5000	2306 7	2307 1	2309 3	2305 0
99 0000	2306 6	2307 0	2309 2	2305 0
99 5000	2306 6	2307 0	2309 2	2305 0
100 000	2306 6	2307 0	2309 2	2304 9
100 500	2306 6	2307 0	2309 1	2304 9
101 000	2306 5	2306 9	2309 1	2304 9

101 500	2306 5	2306 9	2309 1	2304 8
102 000	2306 5	2306 9	2309 1	2304 8
102 500	2306 5	2306 9	2309 1	2304 8
103 000	2306 5	2306 9	2309 1	2304 8
103 500	2306 4	2306 8	2309 0	2304 8
104 000	2306 4	2306 8	2309 0	2304 8
104 500	2306 4	2306 8	2309 0	2304 8
105 000	2306 4	2306 8	2309 0	2304 8
105 500	2306 4	2306 8	2309 0	2304 8
106 000	2306 4	2306 8	2309 0	2304 7
106 500	2306 4	2306 8	2309 0	2304 7
107 000	2306 4	2306 8	2309 0	2304 7
107 500	2306 4	2306 8	2309 0	2304 7
108 000	2306 4	2306 8	2309 0	2304 7
108 500	2306 4	2306 8	2309 0	2304 7
109 000	2306 4	2306 8	2309 0	2304 7
109 500	2306 4	2306 8	2309 0	2304 7
110 000	2306 4	2306 8	2309 0	2304 7
110 500	2306 4	2306 8	2309 0	2304 7
111 000	2306 4	2306 8	2309 0	2304 7
111 500	2306 4	2306 8	2309 0	2304 7
112 000	2306 4	2306 8	2309 0	2304 7
112 500	2306 4	2306 8	2309 0	2304 7
113 000	2306 4	2306 8	2309 0	2304 7
113 500	2306 4	2306 8	2309 0	2304 7
1 time (sec)	p 150010000 (lbf/in2)	p 160010000 (lbf/in2)	p 170010000 (lbf/in2)	p 200430000 (lbf/in2)
114 000	2306 4	2306 8	2309 0	2304 7
114 500	2306 4	2306 8	2309 0	2304 7
115 000	2306 4	2306 8	2309 0	2304 7
115 500	2306 4	2306 8	2309 0	2304 7
116 000	2306 4	2306 8	2309 0	2304 7
116 500	2306 4	2306 8	2309 0	2304 7
117 000	2306 4	2306 8	2309 0	2304 7
117 500	2306 4	2306 8	2309 0	2304 7
118 000	2306 4	2306 8	2309 0	2304 7
118 500	2306 4	2306 8	2309 0	2304 7
119 000	2306 4	2306 8	2309 0	2304 7
119 500	2306 4	2306 8	2309 0	2304 7
120 000	2306 4	2306 8	2309 0	2304 7
120 500	2306 4	2306 8	2309 0	2304 7
121 000	2306 4	2306 8	2309 0	2304 7
121 500	2306 4	2306 8	2309 0	2304 7
122 000	2306 4	2306 8	2309 0	2304 7
122 500	2306 4	2306 8	2309 0	2304 7
123 000	2306 4	2306 8	2309 0	2304 7
123 500	2306 4	2306 8	2309 0	2304 7
124 000	2306 4	2306 8	2309 0	2304 7

124 500	2306 4	2306 8	2309 0	2304 7
125 000	2306 4	2306 8	2309 0	2304 7
125 500	2306 4	2306 8	2309 0	2304 7
126 000	2306 4	2306 8	2309 0	2304 7
126 500	2306 4	2306 8	2309 0	2304 7
127 000	2306 4	2306 8	2309 0	2304 7
127 500	2306 4	2306 8	2309 0	2304 7
128 000	2306 4	2306 8	2309 0	2304 7
128 500	2306 4	2306 8	2309 0	2304 7
129 000	2306 4	2306 8	2309 0	2304 7
129 500	2306 4	2306 8	2309 0	2304 7
130 000	2306 4	2306 8	2309 0	2304 7
130 500	2306 4	2306 8	2309 0	2304 7
131 000	2306 4	2306 8	2309 0	2304 7
131 500	2306 4	2306 8	2309 0	2304 7
132 000	2306 4	2306 8	2309 0	2304 7
132 500	2306 4	2306 8	2309 0	2304 7
133 000	2306 4	2306 8	2309 0	2304 7
133 500	2306 4	2306 8	2309 0	2304 7
134 000	2306 4	2306 8	2309 0	2304 7
134 500	2306 4	2306 8	2309 0	2304 7
135 000	2306 4	2306 8	2309 0	2304 7
135 500	2306 4	2306 8	2309 0	2304 7
136 000	2306 4	2306 8	2309 0	2304 7
136 500	2306 4	2306 8	2309 0	2304 7
137 000	2306 4	2306 8	2309 0	2304 7
137 500	2306 4	2306 8	2309 0	2304 7
138 000	2306 4	2306 8	2309 0	2304 7
138 500	2306 4	2306 8	2309 0	2304 7
1 time (sec)	p 150010000 (lbf/in ²)	p 160010000 (lbf/in ²)	p 170010000 (lbf/in ²)	p 200430000 (lbf/in ²)
139 000	2306 4	2306 8	2309 0	2304 7
139 500	2306 4	2306 8	2309 0	2304 7
140 000	2306 4	2306 8	2309 0	2304 7
140 500	2306 4	2306 8	2309 0	2304 7
141 000	2306 4	2306 8	2309 0	2304 7
141 500	2306 4	2306 8	2309 0	2304 7
142 000	2306 4	2306 8	2309 0	2304 7
142 500	2306 4	2306 8	2309 0	2304 7
143 000	2306 4	2306 8	2309 0	2304 7
143 500	2306 4	2306 8	2309 0	2304 7
144 000	2306 4	2306 8	2309 0	2304 7
144 500	2306 4	2306 8	2309 0	2304 7
145 000	2306 4	2306 8	2309 0	2304 7
145 500	2306 4	2306 8	2309 0	2304 7
146 000	2306 4	2306 8	2309 0	2304 7
146 500	2306 4	2306 8	2309 0	2304 7
147 000	2306 4	2306 8	2309 0	2304 7

147 500	2306 4	2306 8	2309 0	2304 7
148 000	2306 4	2306 8	2309 0	2304 7
148 500	2306 4	2306 8	2309 0	2304 7
149 000	2306 4	2306 8	2309 0	2304 7
149 500	2306 4	2306 8	2309 0	2304 7
150 000	2306 4	2306 8	2309 0	2304 7
150 500	2306 4	2306 8	2309 0	2304 7
151 000	2306 4	2306 8	2309 0	2304 7
151 500	2306 4	2306 8	2309 0	2304 7
152 000	2306 4	2306 8	2309 0	2304 7
152 500	2306 4	2306 8	2309 0	2304 7
153 000	2306 4	2306 8	2309 0	2304 7
153 500	2306 4	2306 8	2309 0	2304 7
154 000	2306 4	2306 8	2309 0	2304 7
154 500	2306 4	2306 8	2309 0	2304 7
155 000	2306 4	2306 8	2309 0	2304 7
155 500	2306 4	2306 8	2309 0	2304 7
156 000	2306 4	2306 8	2309 0	2304 7
156 500	2306 4	2306 8	2309 0	2304 7
157 000	2306 4	2306 8	2309 0	2304 7
157 500	2306 4	2306 8	2309 0	2304 7
158 000	2306 4	2306 8	2309 0	2304 7
158 500	2306 4	2306 8	2309 0	2304 7
159 000	2306 4	2306 8	2309 0	2304 7
159 500	2306 4	2306 8	2309 0	2304 7
160 000	2306 4	2306 8	2309 0	2304 7
160 500	2306 4	2306 8	2309 0	2304 7
161 000	2306 4	2306 8	2309 0	2304 7
161 500	2306 4	2306 8	2309 0	2304 7
162 000	2306 4	2306 8	2309 0	2304 7
162 500	2306 4	2306 8	2309 0	2304 7
163 000	2306 4	2306 8	2309 0	2304 7
163 500	2306 4	2306 8	2309 0	2304 7
1 time (sec)	p 150010000 (1bf/1n2)	p 160010000 (1bf/1n2)	p 170010000 (1bf/1n2)	p 200430000 (1bf/1n2)
164 000	2306 4	2306 8	2309 0	2304 7
164 500	2306 4	2306 8	2309 0	2304 7
165 000	2306 4	2306 8	2309 0	2304 7
165 500	2306 4	2306 8	2309 0	2304 7
166 000	2306 4	2306 8	2309 0	2304 7
166 500	2306 4	2306 8	2309 0	2304 7
167 000	2306 4	2306 8	2309 0	2304 7
167 500	2306 4	2306 8	2309 0	2304 7
168 000	2306 4	2306 8	2309 0	2304 7
168 500	2306 4	2306 8	2309 0	2304 7
169 000	2306 4	2306 8	2309 0	2304 7
169 500	2306 4	2306 8	2309 0	2304 7
170 000	2306 4	2306 8	2309 0	2304 7

170 500	2306 4	2306 8	2309 0	2304 7
171 000	2306 4	2306 8	2309 0	2304 7
171 500	2306 4	2306 8	2309 0	2304 7
172 000	2306 4	2306 8	2309 0	2304 7
172 500	2306 4	2306 8	2309 0	2304 7
173 000	2306 4	2306 8	2309 0	2304 7
173 500	2306 4	2306 8	2309 0	2304 7
174 000	2306 4	2306 8	2309 0	2304 7
174 500	2306 4	2306 8	2309 0	2304 7
175 000	2306 4	2306 8	2309 0	2304 7
175 500	2306 4	2306 8	2309 0	2304 7
176 000	2306 4	2306 8	2309 0	2304 7
176 500	2306 4	2306 8	2309 0	2304 7
177 000	2306 4	2306 8	2309 0	2304 7
177 500	2306 4	2306 8	2309 0	2304 7
178 000	2306 4	2306 8	2309 0	2304 7
178 500	2306 4	2306 8	2309 0	2304 7
179 000	2306 4	2306 8	2309 0	2304 7
179 500	2306 4	2306 8	2309 0	2304 7
180 000	2306 4	2306 8	2309 0	2304 7
180 500	2306 4	2306 8	2309 0	2304 7
181 000	2306 4	2306 8	2309 0	2304 7
181 500	2306 4	2306 8	2309 0	2304 7
182 000	2306 4	2306 8	2309 0	2304 7
182 500	2306 4	2306 8	2309 0	2304 7
183 000	2306 4	2306 8	2309 0	2304 7
183 500	2306 4	2306 8	2309 0	2304 7
184 000	2306 4	2306 8	2309 0	2304 7
184 500	2306 4	2306 8	2309 0	2304 7
185 000	2306 4	2306 8	2309 0	2304 7
185 500	2306 4	2306 8	2309 0	2304 7
186 000	2306 4	2306 8	2309 0	2304 7
186 500	2306 4	2306 8	2309 0	2304 7
187 000	2306 4	2306 8	2309 0	2304 7
187 500	2306 4	2306 8	2309 0	2304 7
188 000	2306 4	2306 8	2309 0	2304 7
188 500	2306 4	2306 8	2309 0	2304 7

1RELAP5/MOD3/V5m5 reactor loss of coolant analysis program

RHR SYSTEM PIPING MODEL

10/02/90 13 18 26

OMAJOR EDIT time= 200 000 sec

0 attempted adv	tot = 2273	edit= 1950	min dt= 0 100000
repeated adv	tot = 21	edit= 0	max dt= 0 100000
successful adv	tot = 2252	edit= 1950	avg dt= 0 100000
requested adv	tot = 2210	edit= 1950	req dt= 0 100000

sec	last dt= 0 100000	sec	ms red= 7 83812	lb
sec	crnt dt= 0 182575	sec	tot ms= 5508 41	lb
sec	err est= 4 431618E-14	sec	m rato= 1 058473E-03	
sec	cpu= 63 9456	sec	time= 200 000	sec

OTrip number, trip time (sec)

501 0 0000000 502 -1 000000 505 0 2500000

506 -1 000000 507 0 2200000

508 -1 000000

OSystem 1 *none* mass= 5508 4 lb mass error= 7 8381 lb err est = 4 43162E-14

200-010000	2305.8	1.0000	0.00000	0.00000	561.817	656.261	656.261	552.73	1048.4	00000	
200-020000	2305.8	1.0000	0.00000	0.00000	561.817	656.260	656.260	552.73	1048.4	00000	
200-030000	2305.8	1.0000	0.00000	0.00000	561.817	656.259	656.259	552.73	1048.4	00000	
200-040000	2305.8	1.0000	0.00000	0.00000	561.817	656.257	656.257	552.73	1048.4	00000	
200-050000	2305.7	1.0000	0.00000	0.00000	561.817	656.255	656.255	552.73	1048.4	00000	
200-060000	2305.7	1.0000	0.00000	0.00000	561.817	656.255	656.255	552.73	1048.4	00000	
200-070000	2305.7	1.0000	0.00000	0.00000	561.817	656.253	656.253	552.73	1048.4	00000	
200-080000	2305.7	1.0000	0.00000	0.00000	561.817	656.253	656.253	552.73	1048.4	00000	
200-090000	2305.6	1.0000	0.00000	0.00000	561.816	656.247	656.247	552.73	1048.4	00000	
200-100000	2305.5	1.0000	0.00000	0.00000	561.816	656.243	656.243	552.73	1048.4	00000	
200-110000	2305.5	1.0000	0.00000	0.00000	561.816	656.242	656.242	552.73	1048.4	00000	
200-120000	2305.4	1.0000	0.00000	0.00000	561.815	656.231	656.231	552.73	1048.4	00000	
200-130000	2305.4	1.0000	0.00000	0.00000	561.815	656.231	656.231	552.73	1048.4	00000	
200-140000	2305.3	1.0000	0.00000	0.00000	561.815	656.230	656.230	552.73	1048.4	00000	
200-150000	2305.3	1.0000	0.00000	0.00000	561.815	656.229	656.229	552.73	1048.4	00000	
200-160000	2305.3	1.0000	0.00000	0.00000	561.815	656.226	656.226	552.73	1048.4	00000	
200-170000	2305.3	1.0000	0.00000	0.00000	561.815	656.226	656.226	552.73	1048.4	00000	
200-180000	2305.3	1.0000	0.00000	0.00000	561.815	656.225	656.225	552.73	1048.4	00000	
200-190000	2305.2	1.0000	0.00000	0.00000	561.815	656.224	656.224	552.73	1048.4	00000	
200-200000	2305.2	1.0000	0.00000	0.00000	561.815	656.223	656.223	552.73	1048.4	00000	
200-210000	2305.2	1.0000	0.00000	0.00000	561.815	656.222	656.222	552.73	1048.4	00000	
200-220000	2305.2	1.0000	0.00000	0.00000	561.815	656.221	656.221	552.73	1048.4	00000	
200-230000	2305.2	1.0000	0.00000	0.00000	561.815	656.220	656.220	552.73	1048.4	00000	
200-240000	2305.2	1.0000	0.00000	0.00000	561.814	656.218	656.218	552.73	1048.4	00000	
200-250000	2305.2	1.0000	0.00000	0.00000	561.814	656.218	656.218	552.73	1048.4	00000	
200-260000	2305.1	1.0000	0.00000	0.00000	561.814	656.217	656.217	552.73	1048.4	00000	
200-270000	2305.1	1.0000	0.00000	0.00000	561.814	656.215	656.215	552.73	1048.4	00000	
200-280000	2305.1	1.0000	0.00000	0.00000	561.814	656.215	656.215	552.73	1048.4	00000	
200-290000	2305.1	1.0000	0.00000	0.00000	561.814	656.213	656.213	552.73	1048.4	00000	
200-300000	2305.0	1.0000	0.00000	0.00000	561.814	656.211	656.211	552.73	1048.4	00000	
200-310000	2305.0	1.0000	0.00000	0.00000	561.814	656.211	656.211	552.73	1048.4	00000	
200-320000	2305.0	1.0000	0.00000	0.00000	561.814	656.210	656.210	552.73	1048.4	00000	
200-330000	2305.0	1.0000	0.00000	0.00000	561.814	656.210	656.210	552.73	1048.4	00000	
200-340000	2305.0	1.0000	0.00000	0.00000	561.814	656.209	656.209	552.73	1048.4	00000	
200-350000	2305.0	1.0000	0.00000	0.00000	561.814	656.208	656.208	552.73	1048.4	00000	
200-360000	2305.0	1.0000	0.00000	0.00000	561.814	656.208	656.208	552.73	1048.4	00000	
200-370000	2305.0	1.0000	0.00000	0.00000	561.814	656.207	656.207	552.73	1048.4	00000	
200-380000	2305.0	1.0000	0.00000	0.00000	561.814	656.207	656.207	552.73	1048.4	00000	
200-390000	2304.8	1.0000	0.00000	0.00000	561.813	656.195	656.195	552.73	1048.4	00000	
200-400000	2304.8	1.0000	0.00000	0.00000	561.813	656.194	656.194	552.73	1048.4	00000	
200-410000	2304.7	1.0000	0.00000	0.00000	561.813	656.191	656.191	552.73	1048.4	00000	
200-420000	2304.7	1.0000	0.00000	0.00000	561.813	656.191	656.191	552.73	1048.4	00000	
200-430000	2304.7	1.0000	0.00000	0.00000	561.813	656.190	656.190	552.73	1048.4	00000	
200-440000	2304.7	1.0000	0.00000	0.00000	561.813	656.188	656.188	552.73	1048.4	00000	
0	Vol.no.	pressure	voidf	voidg	voidgo	tempf	tempg	sat. temp.	uf	ug	volume
		(psia)				(degf)	(degf)	(degf)	(btu/lb)	(btu/lb)	flag
ATMOS	tmdpvol										
997-010000	14.700	0.00000	1.0000	1.0000	212.014	215.000	212.014	180.13	1078.7	11010	
ATMOS	tmdpvol										
998-010000	14.700	0.00000	1.0000	1.0000	212.014	215.000	212.014	180.13	1078.7	11010	

COLDLEG tmdpvol											
999-010000	2349 0	1 0000	0 00000	0 00000	562 013	658 966	658 966	552 72	1044 9	11010	
0 Vol no	rhof	rhog	rho-mix	rho-boron	vel-liquid	vel-vapor	sounde	quality	quality	quality	
	(lb/ft3)	(lb/ft3)	(lb/ft3)	(lb/ft3)	(ft/sec)	(ft/sec)	(ft/sec)	mix-cup	static	non-cond	
140-010000	46 124	6 6363	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-020000	46 124	6 6364	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-030000	46 124	6 6365	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-040000	46 124	6 6366	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-050000	46 124	6 6366	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-060000	46 124	6 6368	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-070000	46 124	6 6369	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-080000	46 124	6 6370	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 360	0 000	0 000	
140-090000	46 124	6 6371	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 361	0 000	0 000	
140-100000	46 124	6 6371	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 361	0 000	0 000	
140-110000	46 124	6 6372	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 361	0 000	0 000	
140-120000	46 124	6 6372	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 361	0 000	0 000	
140-130000	46 124	6 6373	46 124	0 00000	-4 4142	-4 4142	3267 0	-0 361	0 000	0 000	
140-140000	46 124	6 6382	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-150000	46 124	6 6382	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-160000	46 124	6 6383	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-170000	46 124	6 6383	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-180000	46 124	6 6384	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-190000	46 124	6 6384	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-200000	46 124	6 6384	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-210000	46 124	6 6386	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-220000	46 124	6 6387	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-230000	46 124	6 6389	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-240000	46 124	6 6389	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-250000	46 124	6 6390	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-260000	46 124	6 6392	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-270000	46 124	6 6392	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-280000	46 124	6 6394	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-290000	46 124	6 6394	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
140-300000	46 124	6 6394	46 124	0 00000	-4 4142	-4 4142	3267 1	-0 361	0 000	0 000	
150-010000	46 124	6 6384	46 124	0 00000	-6 5585	-6 5588	3267 1	-0 361	0 000	0 000	
160-010000	46 124	6 6404	46 124	0 00000	-13 665	-13 665	3267 1	-0 361	0 000	0 000	
160-020000	46 124	6 6409	46 124	0 00000	-13 665	-13 665	3267 1	-0 361	0 000	0 000	
160-030000	46 124	6 6423	46 124	0 00000	-13 664	-13 664	3267 2	-0 361	0 000	0 000	
160-040000	46 125	6 6429	46 125	0 00000	-13 664	-13 664	3267 2	-0 361	0 000	0 000	
160-050000	46 125	6 6434	46 125	0 00000	-13 664	-13 664	3267 2	-0 361	0 000	0 000	
160-060000	46 125	6 6448	46 125	0 00000	-13 664	-13 664	3267 3	-0 361	0 000	0 000	
170-010000	46 125	6 6510	46 125	0 00000	-23 552	-23 552	3267 4	-0 362	0 000	0 000	
170-020000	46 126	6 6531	46 126	0 00000	-23 552	-23 552	3267 5	-0 362	0 000	0 000	
170-030000	46 126	6 6608	46 126	0 00000	-23 551	-23 551	3267 7	-0 363	0 000	0 000	
170-040000	46 127	6 6656	46 127	0 00000	-23 551	-23 551	3267 9	-0 363	0 000	0 000	
170-050000	46 127	6 6705	46 127	0 00000	-23 547	-23 547	3268 0	-0 364	0 000	0 000	
170-060000	46 143	6 8460	46 143	0 00000	-23 543	-23 543	3272 9	-0 382	0 000	0 000	
200-010000	46 124	6 6358	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000	
200-020000	46 124	6 6357	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000	
200-030000	46 124	6 6357	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000	

200-040000	46 124	6 6356	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-050000	46 124	6 6354	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-060000	46 124	6 6354	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-070000	46 124	6 6352	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-080000	46 124	6 6352	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-090000	46 124	6 6348	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-100000	46 124	6 6344	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-110000	46 124	6 6344	46 124	0 00000	4 4142	4 4142	3267 0	-0 360	0 000	0 000
200-120000	46 124	6 6335	46 124	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-130000	46 124	6 6335	46 124	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-140000	46 124	6 6335	46 124	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-150000	46 124	6 6334	46 124	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-160000	46 124	6 6332	46 124	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-170000	46 124	6 6331	46 124	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
0 Vol no	rhof (lb/ft3)	rhog (lb/ft3)	rho-mix (lb/ft3)	rho-boron (lb/ft3)	vel-liquid (ft/sec)	vel-vapor (ft/sec)	sounde (ft/sec)	quality mix-cup	quality static	quality non-cond
200-180000	46 123	6 6331	46 123	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-190000	46 123	6 6330	46 123	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-200000	46 123	6 6329	46 123	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-210000	46 123	6 6329	46 123	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-220000	46 123	6 6327	46 123	0 00000	4 4142	4 4142	3266 9	-0 360	0 000	0 000
200-230000	46 123	6 6327	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-240000	46 123	6 6326	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-250000	46 123	6 6325	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-260000	46 123	6 6325	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-270000	46 123	6 6323	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-280000	46 123	6 6323	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-290000	46 123	6 6322	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-300000	46 123	6 6320	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-310000	46 123	6 6320	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-320000	46 123	6 6319	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-330000	46 123	6 6319	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-340000	46 123	6 6319	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-350000	46 123	6 6318	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-360000	46 123	6 6318	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-370000	46 123	6 6317	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-380000	46 123	6 6317	46 123	0 00000	4 4143	4 4143	3266 9	-0 360	0 000	0 000
200-390000	46 123	6 6308	46 123	0 00000	4 4143	4 4143	3266 8	-0 360	0 000	0 000
200-400000	46 123	6 6307	46 123	0 00000	4 4143	4 4143	3266 8	-0 360	0 000	0 000
200-410000	46 123	6 6305	46 123	0 00000	4 4143	4 4143	3266 8	-0 360	0 000	0 000
200-420000	46 123	6 6305	46 123	0 00000	4 4143	4 4143	3266 8	-0 360	0 000	0 000
200-430000	46 123	6 6304	46 123	0 00000	4 4143	4 4143	3266 8	-0 360	0 000	0 000
200-440000	46 123	6 6303	46 123	0 00000	4 4143	4 4143	3266 8	-0 360	0 000	0 000
997-010000	59 813	3 71468E-02	3 71468E-02	0 00000	1 58958E-03	1 58958E-03	1556 8	0 322E-01	1 00	0 000
998-010000	59 813	3 71468E-02	3 71468E-02	0 00000	1 87033E-03	1 87033E-03	1556 8	0 275E-01	1 00	0 000
999-010000	46 143	6 8470	46 143	0 00000	-23 543	-23 543	3273 0	-0 382	0 000	0 000
0 vol no	tot ht inp (btu/sec)	vap ht inp (btu/sec)	vapor-gen (lb/sec-ft3)	wall-flashing (lb/sec-ft3)	liq int htc (b/s-ft3-f)	vap int htc (b/s-ft3-f)	mass-flux (lb/sec-ft2)	Reynolds liquid	Reynolds vapor	flow reg1
140-010000	0 00000	0 00000	8 68588E-14	0 00000	1 49066E-05	498 17	-203 58	-2 18922E+06-1	18811E-09	bby
140-020000	0 00000	0 00000	-1 62719E-13	0 00000	1 49066E-05	498 17	-203 58	-2 18922E+06-1	18812E-09	bby

140-030000	0.00000	0.00000	-4.82236E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18812E-09	bby
140-040000	0.00000	0.00000	4.88616E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18813E-09	bby
140-050000	0.00000	0.00000	1.09969E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18814E-09	bby
140-060000	0.00000	0.00000	2.67833E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18815E-09	bby
140-070000	0.00000	0.00000	1.18079E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18816E-09	bby
140-080000	0.00000	0.00000	-1.41751E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18817E-09	bby
140-090000	0.00000	0.00000	4.94719E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18817E-09	bby
140-100000	0.00000	0.00000	1.08568E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18818E-09	bby
140-110000	0.00000	0.00000	-2.34006E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18922E+06-1.18818E-09	bby
140-120000	0.00000	0.00000	1.76338E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18818E-09	bby
140-130000	0.00000	0.00000	2.92710E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18826E-09	bby
140-140000	0.00000	0.00000	-1.40835E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18827E-09	bby
140-150000	0.00000	0.00000	-1.43144E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18827E-09	bby
140-160000	0.00000	0.00000	-4.10526E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18828E-09	bby
140-170000	0.00000	0.00000	-1.98465E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18828E-09	bby
140-180000	0.00000	0.00000	3.32191E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18829E-09	bby
140-190000	0.00000	0.00000	-2.01533E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18829E-09	bby
140-200000	0.00000	0.00000	2.93183E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18830E-09	bby
140-210000	0.00000	0.00000	-2.74720E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18831E-09	bby
140-220000	0.00000	0.00000	2.14791E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18833E-09	bby
140-230000	0.00000	0.00000	1.78104E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18833E-09	bby
140-240000	0.00000	0.00000	-7.65768E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18834E-09	bby
140-250000	0.00000	0.00000	4.55940E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18921E+06-1.18836E-09	bby
140-260000	0.00000	0.00000	-2.72039E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18920E+06-1.18836E-09	bby
140-270000	0.00000	0.00000	-1.16395E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18920E+06-1.18837E-09	bby
140-280000	0.00000	0.00000	-7.83192E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18920E+06-1.18838E-09	bby
140-290000	0.00000	0.00000	-1.47253E-13	0.00000	1.49066E-05	498.17	-203.58	-2.18920E+06-1.18838E-09	bby
140-300000	0.00000	0.00000	7.14154E-14	0.00000	1.49066E-05	498.17	-203.58	-2.18920E+06-1.18829E-09	bby
150-010000	0.00000	0.00000	2.80819E-13	0.00000	1.49066E-05	498.17	-323.36	-3.47716E+06-1.88775E-09	bby
160-010000	0.00000	0.00000	-4.32464E-13	0.00000	1.49066E-05	498.17	-630.22	-5.68265E+06-3.08510E-09	bby
160-020000	0.00000	0.00000	-1.34704E-12	0.00000	1.49066E-05	498.17	-630.22	-5.68264E+06-3.08543E-09	bby
160-030000	0.00000	0.00000	-6.79393E-13	0.00000	1.49066E-05	498.17	-630.22	-5.68262E+06-3.08556E-09	bby
160-040000	0.00000	0.00000	-5.15384E-13	0.00000	1.49066E-05	498.17	-630.22	-5.68261E+06-3.08567E-09	bby
160-050000	0.00000	0.00000	4.89774E-13	0.00000	1.49066E-05	498.17	-630.22	-5.68260E+06-3.08601E-09	bby
160-060000	0.00000	0.00000	-5.11094E-13	0.00000	1.49066E-05	498.17	-630.22	-5.68258E+06-3.08744E-09	bby
170-010000	0.00000	0.00000	4.19235E-13	0.00000	1.49066E-05	498.17	-1086.3	-7.46038E+06-4.05406E-09	bby
170-020000	0.00000	0.00000	-1.64023E-13	0.00000	1.49066E-05	498.17	-1086.3	-7.46033E+06-4.05638E-09	bby
170-030000	0.00000	0.00000	1.40480E-13	0.00000	1.49066E-05	498.17	-1086.3	-7.46020E+06-4.05790E-09	bby
170-040000	0.00000	0.00000	-1.55781E-13	0.00000	1.49066E-05	498.17	-1086.3	-7.46011E+06-4.05940E-09	bby
170-050000	0.00000	0.00000	1.03456E-12	0.00000	1.49066E-05	498.17	-1086.3	-7.46002E+06-4.11214E-09	bby
170-060000	0.00000	0.00000	1.49532E-13	0.00000	1.49066E-05	498.17	-1086.3	-7.45743E+06-4.11346E-09	bby
200-010000	0.00000	0.00000	-1.67172E-13	0.00000	1.49066E-05	498.17	203.58	2.18922E+06 1.18809E-09	bby
200-020000	0.00000	0.00000	-1.23931E-13	0.00000	1.49066E-05	498.17	203.58	2.18922E+06 1.18805E-09	bby
200-030000	0.00000	0.00000	9.92833E-14	0.00000	1.49066E-05	498.17	203.58	2.18922E+06 1.18804E-09	bby
200-040000	0.00000	0.00000	9.17563E-14	0.00000	1.49066E-05	498.17	203.58	2.18922E+06 1.18804E-09	bby
200-050000	0.00000	0.00000	-2.20656E-13	0.00000	1.49066E-05	498.17	203.58	2.18922E+06 1.18803E-09	bby
200-060000	0.00000	0.00000	-9.55154E-14	0.00000	1.49066E-05	498.17	203.58	2.18922E+06 1.18802E-09	bby
200-070000	0.00000	0.00000	-3.52909E-13	0.00000	1.49066E-05	498.17	203.58	2.18923E+06 1.18801E-09	bby
200-080000	0.00000	0.00000	-1.55298E-13	0.00000	1.49066E-05	498.17	203.58	2.18923E+06 1.18800E-09	bby
200-090000	0.00000	0.00000	-2.19103E-13	0.00000	1.49066E-05	498.17	203.58	2.18923E+06 1.18800E-09	bby

140-090000	14	14	0	0	0	0	0	0	0	0	0	0	0	0
140-100000	11	11	0	0	0	0	0	0	0	0	0	0	0	0
140-110000	6	6	0	0	0	0	0	0	0	0	0	0	0	0
140-120000	13	13	0	0	0	0	0	0	0	0	0	0	0	0
140-130000	17	17	0	0	0	0	0	0	0	0	0	0	0	0
140-140000	16	16	0	0	0	0	0	0	0	0	0	0	0	0
140-150000	16	16	0	0	0	0	0	0	0	0	0	0	0	0
140-160000	18	18	0	0	0	0	0	0	0	0	1	0	0	0
140-170000	10	10	0	0	0	0	0	0	0	0	0	0	0	0
140-180000	9	9	0	0	0	0	0	0	0	0	0	0	0	0
140-190000	10	10	0	0	0	0	0	0	0	0	0	0	0	0
140-200000	11	11	0	0	0	0	0	0	0	0	0	0	0	0
140-210000	12	12	0	0	0	0	0	0	0	0	0	0	0	0
140-220000	14	14	0	0	0	0	0	0	0	0	1	0	0	0
140-230000	13	13	0	0	0	0	0	0	0	0	0	0	0	0
140-240000	7	7	0	0	0	0	0	0	0	0	0	0	0	0
140-250000	11	11	0	0	0	0	0	0	0	0	1	0	0	0
140-260000	16	17	0	0	0	0	0	0	0	0	0	0	0	0
140-270000	13	13	0	0	0	0	0	0	0	0	0	0	0	0
140-280000	12	12	0	0	0	0	0	0	0	0	0	0	0	0
140-290000	11	12	0	0	0	0	0	0	0	0	0	0	0	0
140-300000	10	14	0	0	0	0	0	0	0	0	0	0	0	0
150-010000	0	11	0	0	0	0	0	0	0	0	0	0	0	0
160-010000	0	11	0	0	0	0	0	0	0	0	2	0	0	0
160-020000	0	7	0	0	0	0	0	0	0	0	1	0	0	0
160-030000	0	10	0	0	0	0	0	0	0	0	0	0	0	0
160-040000	0	5	0	0	0	0	0	0	0	0	0	0	0	0
160-050000	0	8	0	0	0	0	0	0	0	0	0	0	0	0
160-060000	0	15	0	0	0	0	0	0	0	0	0	0	0	0
170-010000	0	22	0	0	0	0	0	0	0	1938	1982	0	0	0
170-020000	0	22	0	0	0	0	0	0	0	1	2	0	0	0
170-030000	0	32	0	0	0	0	0	0	0	0	1	0	0	0
170-040000	0	33	0	0	0	0	0	0	3	11	64	0	0	0
170-050000	0	48	0	0	0	0	0	0	3	0	54	0	0	0
170-060000	0	72	0	0	0	0	0	0	4	0	137	0	0	0
200-010000	11	11	0	0	0	0	0	0	0	0	0	0	0	0
200-020000	11	11	0	0	0	0	0	0	0	0	0	0	0	0
200-030000	13	13	0	0	0	0	0	0	0	0	1	0	0	0
200-040000	11	11	0	0	0	0	0	0	0	0	1	0	0	0
200-050000	8	8	0	0	0	0	0	0	0	0	0	0	0	0
200-060000	9	9	0	0	0	0	0	0	0	0	0	0	0	0
200-070000	9	9	0	0	0	0	0	0	0	0	0	0	0	0
200-080000	8	8	0	0	0	0	0	0	0	0	0	0	0	0
200-090000	7	7	0	0	0	0	0	0	0	0	0	0	0	0
200-100000	12	12	0	0	0	0	0	0	0	0	0	0	0	0
200-110000	14	14	0	0	0	0	0	0	0	0	0	0	0	0
200-120000	23	23	0	0	0	0	0	0	0	0	1	0	0	0
200-130000	20	20	0	0	0	0	0	0	0	0	0	0	0	0
200-140000	21	21	0	0	0	0	0	0	0	0	0	0	0	0
200-150000	18	18	0	0	0	0	0	0	0	0	0	0	0	0

200-160000	20	20	0	0	0	0	0	0	0	0	0	0	0	0
200-170000	18	18	0	0	0	0	0	0	0	0	0	0	0	0
0 Vol.no.	lrgst.mass	err.	reduce-quality		reduce-extrap.		reduce-mass		reduce-propty.		min.courant		reduce-courant	
	edit	total	edit	total	edit	total	edit	total	edit	total	edit	total	edit	total
200-180000	21	21	0	0	0	0	0	0	0	0	0	1	0	0
200-190000	22	22	0	0	0	0	0	0	0	0	0	0	0	0
200-200000	20	20	0	0	0	0	0	0	0	0	0	0	0	0
200-210000	21	21	0	0	0	0	0	0	0	0	0	0	0	0
200-220000	20	20	0	0	0	0	0	0	0	0	0	0	0	0
200-230000	21	21	0	0	0	0	0	0	0	0	0	0	0	0
200-240000	21	21	0	0	0	0	0	0	0	0	0	0	0	0
200-250000	18	18	0	0	0	0	0	0	0	0	0	0	0	0
200-260000	20	20	0	0	0	0	0	0	0	0	0	0	0	0
200-270000	19	19	0	0	0	0	0	0	0	0	0	0	0	0
200-280000	23	23	0	0	0	0	0	0	0	0	0	0	0	0
200-290000	24	24	0	0	0	0	0	0	0	0	0	0	0	0
200-300000	22	22	0	0	0	0	0	0	0	0	0	0	0	0
200-310000	22	22	0	0	0	0	0	0	0	0	0	0	0	0
200-320000	22	22	0	0	0	0	0	0	0	0	0	0	0	0
200-330000	21	21	0	0	0	0	0	0	0	0	0	0	0	0
200-340000	16	16	0	0	0	0	0	0	0	0	0	0	0	0
200-350000	17	17	0	0	0	0	0	0	0	0	0	0	0	0
200-360000	18	18	0	0	0	0	0	0	0	0	0	0	0	0
200-370000	16	16	0	0	0	0	0	0	0	0	0	0	0	0
200-380000	21	21	0	0	0	0	0	0	0	0	0	0	0	0
200-390000	19	19	0	0	0	0	0	0	0	0	0	0	0	0
200-400000	25	25	0	0	0	0	0	0	0	0	0	0	0	0
200-410000	37	37	0	0	0	0	0	0	0	0	0	0	0	0
200-420000	54	54	0	0	0	0	0	0	0	0	0	0	0	0
200-430000	59	59	0	0	0	0	0	0	0	0	0	0	0	0
200-440000	742	742	0	0	0	0	0	0	0	0	0	0	0	0
997-010000	0	0	0	0	0	0	0	0	0	0	0	0	0	0
998-010000	0	0	0	0	0	0	0	0	0	0	0	0	0	0
999-010000	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0System 1	*none*													
0 Jun.no.	from vol.	to vol	liq j vel.		vap j vel.		mass flow		jun.area		throat	junction	flow no.	advs. choked
			(ft/sec)		(ft/sec)		(lb/sec)		(ft2)		ratio	flags	reg1 last edit	total
JUN-137	sngljun													
137-000000	-140-010001	200-010001	4.4142		4.4142		73.317		0.36010		1.0000	001000	bby 0	0 0 0
P140	pipe													
140-010000	140-010002	140-020001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-020000	140-020002	140-030001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-030000	140-030002	140-040001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-040000	140-040002	140-050001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-050000	140-050002	140-060001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-060000	140-060002	140-070001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-070000	140-070002	140-080001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-080000	140-080002	140-090001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-090000	140-090002	140-100001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0
140-100000	140-100002	140-110001	-4.4142		-4.4142		-73.317		0.36010		1.0000	001000	bby 0	0 0 0

140-110000	140-110002	140-120001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-120000	140-120002	140-130001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-130000	140-130002	140-140001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-140000	140-140002	140-150001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-150000	140-150002	140-160001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-160000	140-160002	140-170001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-170000	140-170002	140-180001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-180000	140-180002	140-190001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-190000	140-190002	140-200001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-200000	140-200002	140-210001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-210000	140-210002	140-220001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-220000	140-220002	140-230001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-230000	140-230002	140-240001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-240000	140-240002	140-250001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-250000	140-250002	140-260001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-260000	140-260002	140-270001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-270000	140-270002	140-280001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-280000	140-280002	140-290001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
140-290000	140-290002	140-300001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
JUN-145 sngl jun												
145-000000	140-300002	150-010001	-4.4142	-4.4142	-73.317	0.36010	1.0000	001000	bby	0	0	0
IND031 valve												
154-000000	150-010003	998-010000	5.1939	5.1939	86.267	0.36010	1.04971E-02	000102	bby	1	1950	2228
JUN-155 sngl jun												
155-000000	150-010002	160-010001	-13.665	-13.665	-159.58	0.25320	1.0000	001000	bby	0	0	0
JUN-160 pipe												
160-010000	160-010002	160-020001	-13.665	-13.665	-159.58	0.25320	1.0000	001000	bby	0	0	0
160-020000	160-020002	160-030001	-13.664	-13.664	-159.58	0.25320	1.0000	001000	bby	0	0	0
160-030000	160-030002	160-040001	-13.664	-13.664	-159.58	0.25320	1.0000	001000	bby	0	0	0
160-040000	160-040002	160-050001	-13.664	-13.664	-159.58	0.25320	1.0000	001000	bby	0	0	0
160-050000	160-050002	160-060001	-13.664	-13.664	-159.58	0.25320	1.0000	001000	bby	0	0	0
JUN-165 sngl jun												
165-000000	160-060002	170-010001	-23.552	-23.552	-159.58	0.14690	1.0000	001000	bby	0	0	0
P170 pipe												
170-010000	170-010002	170-020001	-23.552	-23.552	-159.58	0.14690	1.0000	001000	bby	0	0	0
170-020000	170-020002	170-030001	-23.551	-23.551	-159.58	0.14690	1.0000	001000	bby	0	0	0
170-030000	170-030002	170-040001	-23.551	-23.551	-159.58	0.14690	1.0000	001000	bby	0	0	0
170-040000	170-040002	170-050001	-23.551	-23.551	-159.58	0.14690	1.0000	001000	bby	0	0	0
170-050000	170-050002	170-060001	-23.543	-23.543	-159.58	0.14690	1.0000	001000	bby	0	0	0
INI175 valve												
175-000000	170-060002	999-010000	-23.543	-23.543	-159.58	0.14690	1.0000	001100	bby	0	0	0
PIPE pipe												
200-010000	200-010002	200-020001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0
200-020000	200-020002	200-030001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0
200-030000	200-030002	200-040001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0
200-040000	200-040002	200-050001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0
200-050000	200-050002	200-060001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0
200-060000	200-060002	200-070001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0
200-070000	200-070002	200-080001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0
200-080000	200-080002	200-090001	4.4142	4.4142	73.317	0.36010	1.0000	001000	bby	0	0	0

140-090000	1.0000	0.00000	1.05158E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-100000	1.0000	0.00000	1.05158E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-110000	1.0000	0.00000	1.05158E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-120000	1.0000	0.00000	1.05159E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-130000	1.0000	0.00000	1.05164E-02	9.468E-02	0.000	1.83	1.83	0.000	0.000	0	0	0
140-140000	1.0000	0.00000	1.05164E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-150000	1.0000	0.00000	1.05164E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-160000	1.0000	0.00000	1.05164E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-170000	1.0000	0.00000	1.05165E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-180000	1.0000	0.00000	1.05165E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-190000	1.0000	0.00000	1.05165E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-200000	1.0000	0.00000	1.05166E-02	9.468E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
140-210000	1.0000	0.00000	1.05167E-02	9.468E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
140-220000	1.0000	0.00000	1.05167E-02	9.468E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
140-230000	1.0000	0.00000	1.05168E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-240000	1.0000	0.00000	1.05168E-02	9.468E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
140-250000	1.0000	0.00000	1.05169E-02	9.468E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
140-260000	1.0000	0.00000	1.05169E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-270000	1.0000	0.00000	1.05170E-02	9.468E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
140-280000	1.0000	0.00000	1.05170E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
140-290000	1.0000	0.00000	1.05171E-02	9.468E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
145-000000	1.0000	0.00000	1.05165E-02	0.113	0.000	0.000	0.000	0.000	0.000	0	0	0
154-000000	1.0000	0.00000	1.05165E-02	0.000	0.000	0.000	0.000	0.000	0.000	0	0	0
155-000000	1.0000	0.00000	9.03078E-03	7.153E-02	0.000	1.02	1.02	0.000	0.000	0	0	0
160-010000	1.0000	0.00000	1.05179E-02	0.113	0.000	0.000	0.000	0.000	0.000	0	0	0
160-020000	1.0000	0.00000	1.05187E-02	0.113	0.000	0.210	0.210	0.000	0.000	0	0	0
160-030000	1.0000	0.00000	1.05190E-02	0.113	0.000	0.000	0.000	0.000	0.000	0	0	0
160-040000	1.0000	0.00000	1.05192E-02	0.113	0.000	0.000	0.000	0.000	0.000	0	0	0
160-050000	1.0000	0.00000	1.05200E-02	0.113	0.000	0.210	0.210	0.000	0.000	0	0	0
165-000000	1.0000	0.00000	8.28906E-03	9.560E-02	0.000	1.03	1.03	0.000	0.000	0	0	0
170-010000	1.0000	0.00000	1.05246E-02	0.153	0.000	0.000	0.000	0.000	0.000	0	0	0
170-020000	1.0000	0.00000	1.05289E-02	0.153	0.000	0.420	0.420	0.000	0.000	0	0	0
170-030000	1.0000	0.00000	1.05316E-02	0.153	0.000	0.210	0.210	0.000	0.000	0	0	0
170-040000	1.0000	0.00000	1.05343E-02	0.153	0.000	0.210	0.210	0.000	0.000	0	0	0
170-050000	1.0000	0.00000	1.06324E-02	0.153	0.000	12.8	12.8	0.000	0.000	0	0	0
175-000000	1.0000	0.00000	1.06330E-02	7.657E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
200-010000	1.0000	0.00000	1.05151E-02	9.214E-02	0.000	0.112	0.112	0.000	0.000	0	0	0
200-020000	1.0000	0.00000	1.05150E-02	9.214E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
200-030000	1.0000	0.00000	1.05150E-02	9.214E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
200-040000	1.0000	0.00000	1.05149E-02	9.214E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
200-050000	1.0000	0.00000	1.05148E-02	9.214E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
200-060000	1.0000	0.00000	1.05148E-02	9.214E-02	0.000	0.196	0.196	0.000	0.000	0	0	0
200-070000	1.0000	0.00000	1.05147E-02	9.214E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
200-080000	1.0000	0.00000	1.05147E-02	9.214E-02	0.000	0.840	0.840	0.000	0.000	0	0	0
200-090000	1.0000	0.00000	1.05145E-02	9.214E-02	0.000	0.596	0.596	0.000	0.000	0	0	0
200-100000	1.0000	0.00000	1.05143E-02	9.214E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
200-110000	1.0000	0.00000	1.05143E-02	9.214E-02	0.000	1.68	1.68	0.000	0.000	0	0	0
200-120000	1.0000	0.00000	1.05138E-02	9.214E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
200-130000	1.0000	0.00000	1.05138E-02	9.214E-02	0.000	0.000	0.000	0.000	0.000	0	0	0
200-140000	1.0000	0.00000	1.05138E-02	9.214E-02	0.000	0.000	0.000	0.000	0.000	0	0	0

200-150000	1 0000	0 00000	1 05137E-02	9 214E-02	0 000	0 392	0 392	0 000	0 000	0	0	0
0 Jun no	voidfj	voidgj	f1j	fwalfj	fwalgj	fjunf	fjunr	formfj	formgj	no	advs	ccfl
			(1b-s2/ft5)							last	edit	total
200-160000	1 0000	0 00000	1 05136E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-170000	1 0000	0 00000	1 05136E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-180000	1 0000	0 00000	1 05136E-02	9 214E-02	0 000	0 196	0 196	0 000	0 000	0	0	0
200-190000	1 0000	0 00000	1 05135E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-200000	1 0000	0 00000	1 05135E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-210000	1 0000	0 00000	1 05134E-02	9 214E-02	0 000	0 196	0 196	0 000	0 000	0	0	0
200-220000	1 0000	0 00000	1 05134E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-230000	1 0000	0 00000	1 05133E-02	9 214E-02	0 000	0 196	0 196	0 000	0 000	0	0	0
200-240000	1 0000	0 00000	1 05133E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-250000	1 0000	0 00000	1 05132E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-260000	1 0000	0 00000	1 05132E-02	9 214E-02	0 000	0 196	0 196	0 000	0 000	0	0	0
200-270000	1 0000	0 00000	1 05131E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-280000	1 0000	0 00000	1 05131E-02	9 214E-02	0 000	0 196	0 196	0 000	0 000	0	0	0
200-290000	1 0000	0 00000	1 05130E-02	9 214E-02	0 000	0 196	0 196	0 000	0 000	0	0	0
200-300000	1 0000	0 00000	1 05130E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-310000	1 0000	0 00000	1 05129E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-320000	1 0000	0 00000	1 05129E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-330000	1 0000	0 00000	1 05129E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-340000	1 0000	0 00000	1 05129E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-350000	1 0000	0 00000	1 05128E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-360000	1 0000	0 00000	1 05128E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-370000	1 0000	0 00000	1 05128E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-380000	1 0000	0 00000	1 05128E-02	9 214E-02	0 000	1 83	1 83	0 000	0 000	0	0	0
200-390000	1 0000	0 00000	1 05123E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-400000	1 0000	0 00000	1 05122E-02	9 214E-02	0 000	0 392	0 392	0 000	0 000	0	0	0
200-410000	1 0000	0 00000	1 05121E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-420000	1 0000	0 00000	1 05121E-02	9 214E-02	0 000	0 000	0 000	0 000	0 000	0	0	0
200-430000	1 0000	0 00000	1 05121E-02	9 214E-02	0 000	0 196	0 196	0 000	0 000	0	0	0
205-000000	1 0000	0 00000	1 05120E-02	4 607E-02	0 000	0 000	0 000	2 361E+04	2 361E+04	0	0	0

0 _ _ _ INFO Final time= 200 000 sec advancements attempted= 2273
0 _ _ _ INFO transient terminated by end of time step cards

Appendix E

Core Uncovery Timing Calculations

Small Break ISLOCA Bounding Calculation for a Westinghouse Four Loop PWR

This appendix documents the bounding calculation used to estimate the time for the onset of core uncover for a small break interfacing system LOCA (ISLOCA) sequence in a generic four loop Westinghouse PWR. Calculations shown in this appendix were performed with a desk top computer program called *Mathematica*. In particular, the core uncover time estimate was made after ECCS pump suction was lost from the RWST (this will define the time when the RWST is 'empty'). It should be noted that a possible core uncover may occur prior to losing suction to the RWST, but this issue was not addressed in this document. The uncover time is defined as the time at which the collapsed vessel liquid level will drop below the top of the core fuel rods.

The break was assumed to be outside of containment on one of the LPI injection lines and was approximated as an equivalent .03 ft**2 primary system break. No credit was given for form loss, friction, or other pressure drop effects that may reduce the break mass flow rate. As a consequence the time at which the RWST becomes empty will be shorter than in an actual accident.

These break assumptions were incorporated into a simplified 5 volume RELAP5 model that was used to estimate the time to empty the RWST. The primary system, ECCS, and the break were explicitly modeled. The assumed break configuration will lead to a shorter and therefore more conservative time estimate of when the RWST becomes empty.

Several additional assumptions were incorporated into the RELAP5 model to ensure a conservative time estimate to empty the RWST.

1. The ECCS water supply was limited to 350,000 gal.
2. Auxiliary feedwater was available and it was assumed that the steam generators were depressurized to enhance primary to secondary heat transfer. It was further assumed that any sensible heat added to the primary coolant from core decay power was immediately removed by the steam generators. Thus no stored energy or decay heat was included in the RELAP5 model. These assumptions resulted in a lower primary system pressure and a higher ECCS mass flow rate.

3. The accumulators are available and the drain out time will be estimated by dividing the total break mass flow rate by the total volume of the accumulator liquid inventory. The accumulators were not in the RELAP5 model and this time estimate was done separately.
4. It is assumed that the transient is initiated at 100 % power conditions and that there is no significant delay for reactor scram.
5. It is assumed that the break flow rate will stabilize near the LPI shutoff head and the time averaged break mass flow rate will balance with the time averaged ECCS mass flow rate. This assumption is conservative since most small break LOCA's will stabilize at higher pressures. The steady-state ECCS mass flow rate was modeled by equating it with the steady-state choked break mass flow rate at the LPI activation pressure. This time averaged balance accounted for short intermittent periods when the LPI would activate. Continuous LPI operation was not possible since the total LPI mass flow rate was significantly larger than the break mass flow rate.
6. Two HPI, 1-LPI, and 2-charging trains will be available to refill the primary system.

The simplified RELAP5 LOCA calculation indicated that after about one hour the primary system pressure stabilized to around 200 psia ; the ECCS/mass flow rates stabilized to approximately 330 lbs/sec. Under these conditions the break mass flow was choked. We now define the following variables:

Tank is the mass of the RWST tank in lbs
BRKFLOW is the steady-state break mass flow rate
TIMEP1 is the time in hours for the RWST to empty

Using the above RELAP5 results the estimated time for the RWST to empty was calculated as follows:

```
Tank= 350000. GAL * .1337 CFT/GAL * 62.0 LBS/CFT
BRKFLOW= 330. LBS/SEC
TIME = Tank/BRKFLOW
TIME
8791.79 SEC
TIMEP1=TIME/(3600. SEC/HR)
2.44216 HR
```

Sources of uncertainty in this estimate are as follows:

1. The fact that the primary system pressure is a non-constant, nonlinear function of time.
2. The HPI and charging mass flow rates are nonlinear functions of the primary system pressure.
3. Primary to secondary heat transfer may greatly alter outcome depending on whether the operator decided to depressurize the steam generators.
4. Flooding out the auxiliary building may disrupt or destroy the ECCS pumping equipment before the RWST tank is empty.
5. The uncertainties in the RELAP5 critical flow model.

An additional correction was made for the loop accumulator tanks (the UHI accumulator system is no longer in use). The total liquid volume of the four accumulator tanks equals 4000 ft**3 which was the assumed nominal value for a four loop PWR. The mass flow rate is assumed not to exceed the break flow rate for SBLOCA conditions. Once the accumulators are activated we will assume that they empty at an average rate equal to the break flow which will give us the following result.

We define the additional variables:

ACUVOL is the total liquid volume of the accumulator tanks
TIMEP2 is the estimated time for the accumulator tanks to drain in hours

ACUVOL= 4000. CUFT

BRKFLOW = 330. LBS/SEC

TIMEP2 = (ACUVOL/ BRKFLOW)*(62.4 LBS/CUFT)/(3600. SEC/HR)

TIMEP2

0.210101 HR

Therefore the time to drain the RWST is equal to TIMEP1+TIMEP2 or approximately 2.6 hour. The remainder of this narrative will detail how the time to core boiloff was calculated. The core boiloff estimate is based on the assumption that no additional ECCS liquid is injected into the primary system after 2.6 hr.

In order to estimate the time to core boiloff the following assumptions were made:

1. Liquid in the loop regions was assumed not to drain into the reactor vessel . The liquid in the loop regions was assumed to either exit out the break or accumulate in the loop seals. This assumption is conservative since after the RWST has drained some liquid in the primary loops will drain back into the vessel region. Unless a full scale systems calculation is done there is no easy way of getting a good estimate of how much liquid would drain into the vessel.
2. During the bulk boiling phase, prior to core uncover the vessel level response will be treated in a simplified manner that does not account for system repressurization or the potential for level depression driven by the differential upper plenum/cold leg pressure. If the break is located in the cold leg this pressure differential may shorten the time to which core uncover develops. This assumption is not conservative, however, because of other conservatism in this analysis the above assumption is deemed to be reasonable for simplified hand calculations.
3. It was assumed that the vessel upper head is completely drained by the time the RWST has emptied. This assumption is reasonable since most PWR LOCA scenarios lead to vessel upper head voiding. The assumption of 100 % voiding is conservative since this will lead to an earlier onset of core boiloff.
4. It was assumed that any sensible or latent heat added to the vessel liquid will result in no significant repressurization of the primary system and therefore in the total integrated core decay power needed to initiate core boiloff.
5. It was assumed that the remaining liquid in the vessel available for boiloff is at the bulk subcooled conditions of 100 F at a pressure of 200 psia. This temperature is close to the assumed 70 F temperature of the ECCS liquid. The additional 30 F is assumed to be from the core decay heat. This number is not based on rigorous quantitative arguments. Unless a full scale simulation is performed to model the ECCS/vessel liquid mixing dynamics it is not clear that an exact bulk temperature can be easily defined.

Shown below are the steam table values needed to calculate the enthalpy changes which will be used in the energy balance calculations for the core boiloff estimates. From the steam table data shown below we define the following variables :

1. h_{l200} is the enthalpy of the subcooled vessel liquid at 200 psia, 100 F.
2. h_{ls200} is the saturation enthalpy of the vessel liquid at 200 psia.
3. rho is the bulk density of the liquid at 100 F.

4. VOLREMAIN is the volume of the reactor vessel minus the regions above the hot and cold leg center lines.

5. TOTALEN1 is the total sensible heat needed to heat the reactor vessel coolant from subcooled to saturated conditions.

LIQUID STATE PROPERTIES

p	=	200.0000	psia	T	=	100.0000	F
rho	=	62.03413	lbm/ft3	h	=	68.52453	BTU/lbm
s	=	.1293764	BTU/lbm-R	u	=	67.92792	BTU/lbm
x	=	.0000000		alpha	=	.0000000	
HEAT CAPACITY							
Cp	=	.9972401	BTU/lbm-R	Cv	=	.9739143	BTU/lbm-R

SATURATED LIQUID PROPERTIES

p	=	200.0000	psia	T	=	381.8044	F
rho	=	54.38568	lbm/ft3	h	=	355.5219	BTU/lbm
s	=	.5438131	BTU/lbm-R	u	=	354.8414	BTU/lbm
x	=	.0000000		alpha	=	.0000000	
HEAT CAPACITY							
Cpf	=	1.067540	BTU/lbm-R	Cpg	=	.6612970	BTU/lbm-R
Cvf	=	.8065807	BTU/lbm-R	Cvg	=	.4590623	BTU/lbm-R

SATURATED VAPOR PROPERTIES

p	=	200.0000	psia	T	=	381.8044	F
rho	=	.4372073	lbm/ft3	h	=	1198.293	BTU/lbm
s	=	1.545494	BTU/lbm-R	u	=	1113.642	BTU/lbm
x	=	1.000000		alpha	=	1.000000	
HEAT CAPACITY							
Cpf	=	1.067540	BTU/lbm-R	Cpg	=	.6612970	BTU/lbm-R
Cvf	=	.8065807	BTU/lbm-R	Cvg	=	.4590623	BTU/lbm-R

h1200 = 68.52 btu/lbm

hls200 = 355. btu/lbm

rho=62. (lbm/CUFT)

VOLREMAIN= (4550.-351.-285.-243.-48.-80.-123.-130.) CUFT

TOTALEN1= rho*(hls200-h1200)*VOLREMAIN*1055. J/btu

TOTALEN1

10

6.16502 10 J

The vessel liquid is now assumed to be at bulk boiling conditions. Additional energy added to the vessel liquid will be assumed to result in liquid vaporization since no liquid in the loop regions is assumed to drain into the vessel. In order to begin core voiding, all the liquid above the core will be assumed to be turned into steam. From an actual physics standpoint bulk boiling in the vessel will cause core voiding before the liquid in the upper annulus and upper plenum are completely vaporized. This will not be of concern because the core mixture level will still be above the fuel rod region. The total energy needed to vaporize the liquid region above the fuel rods becomes equal to the energy needed to vaporize the upper plenum region. To complete this energy balance calculation the following variables are defined.

1. volvap is the remaining upper plenum volume of saturated liquid to be vaporized.
2. deltah is the latent heat change
3. rho2 is the liquid saturation density
4. TOTALEN2 is the total energy needed for upper plenum vaporization.

The energy balance then becomes

```

volvap= (71. +99. +123.+283. +281.) CUFT
deltah  = (1198.-355.) btu/lbm
rho2    = 59.4 lbm/CUFT
TOTALEN2= volvap*deltah*rho2*1055. J/btu
TOTALEN2

```

```

10
4.52738 10 J

```

The total energy needed to initiate core uncover then becomes:

```

TOTE = (TOTALEN1 + TOTALEN2)

```

```

11
1.06924 10 J

```

We must now equate the total integrated core decay power between the time (TIMEP1+TIMEP2) and some unknown time to the total energy TOTE. What is implicitly assumed in this analysis is that reactor scram started at the same time ECCS was initiated namely at $t=0.0$ s, the beginning of the transient. This assumption is conservative since any delay in core scram will keep the primary system pressurized and reduce ECCS flow. We shall define the time variable TIMEP3b as:

```
TIMEP3b =TIMEP1+TIMEP2
TIMEP3b
```

```
2.65226 HR
```

We shall further define decay(t) as the normalized ANS core decay power. The expression decay(t) will be calculated as a polynomial with a Mathematica curve fit routine and integrated between TIMEP3b and some unknown time t. This polynomial equation will be multiplied by the core power at 100 % operating conditions and equated to TOTE. Using the Mathematica algebraic/numerical routines t will be solved for.

The normalized decay table from the standard ANS decay model is:

Time(s)	Normalized Power
100.	.0331
400.	.0235
800.	.0196
1000.	.0185
2000.	.0157
4000.	.0128
8000.	.0105
10000.	.00965
20000.	.00795
60000.	.00566
100000.	.00475

In order to estimate the time to reach saturation conditions and core boiloff times, we need to know the core decay power at the time the RWST empties. The nominal 100 % nominal power for the four loop plant is assumed to be 3411 Mw.

We will do a quadratic curve fit with the above data from 4000. to 10000. s as follows:

```
Decpow = { { 8000. , .0105},
           {10000. , .00965},{20000., .00795}}
```

```
decay[t_]=Fit[ Decpow,{1,t,t^2},t]
```

```
0.0156 - 8.075 10-7 t + 2.125 10-11 t2
```

We shall now take the normalized power curve $dcpow(t)$ and integrate the power from TIMEP3b (in seconds) to t . Since *Mathematica* used a quadratic curve fit to the above ANS data the resultant integrated expression is some what messy looking. It should be noted that this expression is a good approximation in the time domain range 8000-20000 s. We now define a new polynomial expression $Q(t)$ which is the integral of $decay(t)$ between the unknown time t and TIMEP3b. Recall that the time TIMEP3b is the time at which the RWST becomes empty.

$$Q[t_]=N[Integrate[decay[x], {x, (TIMEP3b/HR)*3600., t }]]$$

$$-118.308 + 0.0156 t - 4.0375 \cdot 10^{-7} t^2 + 7.08333 \cdot 10^{-12} t^3$$

The expression $Q(t)$ is now multiplied by 3411 Mw and equated to TOTE so that we can solve for t . To do the numerics the units tags have been set equal to 1.0. The expression and Mathematica equivalent forms are as follows:

$$TOTE = 3411 \int_{TIMEP3b}^t decay(t) dt = 3411 * Q(t)$$

```
J=1.
N[Solve[(TOTE)-3411.*1000000.*Q[t]==0., t]]
{{t -> 12945.6}, {t -> 22027.2 - 33865.1 I}, {t -> 22027.2 + 33865.1 I}}
```

Pack the above roots into an array Troot and pick the first element which is the real solution and define it as the variable boiltime

```
Troot= t/.%
boiltime= First[Troot]
12945.6
```

In the above methodology it has been implicitly assumed that no energy removal by the steam generators is taking place after the RWST has emptied. Moreover, only the real solution for t has meaning. The last two roots of the above equation are non-physical imaginary values and will be ignored.

The time to core boiloff relative to the time of accident initiation becomes

```
boiltime/3600. HR
3.596 HR
```

Large Break ISLOCA Bounding Calculation for a Westinghouse Four Loop PWR

This appendix documents the bounding calculation used to estimate the time for the onset of core uncover for a large break interfacing system LOCA (ISLOCA) sequence in a generic four loop Westinghouse PWR. Calculations shown in this appendix were performed with a desk top computer program called *Mathematica*. In particular, the core uncover time estimate was made after ECCS pump suction was lost from the RWST (this will define the time when the RWST is 'empty'). It should be noted that a possible core uncover may occur prior to losing suction to the RWST, but this issue was not addressed in this document. The uncover time is defined as the time at which the collapsed vessel liquid level will drop below the top of the core fuel rods.

The break was assumed to be outside of containment on one of the LPI injection lines and was approximated as an equivalent .7854 ft**2 primary system break. No credit was given for form loss, friction, or other pressure drop effects that may reduce the break mass flow rate. As a consequence the time at which the RWST becomes empty will be shorter than in an actual accident.

These break assumptions were incorporated into a simplified 5 volume RELAP5 model that was used to estimate the time to empty the RWST. The primary system, ECCS, and the break were explicitly modeled. The assumed break configuration will lead to a shorter and therefore more conservative time estimate of when the RWST becomes empty.

Several additional assumptions were incorporated into the RELAP5 model to ensure a conservative time estimate to empty the RWST.

1. The ECCS water supply was limited to 350,000 gal.
2. Auxiliary feedwater was available and it was assumed that the steam generators were depressurized to enhance primary to secondary heat transfer. It was further assumed that any sensible heat added to the primary coolant from core decay power was immediately removed by the steam generators. Thus, no stored energy or decay heat was included in the RELAP5 model. These assumptions resulted in a lower primary system pressure and a higher ECCS mass flow rate.

3. The accumulators are available and the corresponding drain out time was estimated by dividing the total break mass flow rate by the total volume of the accumulator liquid inventory. The accumulators were not in the RELAP5 model and this time estimate was done separately.
4. It was assumed that the transient is initiated at 100 % power conditions and that there is no significant delay for reactor scram.
5. Two HPI, 1-LPI, and 2-charging trains will be available to refill the primary system.

The simplified RELAP5 LOCA calculation indicated after about 300 s, that the primary system pressure stabilized at 15 psia ; the ECCS/mass flow rates stabilized at approximately 930 lbs/sec. Under these conditions the break mass flow was unchoked. We now define the following variables:

Tank is the mass of the RWST tank in lbs
BRKFLOW is the steady-state break mass flow rate
TIMEP1 is the time in hours for the RWST to empty

Using the above RELAP5 results the estimated time for the RWST to empty was calculated as follows:

```
Tank= 350000. GAL * .1337 CFT/GAL * 62.0 LBS/CFT
BRKFLOW= 930. LBS/SEC
TIME = Tank/BRKFLOW
TIME
3119.67 SEC
TIMEP1=TIME/(3600. SEC/HR)
0.866574 HR
```

Sources of uncertainty in this estimate are as follows:

1. The fact that the primary system pressure is a non-constant, nonlinear function of time.
2. The HPI and charging mass flow rates are nonlinear functions of the primary system pressure.
3. Primary to secondary heat transfer may greatly alter outcome depending on whether the operator decided to depressurize the steam generators.
4. Flooding out the auxiliary building may disrupt or destroy the ECCS pumping equipment before the RWST tank is empty.
5. The uncertainties in the RELAP5 critical flow model.

An additional correction was made for the loop accumulator tanks (the UHI accumulator system is no longer in use). The total nominal liquid volume of the four accumulator tanks equals 4000 ft³. The mass flow rate is assumed not to exceed the break flow rate for LBLOCA conditions. Once the accumulators are activated we will assume that they empty at an average rate equal to the break flow which will give us the following result.

We define the additional variables:

ACUVOL is the total liquid volume of the accumulator tanks

TIMEP2 is the estimated time for the accumulator tanks to drain in hours

ACUVOL= 4000. CUFT

BRKFLOW = 1030. LBS/SEC

TIMEP2 = (ACUVOL/ BRKFLOW) * (62.4 LBS/CUFT) / (3600. SEC/HR)

TIMEP2

0.0673139 HR

Therefore the time to drain the RWST is equal to TIMEP1+TIMEP2 or approximately one hour. The remainder of this narrative will detail how the time to core boiloff was calculated. The core boiloff estimate is based on the assumption that no additional ECCS liquid is injected into the primary system after about one hour. In order to estimate the time to core boiloff the following assumptions were made:

1. Liquid in the loop regions was assumed not to drain into the reactor vessel . The liquid in the loop regions was assumed to either exit out the break or accumulate in the loop seals. This assumption is conservative since after the RWST has drained some liquid in the primary loops will drain back into the vessel region. Unless a full scale systems calculation is done there is no way of getting a good estimate of how much liquid would drain into the vessel.
2. During the bulk boiling phase, prior to core uncover the vessel level response will be treated in a simplified manner that does not account for possible system repressurization or the potential for level depression driven by the differential upper plenum/cold leg pressure. If the break is located in the cold leg this pressure differential may shorten the time to which core uncover develops. This assumption is not conservative, however because of other conservatism in this analysis the above assumption is deemed to be reasonable for simplified hand calculations.
3. It was assumed that the vessel upper head is completely drained by the time the RWST has emptied. This assumption is reasonable since most PWR LOCA scenarios lead to vessel upper head voiding. The assumption of 100 % voiding is conservative since this will lead to an earlier onset of core boiloff.
4. It was assumed that any sensible or latent heat added to the vessel liquid will result in no significant repressurization of the primary system and therefore in the total integrated core decay power needed to initiate core boiloff.
5. It was assumed that the remaining liquid in the vessel available for boiloff is at bulk subcooled conditions of 100 F and 15 psia. This temperature is close to the assumed 70 F temperature of the ECCS liquid. The additional 30 F is assumed to be from the core decay heat. This number is not based on rigorous quantitative arguments. Unless a full scale simulation is performed to model the ECCS/vessel liquid mixing dynamics it is not clear that an exact bulk temperature can be easily defined.

Shown below are the steam table values needed to calculate the enthalpy changes which were used in the energy balance calculations for the core boiloff estimates.

Liquid State Properties

p	=	15.00000	psia	t	=	100.0000	F	
rho	=	62.02988	lbm/ft3	h	=	68.02748		BTU/lbm
s	=	.1295396	BTU/lbm-R	u	=	67.98141		BTU/lbm
x	=	.0000000		alpha	=	.0000000		

Saturated Vapor Properties

p	=	15.00000	psia	t	=	213.3267	F	
rho	=	3.7417938E-02	lbm/ft3	h	=	1153.862		BTU/lbm
s	=	1.756776	BTU/lbm-R	u	=	1077.479		BTU/lbm
x	=	1.000000		alpha	=	1.000000		

Saturated Liquid Properties

p	=	15.00000	psia	t	=	213.3267	F	
rho	=	59.78068	lbm/ft3	h	=	181.3563		BTU/lbm
s	=	.3139330	BTU/lbm-R	u	=	181.3084		BTU/lbm
x	=	.0000000		alpha	=	.0000000		

From the above steam table data we define the following variables :

1. h115 is the enthalpy of the subcooled vessel liquid at 15 psia, 100 F.
2. hls15 is the saturation enthalpy of the vessel liquid at 15 psia, 100 F.
3. rho is the bulk density of the liquid at 100 F.
4. VOLREMAIN is the volume of the reactor vessel minus the regions above the hot and cold leg center lines.
5. TOTALEN1 is the total sensible heat needed to heat the reactor vessel coolant from subcooled to saturated conditions.

```

hl15 = 68.03 btu/lbm
hls15 = 181.4 btu/lbm
rho=62. (lbm/CUFT)
VOLREMAIN= (4550.-351.-285.-243.-48.-80.-123.-130.) CUFT
TOTALEN1=      rho*(hls15-hl15)*VOLREMAIN*1055. J/btu
TOTALEN1

10
2.43971 10 J

```

The vessel liquid is now assumed to be at bulk boiling conditions. Additional energy added to the vessel liquid will be assumed to result in liquid vaporization since no liquid in the loop regions is assumed to drain into the vessel. In order to begin core voiding, all the liquid above the core will be assumed to be turned into steam. From an actual physics standpoint bulk boiling in the vessel will cause core voiding before the liquid in the upper annulus and upper plenum are completely vaporized. This will not be of concern because the core mixture level will still be above the fuel rod region. The total energy needed to vaporize the liquid region above the fuel rods becomes equal to the energy needed to vaporize the saturated liquid in the upper plenum. To complete this energy balance calculation the following variables are defined.

1. volvap is the remaining upper plenum volume of saturated liquid to be vaporized.
2. deltah is the latent heat change
3. rho2 is the liquid saturation density
4. TOTALEN2 is the total energy needed for upper plenum vaporization.

The energy balance then becomes

```

volvap= (71. +99. +123.+283. +281.) CUFT
deltah = (1153.9-181.4) btu/lbm
rho2 = 59.8 lbm/CUFT
TOTALEN2= volvap*deltah*rho2*1055. J/btu
TOTALEN2

```

```

10
5.25804 10 J

```

The total energy needed to initiate core uncover becomes:

```

TOTE = (TOTALEN1 + TOTALEN2)

10
7.69775 10 J

```

We must now equate the total integrated core decay power between the time (TIMEP1+TIMEP2) and some unknown time to the total energy TOTE. What is implicitly assumed in this analysis is that reactor scram started at the same time ECCS was initiated namely at $t=0.0$ s the beginning of the transient. This assumption is conservative since any delay in core scram will keep the primary system pressurized and reduce ECCS flow. We shall define the time variable TIMEP3b as:

TIMEP3b =TIMEP1+TIMEP2

TIMEP3b

0.933888 HR

We shall further define decay(t) as the normalized ANS core decay power. The expression decay(t) will be calculated as a polynomial with a Mathematica curve fit routine and integrated between TIMEP3b and some unknown time t. This polynomial equation will be multiplied by the core power at 100 % operating conditions and equated to TOTE. Using the Mathematica algebraic/numerical routines t will be solved for.

The normalized decay table from the standard ANS decay model is:

Time(s)	Normalized Power
100.	.0331
400.	.0235
800.	.0196
1000.	.0185
2000.	.0157
4000.	.0128
8000.	.0105
10000.	.00965
20000.	.00795
60000.	.00566
100000.	.00475

In order to estimate the time to reach saturation conditions and core boiloff we need to know the core decay power at the time the RWST empties. The nominal 100 % power rating is assumed to be 3411 Mw.

We will do a quadratic curve fit with the above data from 4000. to 10000. s as follows:

```
Decpow = { {4000. , .0128} , { 8000. , .0105},
           {10000. , .00965}}
```

```
decay[t_]=Fit[ Decpow, {1,t,t^2},t]
```

```

-7      -11  2
0.0159 - 8.75 10  t + 2.5 10  t
```

We shall now take the normalized power curve dcpow(t) and integrate the power from TIMEP3b (in seconds) to t. Since *Mathematica* used a quadratic curve fit to the above ANS data the resultant integrated expression is some what messy looking. It should be noted that this expression is a good approximation in the range 4000-10000 s and will become progressively less accurate outside of this time interval. We now define a new polynomial expression Q(t) which is the integral of decay(t) between the unknown time t and TIMEP3b.

```
Q[t_]=N[Integrate[ decay[x], {x, (TIMEP3b/HR)*3600.,t }]]
```

```

-7  2      -12  3
-48.8273 + 0.0159 t - 4.375 10  t + 8.33333 10  t
```

The expression Q(t) is now multiplied by 3411 Mw and equated to TOTE so that we can solve for t. To do the calculation the units tags have been set equal to 1.0. The expression and Mathematica equivalent forms are as follows

$$TOTE = 3411 \int_{\text{TIMEP3b}}^t \text{decay}(t) dt = 3411 * Q(t)$$

```
J=1.
```

```
Solve[(TOTE)-3411.*1000000.*Q[t]==0., t]
```

```
{{t -> 5147.94}, {t -> 23676. - 33221.7 I}, {t -> 23676. + 33221.7 I}}
```

Pack the above roots into an array Troot and pick the first element which is the real solution and define it as the variable boiltime

```
Troot= t/.%
```

```
boiltime= First[Troot]
```

```
5147.94
```

In the above methodology it has been implicitly assumed that no energy removal by the steam generators is taking place after the RWST has emptied. Moreover, only the real solution for t has meaning. The last two roots of the above equation are non-physical imaginary values and will be ignored.

The time to core boiloff relative to the time of accident initiation becomes

```
boiltime/3600. HR
```

```
1.42998 HR
```

Appendix F

Calculation of System Rupture Probability

Appendix F

Calculation of System Rupture Probability

Interfacing system rupture probability is calculated from system pressure and component fragility data. System pressures are obtained from RELAP5 analysis of the pressurization scenario and the component fragility data are obtained from structural analyses performed by the ABB Impell Corporation (see Appendix I). The data from each analysis are combined probabilistically to obtain a total system rupture probability. The following sections describe the data required from the RELAP5 analysis, the component fragility data, and the methods used to calculate rupture probabilities. The first section describes the system pressure data. This is followed by a description of the component pressure fragility data and limiting assumptions. Then the individual component failure probability calculations are described, followed by the method used to obtain the system rupture probability from the individual failure probabilities.

SYSTEM PRESSURE

The pressure in the residual heat removal (RHR) system is estimated by RELAP5 simulations of the pressurization scenarios. In this case, with catastrophic failure of the boundary check valves, the pressure is calculated to be essentially constant at 2306 psia throughout the low pressure portion of the interfacing system. This result is driven by the low capacities of the available pressure relief valves, which limit mass flows within the RHR system. These low mass flow rates, in turn, limit the pressure drop between the reactor coolant system (RCS) and the relief valves. The exact pressure at any location within the interfacing system is uncertain, and is therefore represented as a random variable, which is assumed to be distributed normally with a mean pressure calculated by RELAP5, and with an assumed standard deviation of 50 psia. The uncertainty represented in the standard deviation includes only that attributable to instrumentation inaccuracies; modeling uncertainties are not included.

COMPONENT PRESSURE FRAGILITY DATA

The pressure capacities for each type of component in the interfacing system have been evaluated by the ABB Impell Corporation (see Appendix I). For the RHR system, screening calculations have eliminated some pressurization sequences from consideration. The retained sequences include only pressurization of the RHR system from the RHR pump discharge check valves to the high pressure-rated piping at the cold leg isolation check valves. The components included in this portion of the RHR system are valves, piping, flanges, and heat exchangers. The pressure capacities for these component types have been developed under the assumption of quasi-steady state pressure conditions. This assumption is based on RELAP5 analyses that show transient effects to be negligible given the relatively slow (greater than 10-second) rate of pressurization caused by check valve isolation failure. While a variety of analytical techniques have been used to obtain pressure fragility data for the above components, they all are expressed as follows:

$$P = \hat{P} * M * S \quad (1)$$

where

$\hat{P}$ = the component median pressure capacity

M = a lognormally distributed random variable having unit median and a logarithmic standard deviation, β_m , representing modeling uncertainties

S = a lognormally distributed random variable having unit median and a logarithmic standard deviation, β_s , representing material property uncertainties

The random variable P is lognormally distributed with logarithmic standard deviation β_c . In the tabulations provided in Appendix I, the separate uncertainty parameters, β_m and β_s , are not routinely provided. Instead one composite value, β_c , is used to represent both kinds of uncertainty, where β_c is given by

$$\beta_c = \sqrt{\beta_m^2 + \beta_s^2}.$$

The component pressure capacities and uncertainty parameters from Appendix I required for the RHR system analysis are summarized in Table 1. Application of the above data for determining system failure probabilities requires some understanding of the failure modes represented. The following discussions provide a summary of the important failure modes and data limitations. For further details, see Appendix I.

Piping

The piping used in the low pressure-rated portion of the RHR system between the pump discharge check valves and the high pressure-rated piping at the injection isolation check valves includes 8-inch schedule 20, 12-inch schedule 40, and 14-inch schedule 40 piping. Piping failure is assumed to occur when pressure-induced hoop stress in the unflawed pipe causes failure strain to be reached. Variability in the pressure stress results from many factors, such as variability in material strength, wall thickness, etc. Most important among these factors is the existence, size, and orientation of preexisting flaws or cracks in the piping system. Data on these factors are not currently available, so ABB Impell has developed variability data for the lognormal pressure capacity distribution by assuming that there is a failure probability of 10^{-3} at the yield stress of the piping. This implies that the probability of existence of a large flaw is 10^{-3} . The authors of the Impell report believe this may be an overly conservative assumption and recommend that this assumption should be reevaluated if it is found that a significant contribution to risk occurs due to pipe rupture at pressures well below the median failure pressure.

The data provided in Table 1 include no allowance for the number of feet of piping in the system. The data are also dependent on both the degree to which piping has been corroded and on the temperature of the fluid within the piping. In this analysis, a corrosion allowance of 0.020 in. is used. System temperature is, however, time dependent, with the lowest temperatures occurring before peak pressure is reached. Since failure stresses generally decrease with decreasing temperature, it is conservative to assume a higher temperature than is expected at peak pressure. In this analysis, a system temperature of 600° F is used.

Table 1. RHR system component pressure capacity data

Component	Med. Fail. Pressure psig	Log Std. Dev.	Mean Fail. Pressure psig	Log Mean	Error Factor
8 inch sched. 20 piping	2503	0.36	2671	7.825	1.81
12 inch sched. 40 piping	2859	0.36	3050	7.958	1.81
14 inch std wall piping	2369	0.36	2528	7.770	1.81
8 inch 600 psi flanges	4099	0.12	4129	8.318	1.22
Hx tube side cylinder	2040	0.33	2154	7.621	1.72
Hx tube side head buckling	2860	0.23	2937	7.959	1.46
Hx tube side head plastic collapse	1650	0.23	1694	7.409	1.46

Flanges

The flanges in the RHR system are 600 psi-rated 8-inch flanges. The data provided in Appendix I are tabulated according to the initial bolt stress and percent joint relaxation. The initial bolt stress is calculated from data in Table 3-28 of Appendix I, and from the plant's flange installation procedure. From Table 3-28 we obtain the following data for 8-inch flanges.

d = bolt diameter = 1 1/8 in

A = bolt area = .7896 in².

From the flange installation procedure we obtain

T = torque on bolts = 410 ft-lbs.

The force on the bolts is then

$$f = \text{force on bolts} = T/(C*d) = 25,725 \text{ lb} \quad (2)$$

where

C = constant = .71 to .20 (.17 for lubricated bolts).

The stress on the bolts is then

$$\text{stress} = f/A = 25,725/.7896 = 32,580 \text{ psi.} \quad (3)$$

Given this initial bolt stress, and assuming 0% joint relaxation, Table 3-29 indicates a gross leak pressure (GLP) of 4099 psid. This GLP is a median value for a lognormal distribution with a composite logarithmic standard deviation of 0.12, representing both uncertainty in modeling and data variability.

Leakage from flanged connections can occur at pressures below the GLP. Leak rates vary from 10 mg/sec up to about 60 mg/sec at GLP. Above GLP the flange gasket can separate from the flange surface and significant leak areas can develop. These range from 0.04 in² at 1.25*GLP to 0.14 in² at 2.0*GLP. The analysis in this report does not distinguish among the various possible leak rates. Instead the assumption is made that there is no leakage below GLP, and above GLP the system ruptures.

In developing the above pressure capacity and leak area data Impell had to make assumptions about the maximum value of the ratio of incremental gasket load to total pressure load. The authors of the Impell report caution that because high-strength flange bolts are used at the study plant, some flange inelastic behavior can be expected at higher pressures. This may affect the calculated leak rates. Therefore, they caution that additional study of flange behavior may be required if leakage through flanged connections is found to be a significant contributor to risk.

Heat Exchangers

Appendix I examined several failure modes for the RHR system heat exchangers. The modes used in this analysis are tube-side cylinder failure, asymmetric head buckling, and head plastic collapse. Appendix I notes that neither head buckling nor plastic collapse automatically results in a leak, and states that crack formation is only expected about 20% of the time that buckling or plastic collapse occurs. The median failure capacities and variabilities are summarized in Table 1.

Valves

Impell postulated three failure modes for valves within the RHR system. They include failure of the valve body, failure of the stem packing, and

failure of the bolted bonnet. Impell judged that failure of the valve body would not be likely because the wall thickness of the valve body is greater than the surrounding piping. Packing failure was dismissed because overpressurization would tend to improve the packing seal, and failure, if it were to occur, would result in a negligible leak rate. Failure of the bolted bonnet was considered to be the only credible failure mode for valves and was evaluated for a number of valves within the safety systems. The fragility data for these valves is provided in Table 4-1 of Appendix I. Of the valves affected by the RHR system pressurization scenario, all have leak capacities well in excess of 2485 psig and are therefore not included in the component failure calculations.

Calculational Methods

Determination of system rupture probability requires "summing" the probability of component failure over the number of components in the system. The following discussion presents the method used to calculate single component failure probabilities and the resulting system rupture probability.

Convolution

The pressure capacity of a system component is characterized in terms of a random variable P_f , which is distributed lognormally with a median and logarithmic standard deviation. The probability density function of P_f is

$$f(p_f) = \frac{1}{\sqrt{2\pi}\beta_c p_f} \exp\left[-\frac{(\ln p_f - \mu_f)^2}{2\beta_c^2}\right]$$

where P_f takes on values ≥ 0 and μ_f is the natural logarithm of the median failure pressure.¹

The system pressure is characterized in terms of a normally distributed random variable P_s , with a probability density function given by

¹In the following discussion random variables will be denoted with capital letters and values assumed by random variables will be denoted with lowercase letters, e.g., P_f is the random variable representing failure pressure and p_f represents actual values of the failure pressure.

$$g(p_s) = \frac{1}{\sqrt{2\pi}\sigma_s} \exp\left[-\frac{(p_s - \mu_s)^2}{2\sigma_s^2}\right]$$

where μ_s is the mean (median) system pressure and σ_s is the standard deviation.

A random variable representing the component failure margin is then defined as

$$Z = P_f - P_s.$$

The probability density function of Z is obtained from P_f and P_s , by convolution, as follows

$$\begin{aligned} h(z) &= \int_0^{\infty} f_{P_s}(p_f - z) f_{P_f}(p_f) dp_f \\ &= \int_0^{\infty} \frac{1}{2\pi\sigma_s\beta_c p_f} e^{-\frac{(p_f - z - \mu_s)^2}{2\sigma_s^2}} e^{-\frac{(\ln p_f - \beta_c)^2}{2\beta_c^2}} dp_f. \end{aligned}$$

The probability of system failure is obtained by finding the probability that $Z \leq 0$:

$$P(\text{failure}) = P(Z \leq 0) = \int_{-\infty}^0 h(z) dz.$$

It is apparent from the above expressions that $P(\text{failure})$ is not easily obtained. The approach used to evaluate the above expression is described in the following section.

Event Tree Formulation

The above failure probability expression is evaluated by applying a Monte Carlo simulation approach using the EVNTRE code developed at Sandia

[F-1]. A simple EVNTRE model is constructed that questions component failure by asking if system pressure, P_s , is greater than the component failure pressure, P_f . If it is, the component is considered to have failed. Each component is questioned in turn in each run through the event tree, and 10,000 separate Monte Carlo runs are made through the tree. The number of failures counted during the sampling process is summarized in the EVNTRE frequency output file. The EVNTRE code requires, as input, sample sets for each distribution used in the evaluation. Sample sets for this analysis were generated using the Fortran 77 program provided in Listing 6.

The EVNTRE model for evaluating each component in the RHR system is provided in Listing 2. Listing 1 provides the program control keywords required, while Listing 3 provides the required pointer data file. Because of its size (1.4 Megabytes), only one file required by the EVNTRE program is not included in the listings that follow. That file is the sample set data file generated by the program provided in Listing 6. The results from the EVNTRE runs are next applied to determine system rupture probability. That is the subject of the next section.

Fault Tree Formulation

The probability of system rupture is calculated from the Boolean union of the individual component failure probabilities as expressed in the following equation

$$P(\text{component failure}) = \bigcup_{i=1}^n P_{f_i}$$

where P_{f_i} is the failure probability of the i^{th} component in the system.

Utilizing a fault tree approach to evaluate the above expression provides the optional capability of performing uncertainty studies with a minimum of additional effort. The IRRAS program [F-2] and the fault tree shown in Figure 1 are used to solve for $P(\text{rupture})$. Note that flange failure is not included in the fault tree formulation. This is because the probability of flange failure is two orders of magnitude smaller than the next highest probability included in the tree, and three orders of magnitude

smaller than the most probable failure. The results obtained using the above models are summarized in the following section.

Figure 1. IRRAS fault tree model used to evaluate RHR system rupture probability.

Results

The results from the component failure analysis are provided in Table 2, along with the IRRAS-calculated system rupture probability. Three cases are provided to illustrate the effect of reducing pressure fragility uncertainties. Case 1 uses the nominal uncertainty parameters (β_c values) provided in Appendix I. Cases 2 and 3 are sensitivity studies in which the β_c values for each component are reduced by a fixed percentage such that the pipe failure β_c values are .30 and .26, respectively. The results show a very high (98%) probability of system rupture that is dominated by RHR system heat exchanger tube-side cylinder failure. They also demonstrate that further reduction in the uncertainty associated with the pressure fragility analysis will not have a significant affect on the overall system rupture probability.

Table 2. RHR system component rupture probabilities

Component	Failure Probability Case 1	Failure Probability Case 2	Failure Probability Case 3
8 inch sched. 20 piping	4.07E-01	3.87E-01	3.70E-01
12 inch sched. 40 piping	2.70E-01	2.32E-01	2.00E-01
14 inch std wall piping	4.59E-01	4.51E-01	4.44E-01
8 inch 600 psi flanges	1.00E-04	0.00E+00	0.00E+00
Hx cylinder	6.31E-01	6.52E-01	6.76E-01
Hx head buckling	3.35E-02	2.51E-02	2.03E-02
Hx head plastic Collapse	1.85E-01	1.92E-01	1.95E-01
System rupture probability	9.80E-01	9.81E-01	9.82E-01

References

- F-1. J. M. Griesmeyer, L. N. Smith, A Reference Manual for the Event Progression Analysis Code (EVNTRE), NUREG/CR-5174, September 1989.
- F-2. K. D. Russel, et. al., Integrated Reliability and Risk Analysis System (IRRAS) Version 2.0 User's Guide, NUREG/CR-5111, June 1990.

LISTING 1

EVNTRE Calculation Control File Listing

```

$-- Calculation Control Keywords -----
$
MODE 3                $ Specifies the calculational mode for EVNTRE.
                        1 = point estimate
                        3 = sampling mode (one vector each eval)
$
BIN                   $ Turns the binning facility on/off.
$
RUN                   $ Indicates that the tree is to be evaluated
                        after the input data has been processed.
$
KEEPCUT 1.0E-6        $ Specifies the path frequency below which a
                        path is terminated.
$-- Input File Specification Keywords -----
$
TREEIN tree.dat       $ Specifies the input file name for the
                        tree definition input file.
$
BININ  bin.dat        $ Specifies the input file name for the
                        binning and sorting information input file.
$
SAMDIN pntr.dat       $ Specifies the input file name for the
                        sample definition information input file.
$
SAM1IN mcarlo.dat     $ Specifies the input file name for the
                        first set of sample input vectors.
$-- Report Request Keywords -----
$
PRTINP               $ Turns on the annotated echo of input.
$
STATS                $ Indicates that a branch and case frequency
                        table report will be generated.
$
PRUNE                $ Causes unused cases to be dropped from the
                        branch and case frequency table.
$
NWRTBIN              $ Indicates that a binning result report will
                        be generated when the paths through the
                        tree are binned.
$
PRTCUT 1.0E-6        $ Specifies the minimum bin frequency required
                        to report a bin.
$
SAVEBIN              $ Indicates that a binning results file will
                        be generated for post-processing.
$-- Output File Specification Keywords -----

```

\$		
\$	INPOUT	echo.out
\$		\$ Specifies the output file name for the annotated echo of input.
\$		
\$	BINOUT	bin.out
\$		\$ Specifies the output file name for the binning result report.
\$		
\$	STATOUT	freq.out
\$		\$ Specifies the output file name for the branch and case frequency table.
\$		
\$	SAMROUT	post.out
\$		\$ Specifies the output file name for the post-processing file.
\$		
\$	ENDKEY	
		\$ Indicates the end of keyword input.

LISTING 2

EVNTRE Tree Input Description File Listing

ISLOCA Interfacing System Rupture Model -- RHR

```

10
NQ
1      1.000
      'MC Eval'
1 What is the pressure in the Interfacing System?
1      HiP
3      1
      1.000
8
1      7.732      $ Point estimates follow:
2      7.825      $      1Ps' for 2280 psi
3      7.958      $      1Pf' for 8 inch sched. 20
4      7.770      $      1Pf' for 12 inch sched. 40
5      8.319      $      1Pf' for 14 inch std wall
6      7.621      $      1Pf' for 8 inch 600 psi flanges
7      7.959      $      1Pf' for HX tube side cylinder
8      7.409      $      1Pf' for HX tube side head buckling
2 Does the 8 inch sched. 20 piping fail?
2      P820-F      P820-NoF
5      1      2
2      1      2
      1Ps'      1Pf'
      FUN-CMP
      EQUAL      0
      IF 1Ps' .GT. 1Pf' THEN P820-F
3 Does the 12 inch sched. 40 piping fail?
2      P1240-F      P1240-NoF
5      1      2
2      1      3
      1Ps'      1Pf'
      FUN-CMP
      EQUAL      0
      IF 1Ps' .GT. 1Pf' THEN P1240-F
4 Does the 14 inch std. wall piping fail?
2      P14S-F      P14S-NoF
5      1      2
2      1      4
      1Ps'      1Pf'
      FUN-CMP
      EQUAL      0
      IF 1Ps' .GT. 1Pf' THEN P14S-F
5 Do the 8 inch 600 psi flanges fail?
2      F8IN-F      F8IN-NoF
5      1      2
2      1      5
      1Ps'      1Pf'
      FUN-CMP
      EQUAL      0
      IF 1Ps' .GT. 1Pf' THEN F8IN-F

```

6 Does the RHR HX tube side cylinder fail?

2	NDHXC-F	NDHXC-NoF
5	1	2
2	1	6
	1Ps'	1Pf'
	FUN-CMP	
	EQUAL	0
	IF 1Ps' .GT. 1Pf' THEN NDHXC-F	

7 Does the RHR HX tube side head buckle?

2	NDHXH-B	NDHXH-NoB
5	1	2
2	1	7
	1Ps'	1Pf'
	FUN-CMP	
	EQUAL	0
	IF 1Ps' .GT. 1Pf' THEN NDHXH-F	

8 Does the RHR HX tube side head rupture, given buckling?

2	NDHXH-F1	NDHXH-NF1
2	1	2
2		
1	7	
	1	
	NDHXH-B	
	0.200	0.800
	Otherwise	
	0.000	1.000

9 Does the RHR HX tube side head experience plastic collapse?

2	NDHXH-P	NDHXH-NoP
5	1	2
2	1	8
	1Ps'	1Pf'
	FUN-CMP	
	EQUAL	0
	IF 1Ps' .GT. 1Pf' THEN NDHXH-F	

10 Does the RHR HX tube side head rupture, given collapse?

2	NDHXH-F2	NDHXH-NF2
2	1	2
2		
1	9	
	1	
	NDHXH-P	
	0.200	0.800
	Otherwise	
	0.000	1.000

LISTING 3

EVNTRE Pointer Data File Listing

ISLOCA SAMPLE RUN

10000 1

8

M1,1,1,1,A M1,1,2,1,A M1,1,3,1,A M1,1,4,1,A M1,1,5,1,A

M1,1,6,1,A M1,1,7,1,A M1,1,8,1,A

LISTING 4

EVNTRE User Function for Comparing System and Rupture Pressures

```
FUNCTION UFUN(NAME,NARG,IDARG,ARG)
```

```
*
* This user function is used for Monte Carlo analysis. If the first
* argument is larger than the second, a split fraction of 1.0 is
* returned. Otherwise 0.0 is returned. The first argument is
* the system pressure, the second is the component failure pressure.
*
```

```
IMPLICIT NONE
CHARACTER*6 NAME
INTEGER NARG, IDARG(*)
REAL*8 ARG(*), P1, P2, UFUN
```

```
*
*
IF (NAME(:3) .EQ. 'CMP') THEN
  P1 = ARG(IDARG(1))
  P2 = ARG(IDARG(2))
  IF (P1 .GT. P2) THEN
    UFUN = 1.
  ELSE
    UFUN = 0.
  ENDIF

ELSE IF (NAME(:4) .EQ. 'RPSZ') THEN
  P1 = ARG(IDARG(1))
  P2 = ARG(IDARG(2))
  UFUN = EXP(P1) / EXP(P2)
ENDIF

RETURN
END
```


LISTING 5

EVNTRE Frequency Output File Listing

CASE 1, PIPE BETA = .36

1 TREE ID: ISLOCA Interfacing System Rupture Model -- RHR

OF QUESTIONS: 10
OBSERVATIONS: 10000
FOR SERIES: ISLOCA SAMPLE RUN
SEQUENCE ID: MC Eval

```
***** QUESTION:      1  What is the pressure in the Interfacing System?
Q-TYPE/TIMES ASKED:      INDEP. INPUT PROB. INPUT PARM.          10000
  BRANCHES:              HiP
                        1
  REALIZED SPLIT:        1.000E+00

***** QUESTION:      2  Does the 8 inch sched. 20 piping fail?
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                      10000
  BRANCHES:              P820-F      P820-NoF
                        1              2
  REALIZED SPLIT:        4.065E-01 5.935E-01

***** QUESTION:      3  Does the 12 inch sched. 40 piping fail?
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                      10000
  BRANCHES:              P1240-F     P1240-NoF
                        1              2
  REALIZED SPLIT:        2.701E-01 7.299E-01

***** QUESTION:      4  Does the 14 inch std. wall piping fail?
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                      10000
  BRANCHES:              P14S-F      P14S-NoF
                        1              2
  REALIZED SPLIT:        4.591E-01 5.409E-01

***** QUESTION:      5  Do the 8 inch 600 psi flanges fail?
Q-TYPE/TIMES ASKED:      INDEP. CALC. PROB.                      10000
  BRANCHES:              F8IN-F      F8IN-NoF
                        1              2
  REALIZED SPLIT:        1.000E-04 9.999E-01

***** QUESTION:      6  Does the RHR HX tube side cylinder fail?
```

Q-TYPE/TIMES ASKED:	INDEP. CALC. PROB.	10000
BRANCHES:	NDHXC-F NDHXC-NoF	
	1 2	
REALIZED SPLIT:	6.307E-01 3.693E-01	

***** QUESTION: 7 Does the RHR HX tube side head buckle?

Q-TYPE/TIMES ASKED:	INDEP. CALC. PROB.	10000
BRANCHES:	NDHXH-B NDHXH-NoB	
	1 2	
REALIZED SPLIT:	1.674E-01 8.326E-01	

***** QUESTION: 8 Does the RHR HX tube side head rupture, given buckling?

Q-TYPE/TIMES ASKED:	DEP. INPUT PROB.	11674
BRANCHES:	NDHXH-F1 NDHXH-NF1	
	1 2	
REALIZED SPLIT:	3.348E-02 9.665E-01	

SUMMARY BY CASE

CASE NUMBER/SPLIT:	1	1.674E-01
DEPENDENCIES:	7	
REQ. BRANCHES:	1	
DESCRIPTION:	NDHXH-B	
CASE/BRANCH SPLIT:	3.348E-02	1.339E-01
CASE NUMBER/SPLIT:	2	8.326E-01
DESCRIPTION:	Otherwise	
CASE/BRANCH SPLIT:	0.000E+00	8.326E-01

***** QUESTION: 9 Does the RHR HX tube side head experience plastic collapse?

Q-TYPE/TIMES ASKED:	INDEP. CALC. PROB.	11674
BRANCHES:	NDHXH-P NDHXH-NoP	
	1 2	
REALIZED SPLIT:	9.241E-01 7.590E-02	

***** QUESTION: 10 Does the RHR HX tube side head rupture, given collapse?

Q-TYPE/TIMES ASKED:	DEP. INPUT PROB.	22469
BRANCHES:	NDHXH-F2 NDHXH-NF2	
	1 2	
REALIZED SPLIT:	1.848E-01 8.152E-01	

SUMMARY BY CASE

CASE NUMBER/SPLIT: 1 9.241E-01
DEPENDENCIES: 9

REQ. BRANCHES: 1

DESCRIPTION: NDHXH-P

CASE/BRANCH SPLIT: 1.848E-01 7.393E-01

CASE NUMBER/SPLIT: 2 7.590E-02
DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 7.590E-02

CASE 2, PIPE BETA = .30

1 TREE ID: ISLOCA Interfacing System Rupture Model -- RHR

OF QUESTIONS: 10
OBSERVATIONS: 10000
FOR SERIES: ISLOCA SAMPLE RUN
SEQUENCE ID: MC Eval

***** QUESTION: 1 What is the pressure in the Interfacing System?

Q-TYPE/TIMES ASKED: INDEP. INPUT PROB. INPUT PARM. 10000
BRANCHES: HiP
1
REALIZED SPLIT: 1.000E+00

***** QUESTION: 2 Does the 8 inch sched. 20 piping fail?

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 10000
BRANCHES: P820-F P820-NoF
1 2
REALIZED SPLIT: 3.871E-01 6.129E-01

***** QUESTION: 3 Does the 12 inch sched. 40 piping fail?

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 10000
BRANCHES: P1240-F P1240-NoF
1 2
REALIZED SPLIT: 2.315E-01 7.685E-01

***** QUESTION: 4 Does the 14 inch std. wall piping fail?

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 10000
BRANCHES: P14S-F P14S-NoF
1 2
REALIZED SPLIT: 4.507E-01 5.493E-01

***** QUESTION: 5 Do the 8 inch 600 psi flanges fail?

Q-TYPE/TIMES ASKED:	INDEP. CALC.	PROB.	10000
BRANCHES:	F8IN-F	F8IN-NoF	
	1	2	
REALIZED SPLIT:	0.000E+00	1.000E+00	

***** QUESTION: 6 Does the RHR HX tube side cylinder fail?

Q-TYPE/TIMES ASKED:	INDEP. CALC.	PROB.	10000
BRANCHES:	NDHXC-F	NDHXC-NoF	
	1	2	
REALIZED SPLIT:	6.520E-01	3.480E-01	

***** QUESTION: 7 Does the RHR HX tube side head buckle?

Q-TYPE/TIMES ASKED:	INDEP. CALC.	PROB.	10000
BRANCHES:	NDHXH-B	NDHXH-NoB	
	1	2	
REALIZED SPLIT:	1.257E-01	8.743E-01	

***** QUESTION: 8 Does the RHR HX tube side head rupture, given buckling?

Q-TYPE/TIMES ASKED:	DEP. INPUT	PROB.	11257
BRANCHES:	NDHXH-F1	NDHXH-NF1	
	1	2	
REALIZED SPLIT:	2.514E-02	9.749E-01	

SUMMARY BY CASE

CASE NUMBER/SPLIT:	1	1.257E-01
DEPENDENCIES:	7	
REQ. BRANCHES:	1	
DESCRIPTION:	NDHXH-B	
CASE/BRANCH SPLIT:	2.514E-02	1.006E-01
CASE NUMBER/SPLIT:	2	8.743E-01
DESCRIPTION:	Otherwise	
CASE/BRANCH SPLIT:	0.000E+00	8.743E-01

***** QUESTION: 9 Does the RHR HX tube side head experience plastic collapse?

Q-TYPE/TIMES ASKED:	INDEP. CALC.	PROB.	11257
BRANCHES:	NDHXH-P	NDHXH-NoP	
	1	2	
REALIZED SPLIT:	9.584E-01	4.160E-02	

***** QUESTION: 10 Does the RHR HX tube side head rupture, given collapse?
 Q-TYPE/TIMES ASKED: DEP. INPUT PROB. 22054
 BRANCHES: NDHXH-F2 NDHXH-NF2
 1 2
 REALIZED SPLIT: 1.917E-01 8.083E-01

SUMMARY BY CASE

CASE NUMBER/SPLIT: 1 9.584E-01
 DEPENDENCIES: 9

REQ. BRANCHES: 1

DESCRIPTION: NDHXH-P

CASE/BRANCH SPLIT: 1.917E-01 7.667E-01

CASE NUMBER/SPLIT: 2 4.160E-02
 DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 4.160E-02

CASE 3, PIPE BETA = .26

1 TREE ID: ISLOCA Interfacing System Rupture Model -- RHR

OF QUESTIONS: 10
 OBSERVATIONS: 10000
 FOR SERIES: ISLOCA SAMPLE RUN
 SEQUENCE ID: MC Eval

***** QUESTION: 1 What is the pressure in the Interfacing System?
 Q-TYPE/TIMES ASKED: INDEP. INPUT PROB. INPUT PARM. 10000
 BRANCHES: HiP
 1
 REALIZED SPLIT: 1.000E+00

***** QUESTION: 2 Does the 8 inch sched. 20 piping fail?
 Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 10000
 BRANCHES: P820-F P820-NoF
 1 2
 REALIZED SPLIT: 3.701E-01 6.299E-01

***** QUESTION: 3 Does the 12 inch sched. 40 piping fail?
 Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 10000
 BRANCHES: P1240-F P1240-NoF
 1 2
 REALIZED SPLIT: 1.999E-01 8.001E-01

***** QUESTION: 4 Does the 14 inch std. wall piping fail?

Q-TYPE/TIMES ASKED:	INDEP. CALC. PROB.	10000
BRANCHES:	P14S-F P14S-NoF	
	1 2	
REALIZED SPLIT:	4.436E-01 5.564E-01	

***** QUESTION: 5 Do the 8 inch 600 psi flanges fail?

Q-TYPE/TIMES ASKED:	INDEP. CALC. PROB.	10000
BRANCHES:	F8IN-F F8IN-NoF	
	1 2	
REALIZED SPLIT:	0.000E+00 1.000E+00	

***** QUESTION: 6 Does the RHR HX tube side cylinder fail?

Q-TYPE/TIMES ASKED:	INDEP. CALC. PROB.	10000
BRANCHES:	NDHXC-F NDHXC-NoF	
	1 2	
REALIZED SPLIT:	6.764E-01 3.236E-01	

***** QUESTION: 7 Does the RHR HX tube side head buckle?

Q-TYPE/TIMES ASKED:	INDEP. CALC. PROB.	10000
BRANCHES:	NDHXH-B NDHXH-NoB	
	1 2	
REALIZED SPLIT:	1.013E-01 8.987E-01	

***** QUESTION: 8 Does the RHR HX tube side head rupture, given buckling?

Q-TYPE/TIMES ASKED:	DEP. INPUT PROB.	11013
BRANCHES:	NDHXH-F1 NDHXH-NF1	
	1 2	
REALIZED SPLIT:	2.026E-02 9.797E-01	

SUMMARY BY CASE

CASE NUMBER/SPLIT:	1	1.013E-01
DEPENDENCIES:	7	
REQ. BRANCHES:	1	
DESCRIPTION:	NDHXH-B	
CASE/BRANCH SPLIT:	2.026E-02	8.104E-02
CASE NUMBER/SPLIT:	2	8.987E-01
DESCRIPTION:	Otherwise	
CASE/BRANCH SPLIT:	0.000E+00	8.987E-01

***** QUESTION: 9 Does the RHR HX tube side head experience plastic collapse?

Q-TYPE/TIMES ASKED: INDEP. CALC. PROB. 11013

BRANCHES: NDHXH-P NDHXH-NoP

1 2

REALIZED SPLIT: 9.733E-01 2.670E-02

***** QUESTION: 10 Does the RHR HX tube side head rupture, given collapse?

Q-TYPE/TIMES ASKED: DEP. INPUT PROB. 21736

BRANCHES: NDHXH-F2 NDHXH-NF2

1 2

REALIZED SPLIT: 1.947E-01 8.053E-01

SUMMARY BY CASE

CASE NUMBER/SPLIT: 1 9.733E-01

DEPENDENCIES: 9

REQ. BRANCHES: 1

DESCRIPTION: NDHXH-P

CASE/BRANCH SPLIT: 1.947E-01 7.786E-01

CASE NUMBER/SPLIT: 2 2.670E-02

DESCRIPTION: Otherwise

CASE/BRANCH SPLIT: 0.000E+00 2.670E-02

LISTING 6

Program Used to Generate EVNTRE Monte Carlo Data

* PURPOSE:

* This program calculates the normal distributions required for the
* Monte Carlo evaluation of the ISLOCA SEQ-1 scenario. The first distribution
* corresponds to system pressure. The required output for the first parameter
* is actually the natural log of the system pressure. The remaining
* distributions are log normal, and are described by a log mean and
* logarithmic std. dev. The output is written in the format required by
* the EVNTRE program.

* INPUT:

* The input is read from Standard Input (Fortran unit 5). The data set used
* for the RHR system analysis follows the Fortran source code below.
* The program first reads a sort flag (0 indicates no sorting), the number
* of samples to generate for each distribution, the number of distributions,
* and the value of the random seed. The program then reads the distribution
* index (corresponding to the EVNTRE parameter number), the sample mean, and
* the sample standard deviation. Blank lines and comment lines shown in the
* example are required, otherwise input is free format.

* OUTPUT:

* The output is the required normal distributions, and is written to file
* 'MCARLO.DAT'. The data is in the format required by the EVNTRE code for
* use as sample data.

* FILES:

Unit	Description
5	User input from console
6	Program output to console
11	Output for use as an EVNTRE sample file

* WRITTEN BY:

Main program and input processing	John Schroeder 1/11/90
Functions and subroutines	as indicated

PROGRAM NORM

IMPLICIT NONE

INTEGER I, IER, IPOINT, ISEED, ISORT, J, N, NDIS
REAL MEAN(10), STDEV(10), XX(10,10000), X(10000)

* Read data file

CALL READIN(ISEED, ISORT, N, NDIS, MEAN, STDEV)


```

*      The first distribution requires special treatment.

      CALL GENNOR(MEAN(1),STDEV(1),X,N,ISORT,ISEED,IER,IPOINT)
      DO 5  I = 1, N
          XX(1,I) = ALOG(X(I))
5      CONTINUE

*      Generate the remaining distributions

      DO 15  I = 2, NDIS
          CALL GENNOR(MEAN(I),STDEV(I),X,N,ISORT,ISEED,IER,IPOINT)
          DO 10  J = 1, N
              XX(I,J) = X(J)
10          CONTINUE
15      CONTINUE

*      Write out distributions

      OPEN(11,FILE='MCARLO.DAT',STATUS='UNKNOWN')
      DO 20  I = 1, N
          WRITE(11,25) I, NDIS, (XX(J,I), J = 1, NDIS)
20      CONTINUE
25      FORMAT(2I10, 5G12.5,/(20X,5G12.5))
      CLOSE(11)

      END

      SUBROUTINE READIN(ISEED, ISORT, N, NDIS, MEAN, STDEV)

      IMPLICIT NONE
      CHARACTER*80 LINE
      INTEGER I, IDIS, ISEED, ISORT, N, NDIS
      REAL MEAN(*), STDEV(*)

      READ(5, '(A)') LINE
      READ(5, *) ISORT, N, NDIS, ISEED

      READ(5, '(/A)') LINE
      DO 10  I = 1, NDIS
          READ(5, *) IDIS, MEAN(IDIS), STDEV(IDIS)
10      CONTINUE

      RETURN
      END

      SUBROUTINE GENNOR(AMEAN,STDEV,X,N,ISORT,ISEED,IER,IPOINT)

C      GENERATES RANDOM SAMPLE OF N NUMBERS FROM NORMAL POPULATION.
C      INPUTS
C          AMEAN = MEAN OF POPULATION
C          STDEV = STANDARD DEVIATION (= SQRT OF VARIANCE) OF POPULATION
C          N = NUMBER OF VALUES WANTED
C          ISORT = 0 IF VALUES ARE TO BE IN ORDER GENERATED
C                = 1 IF VALUES ARE TO BE SORTED INTO INCREASING ORDER
C          ISEED = INITIAL SEED, AN INTEGER IN RANGE OF INTEGERS ON

```

```

C          THE MACHINE USED.  SEE COMMENTS IN FUNCTION URAND,
C          WHERE THIS RANGE IS DEFINED BY PROGRAMMER AND CHECKED
C          BY PROGRAM.
C  OUTPUT
C      X = REAL ARRAY, DIMENSIONED TO SIZE AT LEAST N IN CALLING
C          PROGRAM.  THE RANDOM SAMPLE IS RETURNED AS X.
C      IER = 0 IF NO ERRORS RECOGNIZED.
C          = 1 IF PROBLEM IN THE TAILS OF THE NORMAL DISTRIBUTION.
C          = 2 IF URAND GOT ANSWER OUTSIDE OF [0., 1.]
C      IPOINT = THE ELEMENT OF X THAT CAUSED ERROR FLAG TO TURN ON.
C          SUBROUTINE RETURNS AS SOON AS IER > 0.

```

```

C  WRITTEN BY C. ATWOOD, DEC.1989, BASED ON EARLIER PROGRAMS
C

```

```

IMPLICIT REAL(A-H,O-Z), INTEGER (I-N)
DOUBLE PRECISION URAND
DIMENSION X(N)
logical debug
data debug /.false./

```

```

C          GENERATE THE UNIFORM SAMPLE
C

```

```

C      DO 40 I=1,N
C          X(I) = URAND(ISEED,IDUMMY)
C          if(debug) write(6,'('' uniform x(i)='',f9.6)' ) x(i)
40      CONTINUE
C          SORT VALUES INTO ASCENDING ORDER
C      IF(ISORT.NE.0) THEN
C          CALL SORT(N,X)
C          if(debug)
+      write(6,'('' sorted uniform x ='',f9.6)' ) (x(i),i=1,n)
C      ENDIF

```

```

C          CONVERT UNIFORM TO NORMAL(0,1)
C          THEN CORRECT FOR MEAN AND ST. DEV.
C

```

```

C      DO 100 I=1,N
C          P = X(I)
C          if(debug) write(6,'('' p ='',g14.6)' ) p
C          Z = ANORIN(P,IER)
C          IF(IER.GT.0) THEN
60      if(debug) write(6,60) ier, i, p, z
C          format(' ier, i, x, z =',2i4,2g14.6)
C          IPOINT = I
C          RETURN
C      ENDIF
C      X(I) = AMEAN + STDEV*Z
C      if(debug) write(6,'('' normal x(i)='',f9.6)' ) x(i)
100     CONTINUE
C  END
C  FUNCTION ANORDF(X)

```

```

c  Calculates standard normal cumulative distribution function

```

```

IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
DATA RT2INV/.7071067812/

```

```

U = ABS(X) * RT2INV
IF(X.GT.0) THEN
    ANS = 1 - ERFC(U)/2
ELSE
    ANS = ERFC(U)/2
ENDIF
c debug print
c write(6,'( '' normal cdf('',g14.6,'') ='',g14.6)' ) x, ans
ANORDF = ANS
RETURN
END
FUNCTION ANORIN(P,IER)

c Evaluates inverse normal cdf PHI-inverse(p)
c For p in tail, starts with Wichura approximation, then refines
c it N times using eq. (5.9.2) of Thisted (1988) Elements of
c Statistical Computing, Chapman and Hall.
c For p in center, uses Beasley-Springer algorithm.
IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
c Phi(XMAX) is about as close to 1 as we can get in single precision
c 1 - Phi(XMAX) = 2.9E-7
DATA XMAX/5./
DATA N/1/
c On return, IER = 1 signals input error, IER = 2 is serious error
IER = 0

IF(P.GE.1. .OR. P.LE.0.) THEN
    IER = 1
    IF(P.GT.1. .OR. P.LT.0) THEN
        IER = 2
        WRITE(6,10) p
10    FORMAT(' Input error to ANORIN('',E14.6,'',IER)' )
    ENDIF
    IF(P.LE.0) ANORIN = -XMAX
    IF(P.GE.1) ANORIN = XMAX
    RETURN
ELSE IF(P.LT..1 .OR. P.GT..9) THEN
    Z = WICHUR(P)
    DO 50 I=1,N
        CDF = ANORDF(Z)
        ARG = 2*P - CDF
        Z = WICHUR(ARG)
50    CONTINUE
    ANORIN = Z
ELSE
    ANORIN = PPND(P,IFault)
    IF(IFault.NE.0) THEN
        WRITE(6,100) P
100    FORMAT(' Error fault in ANORIN('',E14.6,'',IER)' )
        IER = 2
    ENDIF
ENDIF
RETURN
END
SUBROUTINE BINARY(IY,KOEF,INTSIZ)

```

```

C   FINDS COEFFICIENTS OF BINARY EXPANSION OF IY
C
C   IY = SUM OF ( KOEF(I) * 2**(I-1) )
C
C   WHERE SUMMATION IS FOR I FROM 1 TO INTSIZ
C   Y MUST BE .GE. 0 AND .LT. 2**INTSIZ

      IMPLICIT INTEGER(I-N)
      DIMENSION KOEF(INTSIZ)

      N = IY
      DO 50 I=1,INTSIZ
        N2 = N/2
        KOEF(I) = N - (2*N2)
        N = N2
50    CONTINUE
      IF(N.EQ.0) RETURN

C                                     ERROR
      WRITE(6,100) IY, INTSIZ, (KOEF(I),I=1,INTSIZ)
100  FORMAT(' ERROR IN SUBROUTINE BINARY'/
+         ' IY =',I20/' INTSIZ =',I5/' KOEF =',(20I3) )
      STOP
      END
      FUNCTION ERFC(X)

c   Evaluates the complementary error function at X.
c   Uses an algorithm of Press et al., 1986, Numerical Recipes:
c   The Art of Scientific Computing, Cambridge Univ. Press.
c   as presented in Section 5.10.1.1 of Thisted, 1988, Elements
c   of Statistical Computing, Chapman and Hall.
c   Calculation has relative error < 1.2E-7 for X > 0.

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DIMENSION A(10)
      DATA A/-1.26551223, 1.00002368, 0.37409196, 0.09678418,
+          -0.18628806, 0.27886807, -1.13520398, 1.48851587,
+          -0.82215223, 0.17087277/

      ARG = 1 / (1 + X/2)
      PSUM = A(1)
      TERM = 1
      DO 20 I=2,10
        TERM = TERM * ARG
        PSUM = PSUM + TERM*A(I)
20    CONTINUE
      EARG = -(X**2) + PSUM
      ERFC = ARG * EXP(EARG)
      RETURN
      END
      FUNCTION PPND(P,IFALT)

c   Algorithm AS 111 Applied Statistics, 1977, Vol. 26, No. 1.
c   by J. D. Beasley and S. G. Springer
c   Used for inverse of normal cdf, in middle portion of distr.

```

```

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DATA ZERO/0./, HALF/0.5/, ONE/1./
      DATA SPLIT/0.42/
      DATA A0 / 2.50662 82388 4/
      DATA A1 /-18.61500 06252 9/
      DATA A2 / 41.39119 77353 4/
      DATA A3 /-25.44106 04963 7/
      DATA B1 / -8.47351 09309 0/
      DATA B2 / 23.08336 74374 3/
      DATA B3 /-21.06224 10182 6/
      DATA B4 / 3.13082 90983 3/
C      HASH SUM AB 143.70383 55807 6
      DATA C0 / -2.78718 93113 8/
      DATA C1 / -2.29796 47913 4/
      DATA C2 / 4.85014 12713 5/
      DATA C3 / 2.32121 27685 8/
      DATA D1 / 3.54388 92476 2/
      DATA D2 / 1.63706 78189 7/
C      HASH SUM CD 17.43746 52092 4

      IFAULT = 0
      Q = P - HALF
      IF(ABS(Q) .LE. SPLIT) THEN
        R = Q*Q
        PPND = Q * (((A3 * R + A2) * R + A1) * R + A0) /
+          (((B4 * R + B3) * R + B2) * R + B1) * R + ONE)
      ELSE
        R = P
        IF(Q .GT. ZERO) R = ONE - P
        IF(R .LE. ZERO) GO TO 800
        R = SQRT(-LOG(R))
        PPND = (((C3 * R + C2) * R + C1) * R + C0) /
+          ((D2 * R + D1) * R + ONE)
        IF(Q .LT. ZERO) PPND = -PPND
      ENDIF
C      debug prints
C      write(6,100) p,q,r,ppnd
C 100 format(' in ppnd, p, q, r, ppnd =',4g14.6)
      RETURN

800 CONTINUE
      IFAULT = 1
      PPND = ZERO
      RETURN
      END
      SUBROUTINE SORT(N,RA)

C      Implementation of the heapsort algorithm given in
C      Press et al., Numerical Recipes, Cambridge Univ. Press, 1986
C      On input, RA is unsorted. On output, RA is in ascending order.

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DIMENSION RA(N)

      IF (N.LE.0) RETURN

```

```

L=N/2+1
IR=N
10 CONTINUE
  IF(L.GT.1)THEN
    L=L-1
    RRA=RA(L)
  ELSE
    RRA=RA(IR)
    RA(IR)=RA(1)
    IR=IR-1
    IF(IR.EQ.1)THEN
      RA(1)=RRA
      RETURN
    ENDIF
  ENDIF
  I=L
  J=L+L
20 IF(J.LE.IR)THEN
  IF(J.LT.IR)THEN
    IF(RA(J).LT.RA(J+1))J=J+1
  ENDIF
  IF(RRA.LT.RA(J))THEN
    RA(I)=RA(J)
    I=J
    J=J+J
  ELSE
    J=IR+1
  ENDIF
  GO TO 20
ENDIF
RA(I)=RRA
GO TO 10
END
DOUBLE PRECISION FUNCTION URAND(IY,M2RET)

```

```

C  UNIFORM RANDOM NUMBER GENERATOR, TAKEN FROM FORSYTHE, MALCOLM AND
C  MOLER (1977) 'COMPUTER METHODS FOR MATHEMATICAL COMPUTATIONS',
C  PRENTICE HALL. IT IS BASED ON SUGGESTIONS BY KNUTH (1969).
C  M2RET=M2 IS RETURNED ON FIRST CALL, FOR USE IN ADVISING
C  USER AS TO ALLOWABLE SEEDS. IF THIS FIRST CALL IS ONLY
C  TO FIND M2, USE ANY IY, FOR EXAMPLE 0.
C  IY SHOULD BE INITIALIZED TO AN ARBITRARY INTEGER PRIOR TO THE
C  FIRST CALL THAT SERIOUSLY WANTS A RANDOM NUMBER, AND IY
C  SHOULD NOT BE ALTERED BETWEEN SUBSEQUENT CALLS.
C  VALUES OF URAND WILL BE RETURNED IN THE INTERVAL (0,1).

```

```

  IMPLICIT INTEGER(I-N)
  IMPLICIT DOUBLE PRECISION (A-H,O-Z)
C  ON IBM PC, MAX INTEGER IS 2**15 - 1, WITH 4-BYTE INTEGERS,
C  SMALLER WITH 2-BYTE INTEGERS.
  PARAMETER(INTMAX=31)
  DIMENSION KOEFA(INTMAX), KOEFY(INTMAX)
  DATA M2/0/, ITWO/2/
  IF(M2 .NE. 0) GO TO 20

```

```

C                                     FIRST ENTRY
C   COMPUTE MACHINE WORD LENGTH
C   INTSIZ = NUMBER OF BITS IN HOST MACHINE INTEGER WORD
C           E.G. 31 IF INTEGER*4, 15 IF INTEGER*2
C   LARGEST POSSIBLE INTEGER IS (M2 - 1) + M2
      M = 1
      INTSIZ = 0
10  M2 = M
      INTSIZ = INTSIZ + 1
      M = ITWO * M2
      IF(M .GT. M2) GO TO 10
      HALFM = M2
      M2RET = M2

      IF(INTSIZ.GT.INTMAX) THEN
C                                     ERROR IN DIMENSION
          II = INTMAX
          WRITE(6,15) II, INTSIZ
15    FORMAT(' DIMENSIONS TOO SMALL IN FUNCTION URAND'/
+          ' CHANGE THE STATEMENT'/
+          '       PARAMETER(INTMAX=',I2,')'/
+          ' TO THE STATEMENT'/
+          '       PARAMETER(INTMAX=',I2,')'/
+          ' AND RECOMPILE THE PROGRAM')
          STOP
      ENDIF

C   COMPUTE MULTIPLIER AND INCREMENT
C   FOR LINEAR CONGRUENTIAL METHOD
      IA = 8 * INT( HALFM * DATAN(1.DO) / 8.DO ) + 5
      IC = 2 * INT( HALFM * (0.5DO - DSQRT(3.DO)/6.DO ) ) + 1
      MIC = (M2 - IC) + M2

C   FOR BRUTE FORCE MODULAR ARITHMETIC, FIND BINARY COEFFS FOR IA
      CALL BINARY(IA,KOEFA,INTSIZ)

C   S IS THE SCALE FACTOR FOR CONVERTING TO FLOATING POINT
      S = 0.5 / HALFM

C                                     COMPUTE THE NEXT RANDOM NUMBER
20  CONTINUE

C   FIND IY = IY*IA (MOD 2**INTSIZ)
C   IF MACHINE TREATS ORDINARY INTEGER MULTIPLICATION OVERFLOW
C   BY TAKING REMAINDER (MOD 2**INTSIZ), REPLACE NEXT GROUP OF
C   STATEMENTS BY IY = IY * IA

      CALL BINARY(IY,KOEFY,INTSIZ)
      IANS = 0
      IMULT = 1
      DO 60 I=1,INTSIZ
          IF(I.GT.1) IMULT = IMULT * 2
          IF(KOEFA(I).EQ.0) GO TO 60
          JMULT = IMULT
          DO 40 J=1,INTSIZ+1-I
              IF(J.GT.1) JMULT = JMULT * 2

```

```

        IF(KOEFY(J).EQ.0) GO TO 40
        INEW = IANS + JMUL
        IF(INEW.LT.IANS) INEW = ((IANS-M2)-M2)+JMUL
        IANS = INEW
40      CONTINUE
60      CONTINUE
      IY = IANS

C      THE NEXT STATEMENT IS FOR COMPUTERS THAT DO NOT ALLOW
C      INTEGER OVERFLOW ON ADDITION
      IF(IY .GT. MIC) IY = (IY - M2) - M2
      IY = IY + IC

C      THE NEXT STATEMENT IS FOR COMPUTERS WHERE THE WORD LENGTH
C      FOR ADDITION IS GREATER THAN FOR MULTIPLICATION
      IF(IY/2 .GT. M2) IY = (IY - M2) - M2

C      THE NEXT STATEMENT IS FOR COMPUTERS WHERE INTEGER OVERFLOW
C      AFFECTS THE SIGN BIT
      IF(IY .LT. 0) IY = (IY + M2) + M2

      URAND = S * IY
      RETURN
      END
      FUNCTION WICHUR(P)

c      Approximates PHI-inverse(p), the normal value corresponding to a
c      tail probability 1 - p.  If p > .9, it has at least 2-digit
c      accuracy.  Presented as Algorithm 5.10.1 (due to Wichura)
c      by Thisted (1988), Elements of Statistical Computing,
c      Chapman and Hall.

      IMPLICIT REAL(A-H,O-Z), INTEGER(I-N)
      DATA PI2/6.283185308/

      ISIGN = 1
      TAILP = 1 - P
      IF(TAILP.GT..5) THEN
        TAILP = P
        ISIGN = -1
      ENDIF
      V = -2 * LOG(TAILP)
      X = LOG(PI2*V)
      T = ( ( (-14 + 6*X - X**2) / (2*V) + (2-X)) / V + X) / V
      ANS = SQRT(V*(1-T))
c      debug print
c      write(6,('( tail prob =',g14.6)') tailp
c      write(6,('( v,x,t =',3g15.6)') v, x, t
c      write(6,('( normal quantile =',g14.6)') ans
      WICHUR = ISIGN * ANS
      RETURN
      END
      SUBROUTINE SEEDIN(ISEED,IOU5,IOU6,IOSAVE,FNSAVE,FN5)

C      GETS INITIAL SEED

```



```

C      INPUTS
C      IOU5 = (INTERACTIVE) INPUT UNIT, NORMALLY 5
C      FN5 = NAME OF INPUT UNIT, 'CON' ON PC
C      IOU6 = (INTERACTIVE) OUTPUT UNIT, NORMALLY 6
C      ISAVE = FILE UNIT FOR SAVING SEED
C      FNSAVE = FILE NAME CORRESPONDING TO ISAVE
C              FNSAVE IS A CHARACTER VARIABLE.
C      RETURNED VALUES
C      ISEED = THE SEED SELECTED
C
C      FILE-HANDLING STATEMENTS ARE FOR A PC.
C
C      SO-CALLED RANDOM NUMBERS ARE SIMPLY ENORMOUSLY LONG SEQUENCES
C      OF DETERMINED NUMBERS THAT HAVE NO APPARENT RELATION TO
C      THE NEIGHBORING NUMBERS IN THE SEQUENCE.
C      THE SEED IDENTIFIES A POINT IN THE SEQUENCE.
C      THIS SUBROUTINE LOOKS AT THE FILE 'FNSAVE' TO SEE WHAT THE
C      NEXT LOCATION IN THE SEQUENCE IS, FROM THE LAST TIME THE
C      PROGRAM WAS CALLED. THIS IS THE DEFAULT. USER MAY ENTER
C      ANY SEED OTHER THAN THE DEFAULT.

```

```

DOUBLE PRECISION URAND
IMPLICIT REAL(A-H,O-Z), INTEGER (I-N)
CHARACTER*80 KARD
CHARACTER*(*) FNSAVE, FN5
COMMON /FILES/ IO5,IO6
DATA NKARD/80/

```

```

      IO5 = IOU5
      IO6 = IOU6
c      Ask URAND for range of possible seeds
      IDUM = 0
      VALUE = URAND(IDUM,M2)
      NBIG = (M2 - 1) + M2
10  WRITE(IO6,20) NBIG
20  FORMAT(' Please enter seed for random number generation, '//
+        ' an integer from 0 to',I15/
+        ' RETURN for default'/' ->')
      CALL RDKARD(KARD,NKARD,IGOT,FN5)
      IF(IGOT .EQ. 0) GO TO 40
      IFIRST = 1
      CALL RDINT(KARD,IFIRST,NKARD,INEXT,ISEED)
      IF(INEXT.LE.0) GO TO 10
      RETURN

40  OPEN(UNIT=IOSAVE,FILE=FNSAVE,STATUS='OLD',ERR=50)
      READ(IOSAVE,*,END=45,ERR=45) ISEED
      CLOSE(IOSAVE)
      RETURN

45  CLOSE(IOSAVE)
50  WRITE(IO6,60)
60  FORMAT(' File with default seed is not present',
+        ' or has wrong format')
70  WRITE(IO6,80) NBIG

```

```

80 FORMAT(' Please enter integer from 0 to',I15/' ->')
CALL RDKARD(KARD,NKARD,IGOT,FN5)
IF(IGOT .EQ. 0) GO TO 70
IFIRST = 1
CALL RDINT(KARD,IFIRST,NKARD,INEXT,ISEED)
IF(INEXT .LE. 0) GO TO 70
RETURN

END
SUBROUTINE RDINT(KARD,IFIRST,ILAST,INEXT,IAN5)

C
C READS NEXT INTEGER ENTERED ON KARD
C RETURNS IAN5 = INTEGER,
C         INEXT > 1 IF INTEGER FOUND, = -1 OTHERWISE
C

IMPLICIT REAL (A-H,O-Z)
CHARACTER*(*) KARD
CHARACTER*5 FMT
CHARACTER*1 K, KP
LOGICAL INTEGR
COMMON /FILES/ I05,I06

C
INTEGR(K) = ICHAR(K).GE.48 .AND. ICHAR(K).LE.57
THESE VALUES OF ICHAR WORK IN MICROSOFT PC FORTRAN
AND IBM PROFESSIONAL FORTRAN

C
INEXT = -1
IFF = MAX(IFIRST,1)
IF(IFF.GT.ILAST) GO TO 500
ISCAN = 0
NCHAR = 0

C
DO 150 I=IFF,ILAST
  K = KARD(I:I)
110  CONTINUE
  IF(ISCAN.EQ.0) THEN
C                                     AWAITING INTEGER
    IF(.NOT.INTEGR(K)) GO TO 150
    ISCAN = 1
    ISTART = I
C                                     PRECEDED BY MINUS?
    IM = I - 1
    IF(IM.GE.IFF) THEN
      KP = KARD(IM:IM)
      IF(KP.EQ.'-') THEN
        ISTART = IM
        NCHAR = NCHAR + 1
      ENDIF
    ENDIF
    GO TO 110
  ELSE
C                                     AWAITING NON-INTEGER
    IF(.NOT.INTEGR(K)) GO TO 200
    NCHAR = NCHAR + 1
    INEXT = I + 1
  
```

```

        ENDIF
150    CONTINUE
        I = ILAST + 1
200    I = I - 1
        IF(ISCAN.EQ.0) GO TO 500
        IF(NCHAR.LE.9) WRITE(FMT,210) NCHAR
        IF(NCHAR.GT.9) WRITE(FMT,220) NCHAR
210    FORMAT( '(I',I1,')' )
220    FORMAT( '(I',I2,')' )
        READ(KARD(ISTART:I),FMT,ERR=300,END=300) IANS
        GO TO 500
C
                                TROUBLE
300    WRITE(IO6,310)
310    FORMAT(' Trouble reading line.  Line is'//
*      ' .....1.....2.....3.....4.....5',
*      ' .....6.....7.....8')
        WRITE(IO6,320) KARD
320    FORMAT(1X,A80/)
        INEXT = -1
C
500    CONTINUE
        RETURN
        END
        SUBROUTINE RDKARD(KARD,NCHARS,IGOT,FN5)
C
C      READS NCHARS OF INPUT TO KARD
C      RETURNS IGOT=0 IF BLANK OR EOF, RETURNS IGOT=1 OTHERWISE
C
        IMPLICIT INTEGER(I-N)
C      MICROSOFT FORTRAN DOES NOT ALLOW CHARACTER*(*)
        CHARACTER*(*) KARD, FN5
        CHARACTER*4 PROMPT
        COMMON /FILES/ IO5,IO6
        DATA PROMPT/' -> '/

        IGOT = 1
        READ(IO5,'(A)',END=100) KARD
40    FORMAT(A,A)
        DO 50 I=1,NCHARS
            IF(KARD(I:I).NE.' ') GO TO 200
50    CONTINUE
        GO TO 150

100    CALL REOPEN(FN5)
150    IGOT = 0
200    CONTINUE
        RETURN
        END
        SUBROUTINE REOPEN(FN5)
C
        COMMON /FILES/ IO5,IO6
        CHARACTER*(*) FN5

        CLOSE(IO5)
        OPEN (UNIT=IO5,FILE=FN5)

```

```

RETURN
END
SUBROUTINE SEEDOU(ISEED,IOU6,IOSAVE,FNSAVE)

C   WRITES NEXT SEED TO SPECIFIED FILE
C   INPUTS
C       ISEED = VALUE OF ISEED PRODUCED BY LAST CALL OF URAND.
C       IO6 = (INTERACTIVE) OUTPUT UNIT, NORMALLY 6
C       IOSAVE = FILE UNIT FOR SAVING SEED
C       FNSAVE = FILE NAME CORRESPONDING TO IOSAVE
C               FNSAVE IS A CHARACTER VARIABLE.
C
C   FILE-HANDLING STATEMENTS ARE FOR A PC.
C
C   IMPLICIT DOUBLE PRECISION (A-H,O-Z)
C   IMPLICIT INTEGER (I-N)
C   CHARACTER*(*) FNSAVE
C
C                                     WRITE SEED FOR NEXT RUN
C
200 OPEN(UNIT=IOSAVE,FILE=FNSAVE,IOSTAT=IOS,ERR=800)
   WRITE(IOSAVE,'(I20)') ISEED
   CLOSE(IOSAVE)
   GO TO 900

C                                     ERROR
800 CONTINUE
   WRITE(IOU6,850) FNSAVE
850 FORMAT(' Error in opening ',A)
   WRITE(IOU6,860) IOS
860 FORMAT(' IOSTAT =',I10)

C                                     DONE
C
900 CONTINUE
END

```

-- Input Data Listing (Case 1) --

ISORT,	N,	NDIS,	ISEED
0,	10000,	8,	1234567
IDIS,	MEAN,	STDEV,	COMMENTS
1,	2291.,	50.,	System pressure
2,	7.825,	.36,	8 inch sched. 20 piping
3,	7.958,	.36,	12 inch sched. 40 piping
4,	7.770,	.36,	14 inch std wall piping
5,	8.319,	.12,	8 inch 600 psi flanges
6,	7.621,	.33,	HX tube side cylinder
7,	7.959,	.23,	HX tube side head buckling
8,	7.409,	.23,	HX tube side head plastic collapse

-- Input Data Listing (Case 2) --

ISORT,	N,	NDIS,	ISEED
0,	10000,	8,	1234567
IDIS,	MEAN,	STDEV,	COMMENTS
1,	2291.,	50.,	System pressure
2,	7.825,	.30,	8 inch sched. 20 piping
3,	7.958,	.30,	12 inch sched. 40 piping
4,	7.770,	.30,	14 inch std wall piping
5,	8.319,	.10,	8 inch 600 psi flanges
6,	7.621,	.28,	HX tube side cylinder
7,	7.959,	.19,	HX tube side head buckling
8,	7.409,	.19,	HX tube side head plastic collapse

-- Input Data Listing (Case 3) --

ISORT,	N,	NDIS,	ISEED
0,	10000,	8,	1234567
IDIS,	MEAN,	STDEV,	COMMENTS
1,	2291.,	50.,	System pressure
2,	7.825,	.26,	8 inch sched. 20 piping
3,	7.958,	.26,	12 inch sched. 40 piping
4,	7.770,	.26,	14 inch std wall piping
5,	8.319,	.09,	8 inch 600 psi flanges
6,	7.621,	.24,	HX tube side cylinder
7,	7.959,	.17,	HX tube side head buckling
8,	7.409,	.17,	HX tube side head plastic collapse

Appendix G

ISLOCA Consequence Analysis

ISLOCA Consequence Analysis

1.0 INTRODUCTION

This appendix analyzes the effect of decontamination factors (DFs) on the offsite consequences of an ISLOCA accident sequence. The fission product source terms in this analysis are those generated for the Event V accident sequences at the Sequoyah plant for the June 1989 draft of NUREG-1150. In using these ISLOCA accident sequences, source term release information was obtained from the SEQSOR program and partitioned into a small number of source term groups using PARTITION. The consequences of these releases were calculated with the MELCOR Accident Consequence Code System (MACCS), using the Surry plant site input decks developed for the June 1989 draft of NUREG-1150. For this report consequences were calculated for the base case sequences and eight sensitivity studies in which the effect of varying the DF over a range of values was examined. Two additional sensitivities examined the effect of increasing the release elevation from the base case value of 0.0 m to 10.0 m.

2.0 ISLOCA ACCIDENT SEQUENCE DEFINITION AND SELECTION

In the NUREG-1150 terminology, the ISLOCA sequences at Sequoyah are defined by plant damage state (PDS) Group 4 - V. The accident sequences in this PDS group are initiated by failure of the check valves between the reactor coolant system (RCS) and the low pressure injection system (LPIS). The LPIS is then overpressurized and ruptures, resulting in a release path from the RCS to the auxiliary building, bypassing the containment. For Sequoyah, there are two types of V sequences: wet and dry. For the wet V sequences, the fire sprays in the auxiliary building scrub the release. The dry V sequences are not scrubbed by the fire sprays.

The starting point of the analysis in this report was the results obtained from analyzing the Sequoyah V sequence, PDSG 4, accident progression event tree (APET). This APET was analyzed as part of the Containment Performance Improvement (CPI) project studying improvements for the ice condenser containment. Therefore, reanalyzing this APET was not necessary as

the results were readily available. Using the NUREG-1150 methodology, this APET was evaluated 200 times, once for each latin hypercube sample (LHS). Each evaluation or observation resulted in hundreds of different accident sequences. For each observation, the accident sequences were sorted and grouped into a smaller number of similar accident progression bins (APBs). Because the APET analyzed the wet and dry V sequences together, each observation contained both scrubbed (wet) and unscrubbed (dry) APBs. However, for this analysis, the consequences of the wet and dry sequences were calculated separately. Therefore, the APET analysis results were manually edited to separate the dry APBs from the wet APBs. The total number of APBs for all observations was 2,642 and 2,941 for the dry and wet V sequences, respectively. The number of unique APBs over all observations was 42 and 46, respectively. Listings of the unique APBs for the dry and wet V sequences are given in Tables G.1 and G.2. For the sake of brevity, listings of the APBs over all 200 observations are not provided here, but have been stored on magnetic tape and can be made available upon request.

Table G.1 Listing of the unique APBs over all 200 LHS observations for the dry V sequences at Sequoyah.

AHADBCAADAAAAA
 AHADBCAADAAAAB
 AHADBCAADAAABBB
 AHADBCAADBAAAA
 AHADBCAADBAAAAB
 AHADBCAADBABBB
 AHADBCAADCAAAA
 AHADBCAADCAAAB
 AHADBCAADCAAAD
 AHADBCAADCABBA
 AHADBCAADCABBB
 AHADBCAADDAAAA
 AHADBCAADDAAAAB
 AHADBCAADDAAAAD
 AHADBCAADDABBA
 AHADBCAADDABBB
 AHADBCABDAAAAA
 AHADBCABDAAAAB
 AHADBCABDAABBB
 AHADBCABDBAAAA
 AHADBCABDBAAAAB
 AHADBCABDBABBB
 AHADBCABDCAAAA
 AHADBCABDCAAAB
 AHADBCABDCAAAD
 AHADBCABDCABBA
 AHADBCABDCABBB
 AHADBCABDDAAAA
 AHADBCABDDAAAAB
 AHADBCABDDAAAAD
 AHADBCABDDABBA
 AHADBCABDDABBB
 AHADDCAADBBACA
 AHADDCAADBBACB
 AHADDCAADBBACD
 AHADDCAADBBBCA
 AHADDCAADBBBCB
 AHADDCABDDBACA
 AHADDCABDDBACB
 AHADDCABDDBACD
 AHADDCABDDBBCA
 AHADDCABDDBBCB

Table G.2 Listing of the unique APBs over all 200 LHS observations for the wet V sequences at Sequoyah.

BHADBCAADAAAAA
 BHADBCAADAAAAB
 BHADBCAADAAABBA
 BHADBCAADAAABBB
 BHADBCAADBAAAA
 BHADBCAADBAAAAB
 BHADBCAADBABBA
 BHADBCAADBABBB
 BHADBCAADCAAAA
 BHADBCAADCAAAAB
 BHADBCAADCAAAD
 BHADBCAADCABBA
 BHADBCAADCABBB
 BHADBCAADDAAAA
 BHADBCAADDAAAAB
 BHADBCAADDAAAD
 BHADBCAADDABBA
 BHADBCAADDABBB
 BHADBCABDAAAAA
 BHADBCABDAAAAB
 BHADBCABDAABBA
 BHADBCABDAABBB
 BHADBCABDBAAAA
 BHADBCABDBAAAAB
 BHADBCABDBABBA
 BHADBCABDBABBB
 BHADBCABDCAAAA
 BHADBCABDCAAAB
 BHADBCABDCAAAD
 BHADBCABDCABBA
 BHADBCABDCABBB
 BHADBCABDDAAAA
 BHADBCABDDAAAAB
 BHADBCABDDAAAD
 BHADBCABDDABBA
 BHADBCABDDABBB
 BHADDCAADBACAA
 BHADDCAADBACB
 BHADDCAADBACD
 BHADDCAADBBBCA
 BHADDCAADBBBCB
 BHADDCAADBBBAC
 BHADDCAADBBBACB
 BHADDCAADBBBACD
 BHADDCAADBBBBCA
 BHADDCAADBBBBCB

3.0 SOURCE TERM ANALYSIS

3.1 Source Term Description

The next step in the consequence analysis was to determine the source term release information for each APB in each observation. The source term for a given bin contains the release fractions for nine radionuclide classes released during the early and late phases of the accident, and additional information about the release timings, the energy of the releases, and the height of the releases. The nine radionuclide classes are: noble gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium.

The source term analysis was performed with the SEQSOR code. SEQSOR is a parametric computer code used to calculate the source terms for each APB in each observation for Sequoyah. In order to study the effect of varying the V sequence DF, minor modifications were made to the SEQSOR source code and its input files. These modifications are described below.

3.2 Decontamination Factors

For the dry V sequences, the base case DF for all release classes had a uniform value of 1.0 and the release elevation was 0.0 m. The sensitivity analysis involved calculating the source terms with uniform-valued DFs of 5, 10, 50, and 100 for all release classes except the noble gases, for which the DF remained at the base case value of 1.0. For the first two calculations, with DFs of 5 and 10, the release elevation remained at the base case value of 0.0 m. For the last two calculations, with DFs of 50 and 100, the release elevation was specified as 10.0 m. For the dry V sequences, the DFs are specified within the SEQSOR code. Therefore, minor code modifications were required to model the DFs used in the sensitivity analyses. A sample of these code modifications is listed in Table G.3 for the dry V sequence with a DF of 5.0. The release elevation parameter is also specified within the SEQSOR code. The SEQSOR code modification required to specify different elevations is listed in Table G.4.

Table G.3 Listing of SEQSOR code modifications for the dry V sequence with
DF = 5.0.

Modification 1:

```
C DJP
C      VDF=1.0
C      VDF=5.0
C DJP
```

Modification 2:

```
C DJP
C      DFE=1.0
C      DFE=5.0
C DJP
```

Modification 3:

```
C DJP
C      FOR V-SEQUENCE WITH WATER:
C      IF(CHH.EQ.'B')THEN
C          DFE=VDF
C          DO 22621 ISP=2,NSP
C              DFL(ISP)=AMAX1(VDF,VPS(ISP))
C2621  CONTINUE
C DJP
C      FOR DRY V-SEQUENCE:
C      IF(CHH.EQ.'A')THEN
C          DFE=VDF
C          DO 22621 ISP=2,NSP
C              DFL(ISP)=AMAX1(VDF,VPS(ISP))
22621  CONTINUE
C DJP
```

Table G.4 Listing of SEQSOR code modifications required to specify a release elevation of 10.0 m.

Modification 1:

```
C DJP
C      ELEV = 0.
      ELEV = 10.0
C DJP
```

Table G.5 Distribution VDF for the base case analysis of the wet V sequences as specified in the SEQSOR input file ISTDAT.DAT.

0%	1%	5%	25%	50%	75%	95%	99%	100%
5.1E+03	4.5E+03	4.1E+03	1.3E+02	6.2E+00	3.0E+00	1.8E+00	1.7E+00	1.6E+00

For the wet V sequences, the base case DF was specified as a distribution to be sampled upon. This distribution is input into SEQSOR as the parameter VDF in the SEQSOR input file, ISTDAT.DAT. The distribution VDF is given in Table G.5 for the base case analysis of the wet V sequences. As for the dry V sequences, sensitivity analyses were performed with uniform-valued DFs of 5, 10, 50, and 100 for all release classes except the noble gases, for which the DF remained at the base case value of 1.0. For the first two sensitivity calculations, with DFs of 5 and 10, the release elevation remained at the base case value of 0.0 m. The calculations with DFs of 50 and 100 were performed with the release elevation specified both at the base case value of 0.0 m and at 10.0 m. No SEQSOR code modifications were necessary for these wet V sequences. Instead, the distribution of parameter VDF in the SEQSOR input file ISTDAT.DAT was replaced with a uniform-valued DF.

3.3 Partitioning Into Source Term Groups

Because source terms are calculated for each APB for each observation in the sample (a total of 2,642 and 2,941 source terms for the dry and wet V sequences, respectively), the output from the SEQSOR calculations is quite voluminous and will not be reproduced here. However, this output has been stored on magnetic tape and can be made available upon request. Also, because of the very large number of APBs, performing consequence calculations for each APB is not practicable. Instead, APBs with similar source terms were grouped together using the PARTITION computer program. PARTITION takes the source term information output from SEQSOR and estimates the early and chronic health effects of each source term for each APB. Source terms with similar effects are then grouped together into source term groups. PARTITION is an interactive program and the number of source term groups is specified by the user. The number of source term groups can be further reduced by repooling small source term groups. For the analyses in this report, the final number of source term groups for each sensitivity was 2 except for the base case wet V sequence for which 3 source term groups were used. Table G.6 lists the record of the interactive computer session for the dry V sequence with DF = 5.0 sensitivity.

When the partitioning and repooling is completed, PARTITION creates a single frequency-weighted source term for each source term group. This source term information is then output by PARTITION under the file name MACCS.INP.

4.0 CONSEQUENCE ANALYSIS

A PC version of the MELCOR Accident Consequence Code System Version 1.5.11 (MACCS 1.5.11) was utilized for the consequence analysis. The computer code comprises a single FORTRAN77 program consisting of three basic modules, ATMOS, EARLY, and CHRONC, which are exercised in sequence. This program has been developed for the purpose of evaluating the severe accident consequences at commercial LWR power plants.

Table G.6 Listing of PARTITION interactive computer record for the dry V sequence with DF = 5.0.

DO YOU WANT SUMMARY CONSEQUENCE RESULTS FOR EACH SAMPLE ELEMENT
(NO PARTITIONING PERFORMED)? (Y OR N)

N

LIST OF FILES READ FROM PARTITION.INP:

MASTER BIN LIST FILE =../dry_nbr_mas.kep
SOURCE TERM WEIGHT FILE =../seqwgt.inp
SOURCE TERM FILE =../dry5/release.out
APB COND PROB FILE =../dry_seq.frq
PDS FREQ FILE =../temac.is1

SOURCE TERM FILE CONTAINS SOURCE TERMS WITH 9 RELEASE FRACTIONS

	NUMBER OF SOURCE TERMS	PERCENT OF TOTAL WEIGHTED FREQUENCIES
EF>0 AND CF>0	2642	100.00
EF=0 AND CF>0	0	0.00
EF=0 AND CF=0	0	0.00
TOTAL	2642	100.00

FOR EF>0 AND CF>0, RANGE OF X=LOG10(CF)= 3.0140 TO 4.5220
RANGE OF Y=LOG10(EF)= 0.8222 TO 1.7031

<PRESS RETURN TO CONTINUE>

THE GRID SIZE FOR THE DATA IS DETERMINED BY THE RATIO:

(MAX VALUE IN A GRID CELL)/(MIN VALUE IN A GRID CELL)

GRID SIZES CORRESPONDING TO POSSIBLE RATIOS USED WITH THESE
DATA ARE AS FOLLOWS:

RATIO	GRID SIZE
1.34 TO 1.37	7 X 12
1.38 TO 1.40	7 X 11
1.41 TO 1.41	6 X 11
1.42 TO 1.47	6 X 10
1.48 TO 1.50	6 X 9
1.51 TO 1.54	5 X 9
1.55 TO 1.64	5 X 8
1.65 TO 1.66	5 X 7
1.67 TO 1.78	4 X 7
1.79 TO 1.96	4 X 6
1.97 TO 2.00	3 X 6
2.01 TO 2.38	3 X 5
2.39 TO 2.75	3 X 4
2.76 TO 3.18	2 X 4
3.19 TO 5.67	2 X 3
5.68 TO 7.60	2 X 2
7.61 TO 10.00	1 X 2

Table G.6 Continued

SELECT GRID SIZE BY ENTERING DESIRED RATIO

9.000000

THE SPECIFIED RATIO PRODUCES A GRID SIZE OF 1 X 2

IS THIS GRID SIZE ACCEPTABLE? (Y OR N)

Y

CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 2642:

	1	2
1	681	1961

<PRESS RETURN TO CONTINUE>

PERCENTAGE OF WEIGHTED FREQUENCIES CONTAINED IN EACH CELL:

	1	2
1	16.12	83.88

DO YOU WISH TO POOL THE VALUES IN ANY CELLS WITH
ADJACENT CELLS? (Y OR N)

N

MEAN CF AND EF COORDINATES FOR EACH CELL:

	1		2	
	CF	EF	CF	EF
1	3.39	0.95	4.04	1.35

FOR EF=0 AND CF>0, RANGE OF $X = \log_{10}(CF)$ =***** TO *****
ENTER THE NUMBER OF GROUPS TO BE USED TO PARTITION
BASED ONLY ON CF (0 < NUMBER OF GROUPS < 13)

1

CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 0:

	1
1	

PERCENTAGE OF WEIGHTED FREQUENCIES CONTAINED IN EACH CELL:

	1
1	

Table G.6 Continued

DO YOU WISH TO POOL THE VALUES IN ANY CELLS WITH
ADJACENT CELLS? (Y OR N)

N

DO YOU WISH TO REDO POOLING? (Y OR N)

N

ENTER THE EVACUATION ESCAPE TIME (SEC)
1800.000

ENTER THE EVACUATION DELAY TIME (SEC)
7200.000

DO YOU WISH TO ENTER A SUBGROUP 2 CUTOFF TIME FOLLOWING
START OF RELEASE 1? (Y OR N)

Y

ENTER CUTOFF TIME (S)
3600.000

DO YOU WISH TO POOL THE SUBGROUP 3 SOURCE TERMS WITH SUBGROUP 4
SOURCE TERMS? (Y OR N)

Y

ENTER THE CONDITIONAL PROBABILITY BELOW WHICH
SUBCELLS WILL BE COMBINED
0.0000000E+00

DO YOU WISH TO GENERATE LINE PRINTER PLOTS FOR
INDIVIDUAL SOURCE TERM PARAMETERS FOR SOURCE TERM
CELLS AND SUBCELLS? (Y OR N)

N

DO YOU WISH TO GENERATE FILE FOR USE IN CALCULATION
OF PARTIAL CORRELATION COEFFICIENTS? (Y OR N)

Y

Table G.7 Listing of the PARTITION output file MACCS.INP containing the source term information of the source term groups for the dry V sequence with DF = 5.0.

SEQSOR SOURCE TERMS FOR MACCS

9	3			
1	4	2.1082160E-08	0.1612279	
-4799.998	8449.999	1250.000	3649.999	1800.000
10250.00	21599.99	0.0000000E+00	3699999.	0.9902838
2.1361155E-02	2.1777147E-02	2.3204859E-03	6.5133086E-04	1.5444255E-04
3.5504778E-05	1.4072780E-04	7.4236863E-04	169999.9	9.7354446E-03
2.2498136E-02	9.7113749E-04	1.0220926E-02	4.1008261E-03	4.6844118E-05
4.1119891E-04	4.2428359E-04	3.1941610E-03		
1		0.0000000E+00	0.0000000E+00	0.0000000E+00
2		0.0000000E+00	0.0000000E+00	0.0000000E+00
3		0.0000000E+00	0.0000000E+00	0.0000000E+00
4		2.1082160E-08	1.000000	0.1612279
-4799.998	8449.999	1250.000	3649.999	1800.000
10250.00	21599.99	0.0000000E+00	3699999.	0.9902838
2.1361155E-02	2.1777147E-02	2.3204859E-03	6.5133086E-04	1.5444255E-04
3.5504778E-05	1.4072780E-04	7.4236863E-04	169999.9	9.7354446E-03
2.2498136E-02	9.7113749E-04	1.0220926E-02	4.1008261E-03	4.6844118E-05
4.1119891E-04	4.2428359E-04	3.1941610E-03		
2	4	1.0967797E-07	0.8387731	
-4800.000	8449.995	1250.000	3650.001	1800.001
10250.01	21600.01	0.0000000E+00	3700003.	0.9866727
7.8708678E-02	7.9747535E-02	1.9105045E-02	1.9408649E-02	3.5171590E-03
1.9877709E-03	1.2582630E-02	1.9511884E-02	170000.1	1.3340072E-02
7.0597276E-02	1.1536309E-02	3.9632775E-02	1.2491394E-02	1.3667987E-04
1.0439638E-03	8.7401050E-04	1.0390015E-02		
1		0.0000000E+00	0.0000000E+00	0.0000000E+00
2		0.0000000E+00	0.0000000E+00	0.0000000E+00
3		0.0000000E+00	0.0000000E+00	0.0000000E+00
4		1.0967797E-07	1.000000	0.8387731
-4800.000	8449.995	1250.000	3650.001	1800.001
10250.01	21600.01	0.0000000E+00	3700003.	0.9866727
7.8708678E-02	7.9747535E-02	1.9105045E-02	1.9408649E-02	3.5171590E-03
1.9877709E-03	1.2582630E-02	1.9511884E-02	170000.1	1.3340072E-02
7.0597276E-02	1.1536309E-02	3.9632775E-02	1.2491394E-02	1.3667987E-04
1.0439638E-03	8.7401050E-04	1.0390015E-02		
3	4	0.0000000E+00	0.0000000E+00	

MACCS requires six input files. The EARLY and CHRONC modules each require their own input files, as well as two auxiliary data files: a site data file and a dose conversion file. The June 1989 draft NUREG-1150 EARLY and CHRONC input files for Surry were obtained from Sandia National Laboratories (SNL). They are not reproduced here but are available upon request. These two input files did not require modification for this analysis.

The dose conversion file used was "MACCS DOSE CONVERSION FILE: MOD SER #32, 6-JUL-89, 15:59:19 SANDIA NATIONAL LABORATORIES, J. JOHNSON." This is the most recent dosimetry file provided by SNL for MACCS 1.5. The site data file used was also from the NUREG-1150 MACCS 1.5.11 Surry model. These files are not reproduced here but are available upon request.

The ATMOS module requires a meteorological file for the site and an ATMOS input deck. The meteorological data file used was the NUREG-1150 MACCS 1.5.11 METSUR data file for the Surry site and was obtained from SNL. The ATMOS input deck contains the source term release information and a number of other input parameters which are site and plant-specific. Because this analysis combines the Sequoyah plant with the Surry site, the ATMOS input deck obtained from SNL had to be modified. Also, the source term information output from PARTITION was reformatted and then appended to the MACCS ATMOS input deck. A detailed listing of the ATMOS input file used for this analysis follows:

ATMOS Input Description

Variable Name - **FILE25**

Purpose - Specify EARLY User Input File.

FILE25='C:\MACCS\ISLOCA\V1_EAR.INP', STATUS='OLD', ACTION='READ'

Variable Name - **FILE26**

Purpose - Specify CHRONC User Input File.

FILE26='C:\MACCS\ISLOCA\V1_CHR.INP', STATUS='OLD', ACTION='READ'

Variable Name - **FILE27**

Purpose - Specify DOSE Factors Input Data File.

FILE27='C:\MACCS\ISLOCA\dosdat8', STATUS='OLD', ACTION='READ'

Variable Name - **FILE28**

Purpose - Specify Meteorological Input Data File.

FILE28='C:\MACCS\ISLOCA\metstur', STATUS='OLD', ACTION='READ'

Variable Name - **FILE29**

Purpose - Specify Site Input Data File.

FILE29='C:\MACCS\ISLOCA\sursi10', STATUS='OLD', ACTION='READ'

Variable Name - **FILE06**

Purpose - Specify MACCS Output File (Case Specific).

CASE 1, STG 1: FILE06='C:\MACCS\ISLOCA\DRY\V1_MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 1, STG 2: FILE06='C:\MACCS\ISLOCA\DRY\V1_MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

ATMOS Input (cont.)

CASE 2, STG 1: FILE06='C:\MACCS\ISLOCA\DRY5\V5_MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 2, STG 2: FILE06='C:\MACCS\ISLOCA\DRY5\V5_MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 3, STG 1: FILE06='C:\MACCS\ISLOCA\DRY10\V10MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 3, STG 2: FILE06='C:\MACCS\ISLOCA\DRY10\V10MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 4, STG 1: FILE06='C:\MACCS\ISLOCA\DRY50\V50MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 4, STG 2: FILE06='C:\MACCS\ISLOCA\DRY50\V50MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 5, STG 1: FILE06='C:\MACCS\ISLOCA\DRY100\V100MACC.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 5, STG 2: FILE06='C:\MACCS\ISLOCA\DRY100\V100MACC.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 6, STG 1: FILE06='C:\MACCS\ISLOCA\WET\W1_MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 6, STG 2: FILE06='C:\MACCS\ISLOCA\WET\W1_MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 6, STG 3: FILE06='C:\MACCS\ISLOCA\WET\W1_MACCS.03', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

ATMOS Input (cont.)

CASE 7, STG 1: FILE06='C:\MACCS\ISLOCA\WET5\W5_MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 7, STG 2: FILE06='C:\MACCS\ISLOCA\WET5\W5_MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 8, STG 1: FILE06='C:\MACCS\ISLOCA\WET10\W10MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 8, STG 2: FILE06='C:\MACCS\ISLOCA\WET10\W10MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 9, STG 1: FILE06='C:\MACCS\ISLOCA\WET50\W50MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 9, STG 2: FILE06='C:\MACCS\ISLOCA\WET50\W50MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 10, STG 1: FILE06='C:\MACCS\ISLOCA\WET50\W51MACCS.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 10, STG 2: FILE06='C:\MACCS\ISLOCA\WET50\W51MACCS.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 11, STG 1: FILE06='C:\MACCS\ISLOCA\WET100\W100MACC.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 11, STG 2: FILE06='C:\MACCS\ISLOCA\WET100\W100MACC.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

CASE 12, STG 1: FILE06='C:\MACCS\ISLOCA\WET100\W110MACC.01', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

ATMOS Input (cont.)

CASE 12, STG 2: FILE06='C:\MACCS\ISLOCA\WET100\W110MACC.02', STATUS='UNKNOWN',
CARRIAGE CONTROL='FORTRAN'

Variable Name - **ATNAM1**
Purpose - Identifier for specific ATMOS case.

RIATNAM1001 'SURRY SITE, SEQUOYAH SOURCE TERMS FOR ISLOCA'

Variable Name - **ENDAT1**
Purpose - Flag to indicate that this is the last program in the series to be run.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

OCENDAT1001 .FALSE. (SET THIS VALUE TO .TRUE. TO SKIP EARLY AND CHRONC)

Variable Name - **NUMRAD**
Purpose - Number of radial spatial elements defined in the model.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

GENUMRAD001 26

Variable Name - **SPAEND**
Purpose - Distance in meters to the end of the spacial intervals.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

* SURRY

*

GESPAEND001	.16	.52	1.21	1.61	2.13
GESPAEND002	3.22	4.02	4.83	5.63	8.05
GESPAEND003	11.27	16.09	20.92	25.75	32.19
GESPAEND004	40.23	48.28	64.37	80.47	112.65
GESPAEND005	160.93	241.14	321.87	563.27	804.67
GESPAEND006	1609.34				

ATMOS Input (cont.)

Variable Name - **NUMISO**
 Purpose - Number of nuclides defined in the model.
 Source - NUREG-1150 MACCS 1.5.11 Sequoyah Model.

ISNUMISO001 60

Variable Name - **MAXGRP**
 Purpose - Number of nuclide groups defined in the model.
 Source - NUREG-1150 MACCS 1.5.11 Sequoyah Model.

ISMAXGRP001 9

Variable Name - **WETDEP**
 Purpose - Logical flag for each of the nuclide groups that indicate whether they are subject to wet deposition.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

Variable Name - **DRYDEP**
 Purpose - Logical flag for each of the nuclide groups that indicate whether they are subject to dry deposition.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

*	WETDEP	DRYDEP
*		
ISDEPFLA001	.FALSE.	.FALSE.
ISDEPFLA002	.TRUE.	.TRUE.
ISDEPFLA003	.TRUE.	.TRUE.
ISDEPFLA004	.TRUE.	.TRUE.
ISDEPFLA005	.TRUE.	.TRUE.
ISDEPFLA006	.TRUE.	.TRUE.
ISDEPFLA007	.TRUE.	.TRUE.
ISDEPFLA008	.TRUE.	.TRUE.
ISDEPFLA009	.TRUE.	.TRUE.

ATMOS Input (cont.)

Variable Name - **NUCNAM**
 Purpose - Identifying name associated with each on the nuclides.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

Variable Name - **PARENT**
 Purpose - Name of parent nuclide if any.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

Variable Name - **IGROUP**
 Purpose - Chemical group to which nuclide is assigned.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

Variable Name - **HAFLIF**
 Purpose - Half-life of the isotope in seconds.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

*	NUCNAM	PARENT	IGROUP	HAFLIF(S)	
*					
ISOTPGRP001	CO-58	NONE	6	6.160E+06	
ISOTPGRP002	CO-60	NONE	6	1.660E+08	
ISOTPGRP003	KR-85	NONE	1	3.386E+08	
ISOTPGRP004	KR-85M	NONE	1	1.613E+04	
ISOTPGRP005	KR-87	NONE	1	4.560E+03	
ISOTPGRP006	KR-88	NONE	1	1.008E+04	
ISOTPGRP007	RB-86	NONE	3	1.611E+06	
ISOTPGRP008	SR-89	NONE	5	4.493E+06	
ISOTPGRP009	SR-90	NONE	5	8.865E+08	
ISOTPGRP010	SR-91	NONE	5	3.413E+04	
ISOTPGRP011	SR-92	NONE	5	9.756E+03	NEW
ISOTPGRP012	Y-90	SR-90	7	2.307E+05	
ISOTPGRP013	Y-91	SR-91	7	5.080E+06	
ISOTPGRP014	Y-92	SR-92	7	1.274E+04	NEW
ISOTPGRP015	Y-93	NONE	7	3.636E+04	NEW

ATMOS Input (cont.)

ISOTPGRP016	ZR-95	NONE	7	5.659E+06	
ISOTPGRP017	ZR-97	NONE	7	6.048E+04	
ISOTPGRP018	NB-95	ZR-95	7	3.033E+06	
ISOTPGRP019	MO-99	NONE	6	2.377E+05	
ISOTPGRP020	TC-99M	MO-99	6	2.167E+04	
ISOTPGRP021	RU-103	NONE	6	3.421E+06	
ISOTPGRP022	RU-105	NONE	6	1.598E+04	
ISOTPGRP023	RU-106	NONE	6	3.188E+07	
ISOTPGRP024	RH-105	RU-105	6	1.278E+05	
ISOTPGRP025	SB-127	NONE	4	3.283E+05	
ISOTPGRP026	SB-129	NONE	4	1.562E+04	
ISOTPGRP027	TE-127	SB-127	4	3.366E+04	
ISOTPGRP028	TE-127M	NONE	4	9.418E+06	
ISOTPGRP029	TE-129	SB-129	4	4.200E+03	
ISOTPGRP030	TE-129M	NONE	4	2.886E+06	
ISOTPGRP031	TE-131M	NONE	4	1.080E+05	
ISOTPGRP032	TE-132	NONE	4	2.808E+05	
ISOTPGRP033	I-131	TE-131M	2	6.947E+05	
ISOTPGRP034	I-132	TE-132	2	8.226E+03	
ISOTPGRP035	I-133	NONE	2	7.488E+04	
ISOTPGRP036	I-134	NONE	2	3.156E+03	
ISOTPGRP037	I-135	NONE	2	2.371E+04	
ISOTPGRP038	XE-133	I-133	1	4.571E+05	
ISOTPGRP039	XE-135	I-135	1	3.301E+04	
ISOTPGRP040	CS-134	NONE	3	6.501E+07	
ISOTPGRP041	CS-136	NONE	3	1.123E+06	
ISOTPGRP042	CS-137	NONE	3	9.495E+08	
ISOTPGRP043	BA-139	NONE	9	4.986E+03	NEW
ISOTPGRP044	BA-140	NONE	9	1.105E+06	
ISOTPGRP045	LA-140	BA-140	7	1.448E+05	
ISOTPGRP046	LA-141	NONE	7	1.418E+04	NEW
ISOTPGRP047	LA-142	NONE	7	5.724E+03	NEW
ISOTPGRP048	CE-141	LA-141	8	2.811E+06	PARENT ADDED

ATMOS Input (cont.)

ISOTPGRP049	CE-143	NONE	8	1.188E+05
ISOTPGRP050	CE-144	NONE	8	2.457E+07
ISOTPGRP051	PR-143	CE-143	7	1.173E+06
ISOTPGRP052	ND-147	NONE	7	9.495E+05
ISOTPGRP053	NP-239	NONE	8	2.030E+05
ISOTPGRP054	PU-238	CM-242	8	2.809E+09
ISOTPGRP055	PU-239	NP-239	8	7.700E+11
ISOTPGRP056	PU-240	CM-244	8	2.133E+11
ISOTPGRP057	PU-241	NONE	8	4.608E+08
ISOTPGRP058	AM-241	PU-241	7	1.366E+10
ISOTPGRP059	CM-242	NONE	7	1.408E+07
ISOTPGRP060	CM-244	NONE	7	5.712E+08

Variable Name - **CWASH1**
 Purpose - Linear term of the washout factor.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

WDCWASH1001 9.5E-5 (JON HELTON AFTER JONES, 1986)

Variable Name - **CWASH2**
 Purpose - The exponential term for the washout factor.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

WDCWASH2001 0.8 (JON HELTON AFTER JONES, 1986)

Variable Name - **NPSGRP**
 Purpose - The number of particle size groups that are used for dry deposition.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

DDNPSGRP001 1

ATMOS Input (cont.)

Variable Name - **VDEPOS**
 Purpose - The representative dry deposition velocities associated with each of the particle size groups.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

DDVDEPOS001 0.01

Variable Name - **CYSIGA**
 Purpose - The linear term in the expression for sigma-y for 6 stability classes.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

* STABILITY CLASS: A B C D E F
 *
 DPCYSIGA001 0.3658 0.2751 0.2089 0.1474 0.1046 0.0722

Variable Name - **CYSIGB**
 Purpose - The exponential term of the expression for sigma-y, 6 stability classes.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

* STABILITY CLASS: A B C D E F
 *
 DPCYSIGB001 .9031 .9031 .9031 .9031 .9031 .9031

Variable Name - **CZSIGA**
 Purpose - The linear term of the expression for sigma-z, 6 stability classes.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

* STABILITY CLASS: A B C D E F
 *
 DPCZSIGA001 2.5E-4 1.9E-3 .2 .3 .4 .2

ATMOS Input (cont.)

Variable Name - **CZSIGB**
 Purpose - The exponential term of the expression for sigma-z, 6 stability classes.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

* STABILITY CLASS;	A	B	C	D	E	F
*						
DPCZSIGB001	2.125	1.6021	.8543	.6532	.6021	.6020

Variable Name - **YSCALE**
 Purpose - The linear scaling factor for the sigma-y function.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

DPYSCALE001 1.

Variable Name - **ZSCALE**
 Purpose - The linear scaling factor for the sigma-z function.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

DPZSCALE001 1.27

Variable Name - **TIMBAS**
 Purpose - The time base for the expansion factor (seconds).
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

PMTIMBAS001 600. (10 MINUTES)

Variable Name - **BRKPNT**
 Purpose - The break point in the formula used for calculating the plume meander expansion factor.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

PMBRKPNT001 3600. (1 HOUR)

ATMOS Input (cont.)

Variable Name - **XPFAC1**
Purpose - Exponential expansion factor number 1.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

PMXPFAC1001 0.2

Variable Name - **XPFAC2**
Purpose - Exponential expansion factor number 2.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

PMXPFAC2001 0.25

Variable Name - **SCLCRW**
Purpose - Scaling factor for the critical wind speed for entrainment of a buoyant plume.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

PRSCLCRW001 1.

Variable Name - **SCLADP**
Purpose - Scaling factor for the a-d stability plume rise formula.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

PRSCLADP001 1.

Variable Name - **SCLEFP**
Purpose - Scaling factor for the e-f stability plume rise formula.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

PRSCLEFP001 1.

ATMOS Input (cont.)

Variable Name - **BUILDW**
Purpose - Width of the reactor building in meters.
Source - NUREG-1150 MACCS 1.5.11 Sequoyah Model.

WEBUILDW001 40. * SEQUOYAH

Variable Name - **BUILDH**
Purpose - Height of the reactor building in meters.
Source - NUREG-1150 MACCS 1.5.11 Sequoyah Model.

WEBUILDH001 40. * SEQUOYAH

Variable Name - **IDEBUG**
Purpose - Debug output flag (0 - no debug).
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

OCIDEBUG001 0

Variable Name - **NUCOUT**
Purpose - Specifies which nuclide will appear on the dispersion listing if one is produced. The dispersion listing is only produced if IDEBUG is greater than zero.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

OCNUCOUT001 CS-137

Variable Name - **METCOD**
Purpose - Meteorological sampling option code.
metcod = 1, user specified day and hour in the year,
2, weather category bin sampling,
3, 120 hours of weather specified on the atmos user input file,
4, constant met,

ATMOS Input (cont.)

5, stratified random samples for each day of the year.

Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M1METCOD001 2

Variable Name - **ISTRDY**

Purpose - Day in the year on which the weather sequence is to begin
(Not used if METCOD is specified as 2.)

Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M3ISTRDY001 1

Variable Name - **ISTRDY**

Purpose - Hour of day on which the weather sequence is to begin (Not
used if METCOD is specified as 2.)

Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M3ISTRHR001 1

Variable Name - **LIMSPA**

Purpose - Last Spacial Interval for Measured Weather

Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M2LIMSPA001 25

Variable Name - **BNDMXH**

Purpose - Boundary weather mixing layer height.

Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M2BNDMXH001 1000. (METERS)

ATMOS Input (cont.)

Variable Name - **IBDSTB**
Purpose - Boundary weather stability class index.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M2IBDSTB001 4 (D-STABILITY)

Variable Name - **BNDRAN**
Purpose - Boundary weather rain rate.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M2BNDRAN001 5.0 (MM/HR)

Variable Name - **BNDWND**
Purpose - Boundary weather wind speed.
Source - NUREG-1150 MACCS 1.5.11 Surry Model specifies 4.0 M/S.
However, 5.0 was used in these calculations. Sensitivity calculations indicate this difference in wind speed has no effect on the consequence results.

M2BNDWND001 5.0 (M/S)

Variable Name - **NSMPLS**
Purpose - Number of samples per bin.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M4NSMPLS001 4 (THIS NUMBER SHOULD BE SET TO 4 FOR RISK ASSESSMENT)

Variable Name - **NRNINT**
Purpose - Number of rain distance intervals for binning.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M4NRNINT001 6

ATMOS Input (cont.)

Variable Name - **RNDSTS**
 Purpose - Endpoints of the rain distance intervals (kilometers).
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M4RNDSTS001 3.22 5.63 11.27 20.92 40.23 80.47

Variable Name - **NRINTN**
 Purpose - Number of rain intensity breakpoints.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M4NRINTN001 3

Variable Name - **RNRATE**
 Purpose - Rain intensity breakpoints for weather binning
 (millimeters per hour).
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M4RNRATE001 2. 4. 6.

Variable Name - **IRSEED**
 Purpose - Initial seed for random number generator.
 Source - NUREG-1150 MACCS 1.5.11 Surry Model.

M4IRSEED001 79

Variable Name - **CORINV**
 Purpose - Defines the total core inventory for each nuclide, NUCNAM.
 Source - 3412 MWTH PWR core inventory, end-of-cycle, supplied by D.
 E. Bennett, 5/14/86, NUREG-1150 MACCS 1.5.11 Model.

* NUCNAM CORINV(CI)

*

RDCORINV001 CO-58 3.223E+16

ATMOS Input (cont.)

RDCORINV002	CO-60	2.465E+16
RDCORINV003	KR-85	2.475E+16
RDCORINV004	KR-85M	1.159E+18
RDCORINV005	KR-87	2.118E+18
RDCORINV006	KR-88	2.864E+18
RDCORINV007	RB-86	1.888E+15
RDCORINV008	SR-89	3.590E+18
RDCORINV009	SR-90	1.938E+17
RDCORINV010	SR-91	4.616E+18
RDCORINV011	SR-92	4.803E+18
RDCORINV012	Y-90	2.079E+17
RDCORINV013	Y-91	4.374E+18
RDCORINV014	Y-92	4.821E+18
RDCORINV015	Y-93	5.454E+18
RDCORINV016	ZR-95	5.526E+18
RDCORINV017	ZR-97	5.759E+18
RDCORINV018	NB-95	5.224E+18
RDCORINV019	MO-99	6.098E+18
RDCORINV020	TC-99M	5.263E+18
RDCORINV021	RU-103	4.542E+18
RDCORINV022	RU-105	2.954E+18
RDCORINV023	RU-106	1.032E+18
RDCORINV024	RH-105	2.046E+18
RDCORINV025	SB-127	2.787E+17
RDCORINV026	SB-129	9.872E+17
RDCORINV027	TE-127	2.692E+17
RDCORINV028	TE-127M	3.564E+16
RDCORINV029	TE-129	9.267E+17
RDCORINV030	TE-129M	2.443E+17
RDCORINV031	TE-131M	4.680E+17
RDCORINV032	TE-132	4.658E+18
RDCORINV033	I-131	3.206E+18
RDCORINV034	I-132	4.725E+18

ATMOS Input (cont.)

RDCORINV035	I-133	6.779E+18
RDCORINV036	I-134	7.440E+18
RDCORINV037	I-135	6.392E+18
RDCORINV038	XE-133	6.782E+18
RDCORINV039	XE-135	1.273E+18
RDCORINV040	CS-134	4.324E+17
RDCORINV041	CS-136	1.316E+17
RDCORINV042	CS-137	2.417E+17
RDCORINV043	BA-139	6.282E+18
RDCORINV044	BA-140	6.216E+18
RDCORINV045	LA-140	6.352E+18
RDCORINV046	LA-141	5.826E+18
RDCORINV047	LA-142	5.616E+18
RDCORINV048	CE-141	5.651E+18
RDCORINV049	CE-143	5.494E+18
RDCORINV050	CE-144	3.405E+18
RDCORINV051	PR-143	5.395E+18
RDCORINV052	ND-147	2.412E+18
RDCORINV053	NP-239	6.464E+19
RDCORINV054	PU-238	3.664E+15
RDCORINV055	PU-239	8.263E+14
RDCORINV056	PU-240	1.042E+15
RDCORINV057	PU-241	1.755E+17
RDCORINV058	AM-241	1.159E+14
RDCORINV059	CM-242	4.436E+16
RDCORINV060	CM-244	2.596E+15

Variable Name - SCLCRW

Purpose - Scaling factor to adjust the core inventory.

Source - NUREG-1150 MACCS 1.5.11 Sequoyah Model.

RDCORSCA001 1.003

ATMOS Input (cont.)

Variable Name - **PSDIST**
Purpose - Particle size distribution for each nuclide group.
Source - NUREG-1150 MACCS 1.5.11 Surry Model.

RDPSDIST001 1.0
RDPSDIST002 1.0
RDPSDIST003 1.0
RDPSDIST004 1.0
RDPSDIST005 1.0
RDPSDIST006 1.0
RDPSDIST007 1.0
RDPSDIST008 1.0
RDPSDIST009 1.0

Variable Name - **ATNAM2**
Purpose - Descriptive text identifying the source term. This text is used to identify specific source terms in the output (case specific.)

Case 1, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY BASE CASE, DFV=1.0'
RDATNAM2003 'SOURCE TERM-001, GROUP-003'

Case 1, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY BASE CASE, DFV=1.0'
RDATNAM2003 'SOURCE TERM-002, GROUP-004'

Case 2, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=5.0'
RDATNAM2003 'SOURCE TERM-001, GROUP-001'

ATMOS Input (cont.)

Case 2, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=5.0'
RDATNAM2003 'SOURCE TERM-002, GROUP-002'

Case 3, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=10.0'
RDATNAM2003 'SOURCE TERM-001, GROUP-001'

Case 3, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=10.0'
RDATNAM2003 'SOURCE TERM-002, GROUP-002'

Case 4, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=50.0'
RDATNAM2003 'RELEASE ELEVATION = 10.0 m.'
RDATNAM2004 'SOURCE TERM-001, GROUP-002'

Case 4, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=50.0'
RDATNAM2003 'RELEASE ELEVATION = 10.0 m.'
RDATNAM2004 'SOURCE TERM-002, GROUP-003'

Case 5, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=100.0'
RDATNAM2003 'RELEASE ELEVATION = 10.0 m.'
RDATNAM2004 'SOURCE TERM-001, GROUP-002'

ATMOS Input (cont.)

Case 5, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET DRY CASE, DFV=100.0'
RDATNAM2003 'RELEASE ELEVATION = 10.0 m.'
RDATNAM2004 'SOURCE TERM-002, GROUP-003'

Case 6, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET BASE, DFV SAMPLED'
RDATNAM2003 'SOURCE TERM-001, GROUP-006'

Case 6, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET BASE, DFV SAMPLED'
RDATNAM2003 'SOURCE TERM-002, GROUP-007'

Case 6, STG 3:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET BASE, DFV SAMPLED'
RDATNAM2003 'SOURCE TERM-003, GROUP-008'

Case 7, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=5.0'
RDATNAM2003 'SOURCE TERM-001, GROUP-001'

Case 7, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=10.0'
RDATNAM2003 'SOURCE TERM-002, GROUP-002'

ATMOS Input (cont.)

Case 8, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=10.0'
RDATNAM2003 'SOURCE TERM-001, GROUP-001'

Case 8, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=10.0'
RDATNAM2003 'SOURCE TERM-002, GROUP-002'

Case 9, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=50.0'
RDATNAM2003 'RELEASE ELEVATION IS 0.0 m.'
RDATNAM2004 'SOURCE TERM-001, GROUP-002'

Case 9, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=50.0'
RDATNAM2003 'RELEASE ELEVATION IS 0.0 m.'
RDATNAM2004 'SOURCE TERM-002, GROUP-003'

Case 10, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=50.0'
RDATNAM2003 'RELEASE ELEVATION IS 10.0 m.'
RDATNAM2004 'SOURCE TERM-001, GROUP-002'

Case 10, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=50.0'
RDATNAM2003 'RELEASE ELEVATION IS 10.0 m.'
RDATNAM2004 'SOURCE TERM-002, GROUP-003'

ATMOS Input (cont.)

Case 11, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=100.0'
RDATNAM2003 'RELEASE ELEVATION IS BASE VALUE, 0.0 m.'
RDATNAM2004 'SOURCE TERM-001, GROUP-002'

Case 11, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=100.0'
RDATNAM2003 'RELEASE ELEVATION IS BASE VALUE, 0.0 m.'
RDATNAM2004 'SOURCE TERM-002, GROUP-003'

Case 12, STG 1:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=100.0'
RDATNAM2003 'RELEASE ELEVATION IS 10.0 m.'
RDATNAM2004 'SOURCE TERM-001, GROUP-002'

Case 12, STG 2:

RDATNAM2001 'SEQUOYAH SOURCE TERM AS OBTAINED FROM'
RDATNAM2002 'SEQSOR, FULL APET WET CASE, DFV=100.0'
RDATNAM2003 'RELEASE ELEVATION IS 10.0 m.'
RDATNAM2004 'SOURCE TERM-002, GROUP-003'

Variable Name - OALARM

Purpose - Time after accident initiation when the accident reaches general emergency conditions (as defined in NUREG-0654), or when plant personnel can reliably predict that general emergency conditions will be attained

Source - NUREG-1150 SEQSOR source term information for Sequoyah Model.

ATMOS Input (cont.)

RDOALARM001 1250.0

Variable Name - **MAXRIS**

Purpose - Selection of risk dominant plume.

Source - NUREG-1150 MACCS 1.5.11 Sequoyah Model (for small release fractions, on the order of 1.0E-02 or less, plume selection has little effect on consequence results.)

RDMAXRIS001 1

Variable Name - **REFTIM**

Purpose - Specifies the representative time point of each plume segment (0.0 = leading edge, 0.5 = midpoint, 1.0 = trailing edge). The radioactive decay, dry deposition, and dispersion calculations are all performed as if the entire contents of the plume segment are located at this point.

Source - A value of 0.0 is assumed for the first plume in this analysis. This results in a conservative calculation. For the second plume, a value of 0.5 is used in accordance with the NUREG-1150 MACCS 1.5.11 Surry Model.

RDREFTIM001 0.00 0.50

Variable Name - **NUMREL**

Purpose - Number of plume segments that are released.

Source - NUREG-1150 SEQSOR source term information for Sequoyah Model.

RDNUMREL001 2

Variable Name - **PLHEAT**

Purpose - Heat content of plume release (w).

Source - NUREG-1150 SEQSOR source term information for Sequoyah Model.

ATMOS Input (cont.)

Dry V sequences:

RDPLHEAT001 3.70E+06 1.70E+05

Wet V sequences:

RDPLHEAT001 1.85E+06 1.70E+05

Variable Name - **PLHITE**

Purpose - Height of plume segments at release (m).

Source - NUREG-1150 SEQSOR source term information for Sequoyah Model for cases 1, 2, 3, 6, 7, 8, 9, and 11. MELCOR calculations for cases 4, 5, 10, and 12.

Cases 1, 2, 3, 6, 7, 8, 9, and 11:

RDPLHITE001 0. 0.

Cases 4, 5, 10, and 12:

RDPLHITE001 10.0 10.0

Variable Name - **PLUDUR**

Purpose - Duration of plume segments (s).

Source - NUREG-1150 SEQSOR source term information for Sequoyah Model.

RDPLUDUR001 1800. 21600.

Variable Name - **PDELAY**

Purpose - Specifies the start time of each plume segment in seconds from the time of accident initiation, e.g., reactor scram.

Source - NUREG-1150 SEQSOR source term information for Sequoyah Model.

RDPDELAY001 3650. 10250.

ATMOS Input (cont.)

Variable Name - **RELFAC**
Purpose - Release fractions for isotope groups in release (Case Dependant).
Sources - Cases 1 and 6 represent the base case analyses for the dry and wet V sequences, respectively, as obtained from the NUREG-1150 SEQSOR source term information for Sequoyah. The remaining cases are obtained by modifying the V sequence DFs of the NUREG-1150 SEQSOR source term code for Sequoyah. For all cases except one, the accident progression bins have been partitioned into two source term groups. The one exception is case 6 for which three source term groups were used.

* ISOTOPE GROUPS: XE/KR I CS TE SR RU LA CE BA

Case 1, Source Term Group 1:

* Dry V sequence, DF=1.0, Release Elevation=0.0

RDRELFR001 9.84E-01 4.56E-01 4.63E-01 1.16E-01 1.21E-01 2.19E-02 1.24E-02
7.88E-02 1.22E-01
RDRELFR002 1.62E-02 3.32E-02 4.31E-03 1.84E-01 6.56E-02 7.78E-04 4.71E-03
3.46E-03 5.52E-02

Case 1, Source Term Group 2:

* Dry V sequence, DF=1.0, Release Elevation=0.0

RDRELFR001 9.94E-01 1.29E-01 1.27E-01 1.25E-02 2.48E-03 6.69E-04 1.25E-04
4.72E-04 2.92E-03
RDRELFR002 5.73E-03 6.17E-02 2.10E-02 8.03E-02 3.56E-02 2.73E-04 4.72E-03
5.12E-03 2.78E-02

Case 2, Source Term Group 1:

* Dry V sequence, DF=5.0, Release Elevation=0.0

ATMOS Input (cont.)

RDRELFRC001 9.90E-01 2.14E-02 2.18E-02 2.32E-03 6.51E-04 1.54E-04 3.55E-05
1.41E-04 7.42E-04

RDRELFRC002 9.74E-03 2.25E-02 9.71E-04 1.02E-02 4.10E-03 4.68E-05 4.11E-04
4.24E-04 3.19E-03

Case 2, Source Term Group 2:

* Dry V sequence, DF=5.0, Release Elevation=0.0

RDRELFRC001 9.87E-01 7.87E-02 7.97E-02 1.91E-02 1.94E-02 3.52E-03 1.99E-03
1.26E-02 1.95E-02

RDRELFRC002 1.33E-02 7.06E-02 1.15E-02 3.96E-02 1.25E-02 1.37E-04 1.04E-03
8.74E-04 1.04E-02

Case 3, Source Term Group 1:

* Dry V sequence, DF=10.0, Release Elevation=0.0

RDRELFRC001 9.91E-01 1.39E-02 1.43E-02 1.42E-03 2.65E-04 6.55E-05 1.39E-05
5.40E-05 3.11E-04

RDRELFRC002 8.73E-03 2.14E-02 1.14E-03 1.12E-02 6.01E-03 1.23E-04 8.06E-04
7.03E-04 5.21E-03

Case 3, Source Term Group 2:

* Dry V sequence, DF=10.0, Release Elevation=0.0

RDRELFRC001 9.86E-01 4.09E-02 4.14E-02 1.02E-02 1.06E-02 1.91E-03 1.08E-03
6.85E-03 1.06E-02

RDRELFRC002 1.40E-02 7.84E-02 1.24E-02 2.33E-02 5.44E-03 4.27E-05 3.71E-04
3.11E-04 4.44E-03

Case 4, Source Term Group 1:

* Dry V sequence, DF=50.0, Release Elevation=10.0

ATMOS Input (cont.)

RDRELFRC001 9.85E-01 7.75E-03 7.89E-03 1.92E-03 1.97E-03 3.56E-04 2.02E-04
1.28E-03 1.98E-03
RDRELFRC002 1.50E-02 5.72E-02 3.46E-03 4.72E-03 1.14E-03 1.35E-05 8.39E-05
6.40E-05 9.52E-04

Case 4, Source Term Group 2:

* Dry V sequence, DF=50.0, Release Elevation=10.0

RDRELFRC001 9.98E-01 3.04E-03 2.96E-03 2.86E-04 3.65E-05 1.28E-05 1.60E-06
5.35E-06 4.62E-05
RDRELFRC002 1.79E-03 1.16E-01 4.06E-02 3.01E-02 1.00E-03 6.15E-06 1.44E-04
1.58E-04 7.82E-04

Case 5, Source Term Group 1:

* Dry V sequence, DF=100.0, Release Elevation=10.0

RDRELFRC001 9.85E-01 3.88E-03 3.94E-03 9.61E-04 9.86E-04 1.78E-04 1.01E-04
6.38E-04 9.91E-04
RDRELFRC002 1.50E-02 5.74E-02 3.43E-03 3.14E-03 5.68E-04 6.74E-06 4.20E-05
3.20E-05 4.76E-04

Case 5, Source Term Group 2:

* Dry V sequence, DF=100.0, Release Elevation=10.0

RDRELFRC001 9.98E-01 1.52E-03 1.48E-03 1.43E-04 1.82E-05 6.38E-06 7.96E-07
2.66E-06 2.30E-05
RDRELFRC002 1.79E-03 1.16E-01 4.06E-02 2.92E-02 5.02E-04 3.07E-06 7.18E-05
7.87E-05 3.92E-04

Case 6, Source Term Group 1:

* Wet V sequence, Base case sampled DF, Release Elevation=0.0

ATMOS Input (cont.)

RDRELFRC001 9.86E-01 1.81E-02 1.84E-02 1.87E-03 2.06E-04 6.24E-05 9.88E-06
3.38E-05 2.73E-04

RDRELFRC002 1.40E-02 4.18E-02 3.26E-03 1.66E-02 9.26E-03 2.09E-04 1.34E-03
1.22E-03 8.13E-03

Case 6, Source Term Group 2:

* Wet V sequence, Base case sampled DF, Release Elevation=0.0

RDRELFRC001 9.94E-01 1.82E-01 1.81E-01 5.77E-02 5.17E-02 9.37E-03 5.32E-03
3.38E-02 5.20E-02

RDRELFRC002 5.91E-03 4.64E-02 1.66E-03 4.12E-02 1.34E-02 7.55E-04 5.70E-04
2.29E-03 1.21E-02

Case 6, Source Term Group 3:

* Wet V sequence, Base case sampled DF, Release Elevation=0.0

RDRELFRC001 9.98E-01 5.06E-02 4.98E-02 4.39E-03 4.39E-04 1.65E-04 1.83E-05
5.98E-05 5.66E-04

RDRELFRC002 1.95E-03 1.03E-01 3.80E-02 5.18E-02 1.13E-02 6.89E-05 1.62E-03
1.77E-03 8.87E-03

Case 7, Source Term Group 1:

* Wet V sequence, DF=5.0, Release Elevation=0.0

RDRELFRC001 9.90E-01 2.12E-02 2.14E-02 1.87E-03 3.64E-04 9.49E-05 2.09E-05
8.32E-05 4.31E-04

RDRELFRC002 9.56E-03 2.38E-02 1.05E-03 1.13E-02 6.23E-03 1.06E-04 7.64E-04
8.10E-04 5.00E-03

Case 7, Source Term Group 2:

* Wet V sequence, DF=5.0, Release Elevation=0.0

ATMOS Input (cont.)

RDRELFRC001 9.91E-01 7.19E-02 7.25E-02 1.53E-02 1.17E-02 2.15E-03 1.18E-03
7.44E-03 1.19E-02

RDRELFRC002 8.95E-03 6.48E-02 1.19E-02 3.73E-02 1.24E-02 2.56E-04 1.28E-03
1.42E-03 1.06E-02

Case 8, Source Term Group 1:

* Wet V sequence, DF=10.0, Release Elevation=0.0

RDRELFRC001 9.91E-01 1.41E-02 1.43E-02 1.27E-03 1.52E-04 4.12E-05 8.11E-06
3.13E-05 1.88E-04

RDRELFRC002 8.61E-03 2.16E-02 1.08E-03 1.20E-02 7.20E-03 1.61E-04 1.02E-03
9.26E-04 6.27E-03

Case 8, Source Term Group 2:

* Wet V sequence, DF=10.0, Release Elevation=0.0

RDRELFRC001 9.91E-01 3.81E-02 3.84E-02 8.48E-03 6.71E-03 1.23E-03 6.78E-04
4.27E-03 6.80E-03

RDRELFRC002 9.30E-03 7.48E-02 1.35E-02 2.21E-02 4.79E-03 8.90E-05 3.82E-04
5.17E-04 3.97E-03

Case 9, Source Term Group 1:

* Wet V sequence, DF=50.0, Release Elevation=0.0

RDRELFRC001 9.89E-01 6.81E-03 6.90E-03 1.46E-03 1.13E-03 2.06E-04 1.14E-04
7.16E-04 1.14E-03

RDRELFRC002 1.07E-02 4.75E-02 2.56E-03 3.81E-03 1.09E-03 2.57E-05 1.04E-04
1.16E-04 9.41E-04

Case 9, Source Term Group 2:

* Wet V sequence, DF=50.0, Release Elevation=0.0

ATMOS Input (cont.)

RDRELFRC001 9.98E-01 2.92E-03 2.85E-03 2.77E-04 4.07E-05 1.28E-05 1.79E-06
6.12E-06 5.08E-05

RDRELFRC002 1.70E-03 1.14E-01 4.09E-02 3.01E-02 1.20E-03 7.39E-06 1.73E-04
1.89E-04 9.37E-04

Case 10, Source Term Group 1:

* Wet V sequence, DF=50.0, Release Elevation=10.0

RDRELFRC001 9.89E-01 6.81E-03 6.90E-03 1.46E-03 1.13E-03 2.06E-04 1.14E-04
7.16E-04 1.14E-03

RDRELFRC002 1.07E-02 4.75E-02 2.56E-03 3.81E-03 1.09E-03 2.57E-05 1.04E-04
1.16E-04 9.41E-04

Case 10, Source Term Group 2:

* Wet V sequence, DF=50.0, Release Elevation=10.0

RDRELFRC001 9.98E-01 2.92E-03 2.85E-03 2.77E-04 4.07E-05 1.28E-05 1.79E-06
6.12E-06 5.08E-05

RDRELFRC002 1.70E-03 1.14E-01 4.09E-02 3.01E-02 1.20E-03 7.39E-06 1.73E-04
1.89E-04 9.37E-04

Case 11, Source Term Group 1:

* Wet V sequence, DF=100.0, Release Elevation=0.0

RDRELFRC001 9.89E-01 3.41E-03 3.45E-03 7.28E-04 5.63E-04 1.03E-04 5.69E-05
3.58E-04 5.71E-04

RDRELFRC002 1.07E-02 4.77E-02 2.54E-03 2.43E-03 5.45E-04 1.29E-05 5.22E-05
5.79E-05 4.70E-04

Case 11, Source Term Group 2:

* Wet V sequence, DF=100.0, Release Elevation=0.0

ATMOS Input (cont.)

RDRELFRC001 9.98E-01 1.46E-03 1.43E-03 1.38E-04 2.03E-05 6.40E-06 8.93E-07
3.05E-06 2.54E-05

RDRELFRC002 1.70E-03 1.14E-01 4.08E-02 2.92E-02 6.01E-04 3.69E-06 8.61E-05
9.44E-05 4.69E-04

Case 12, Source Term Group 1:

* Wet V sequence, DF=100.0, Release Elevation=10.0

RDRELFRC001 9.89E-01 3.41E-03 3.45E-03 7.28E-04 5.63E-04 1.03E-04 5.69E-05
3.58E-04 5.71E-04

RDRELFRC002 1.07E-02 4.77E-02 2.54E-03 2.43E-03 5.45E-04 1.29E-05 5.22E-05
5.79E-05 4.70E-04

Case 12, Source Term Group 2:

* Wet V sequence, DF=100.0, Release Elevation=10.0

RDRELFRC001 9.98E-01 1.46E-03 1.43E-03 1.38E-04 2.03E-05 6.40E-06 8.93E-07
3.05E-06 2.54E-05

RDRELFRC002 1.70E-03 1.14E-01 4.08E-02 2.92E-02 6.01E-04 3.69E-06 8.61E-05
9.44E-05 4.69E-04

5.0 RESULTS

The consequence results for the dry and wet V sequences are summarized in this section. The complete output files from the 25 MACCS calculations are stored on magnetic tape and are available upon request. For the dry V sequence, five cases were analyzed, one base case and four sensitivity cases. For each case, the APBs were partitioned into two source term groups. For each source term group, a MACCS calculation was performed - a total of ten for the dry V sequence cases. Table G.8 reports three consequence measures for each source term group for each dry V sequence case. Table G.8 also includes the conditional frequency weighting for each source term group for each case as obtained from PARTITION. The three consequence measures reported here are the mean number of early fatalities and latent cancer fatalities within a 1000-mile radius, and the mean population dose, in person-rem, within a 50-mile radius. The uncertainty distributions calculated by MACCS for each measure are not reported here but can be obtained from the stored MACCS output.

Table G.9 reports similar results for the wet V sequence cases. For the wet V sequence, seven cases were analyzed, one base case and six sensitivity cases. Again, the reported consequence measures are mean values.

The mean consequences for each sensitivity case, conditional upon the occurrence of that particular V sequence, are obtained by multiplying the mean consequence measure for each source term group by the conditional probability of the source term group and summing over all source term groups:

$$CONS_k = \sum_{i=1}^n CSTGP_i * CONS_{ik}(FP_i)$$

where,

$CONS_k$ = mean magnitude of consequence measure k

$CSTGP_i$ = the conditional probability of source term group i

FP_i = fission product source term for source term group i

$CONS_{ik}$ = mean magnitude of consequence measure k for source term group i, given fission product source term (FP_i)

Table G.8 Conditional probabilities and MACCS consequence results for each source term group for each dry V sequence sensitivity.

<u>Sensitivity Case</u>		<u>Source Term Group</u>	<u>Conditional Probability</u>	<u>Mean Early Fatalities</u>	<u>Mean Latent Fatalities</u>	<u>Mean 50-Mile Dose (person-rem)</u>
<u>DF</u>	<u>Release Elevation</u>					
1.0	0.0	1	6.69E-01	1.49E+02	6.79E+03	8.05E+06
		2	3.31E-01	8.08E-01	2.47E+03	2.22E+06
5.0	0.0	1	1.61E-01	1.33E-01	6.06E+02	9.33E+05
		2	8.39E-01	1.92E+00	2.05E+03	2.38E+06
10.0	0.0	1	2.29E-01	1.11E-01	5.04E+02	8.59E+05
		2	7.71E-01	4.90E-01	1.37E+03	1.76E+06
50.0	10.0	1	8.29E-01	7.38E-02	4.34E+02	8.20E+05
		2	1.71E-01	5.86E-02	9.14E+02	1.19E+06
100.0	10.0	1	8.28E-01	6.20E-02	2.99E+02	6.41E+05
		2	1.72E-01	5.63E-02	8.70E+02	1.12E+06

Table G.9 Conditional probabilities and MACCS consequence results for each source term group for each wet V sequence sensitivity.

<u>Sensitivity Case</u>		<u>Source Term Group</u>	<u>Conditional Probability</u>	<u>Mean Early Fatalities</u>	<u>Mean Latent Fatalities</u>	<u>Mean 50-Mile Dose (person-rem)</u>
<u>DF</u>	<u>Release Elevation</u>					
Base Case ^a	0.0	1	4.87E-01	2.09E-01	6.33E+02	1.04E+06
		2	3.17E-01	1.11E+01	3.42E+03	3.88E+06
		3	1.96E-01	3.94E-01	1.63E+03	1.77E+06
5.0	0.0	1	2.14E-01	2.28E-01	5.94E+02	9.48E+05
		2	7.86E-01	1.20E+00	1.73E+03	2.02E+06
10.0	0.0	1	3.16E-01	1.96E-01	4.98E+02	8.63E+05
		2	6.84E-01	4.96E-01	1.21E+03	1.61E+06
50.0	0.0	1	8.17E-01	1.80E-01	3.40E+02	6.90E+05
		2	1.83E-01	1.59E-01	9.01E+02	1.17E+06
50.0	10.0	1	8.17E-01	1.21E-01	3.48E+02	6.98E+05
		2	1.83E-01	1.06E-01	9.13E+02	1.18E+06
100.0	0.0	1	8.17E-01	1.60E-01	2.37E+02	5.37E+05
		2	1.83E-01	1.54E-01	8.60E+02	1.10E+06
100.0	10.0	1	8.17E-01	1.07E-01	2.41E+02	5.42E+05
		2	1.83E-01	1.03E-01	8.70E+02	1.11E+06

a. Wet V sequence base case DF is a sampled distribution as given in Table G.5.

Tables G.10 and G.11 present the mean consequence results for the dry and wet V sequence sensitivity cases, respectively. These results are also presented graphically on Figures 1 through 6. Comparisons of the dry and wet V sequence consequence results are also presented graphically in Figures 7, 8, and 9.

Table G.10 Mean MACCS consequence results for each dry V sequence sensitivity.

<u>Sensitivity Case</u>		<u>Mean Early</u> <u>Fatalities</u>	<u>Mean Latent</u> <u>Fatalities</u>	<u>Mean 50-Mile</u> <u>Dose</u> <u>(person-rem)</u>
<u>DF</u>	<u>Release</u> <u>Elevation</u>			
1.0	0.0	9.99E+01	5.36E+03	6.12E+06
5.0	0.0	1.63E+00	1.82E+03	2.15E+06
10.0	0.0	4.03E-01	1.17E+03	1.55E+06
50.0	10.0	7.12E-02	5.16E+02	8.83E+05
100.0	10.0	6.10E-02	3.97E+02	7.24E+05

Table G.11 Mean MACCS consequence results for each wet V sequence sensitivity.

<u>Sensitivity Case</u>		<u>Mean Early</u> <u>Fatalities</u>	<u>Mean Latent</u> <u>Fatalities</u>	<u>Mean 50-Mile</u> <u>Dose</u> <u>(person-rem)</u>
<u>DF</u>	<u>Release</u> <u>Elevation</u>			
Base ^a	0.0	3.69E+00	1.71E+03	2.08E+06
5.0	0.0	9.92E-01	1.49E+03	1.79E+06
10.0	0.0	4.01E-01	9.85E+02	1.37E+06
50.0	0.0	1.76E-01	4.43E+02	7.78E+05
50.0	10.0	1.18E-01	4.51E+02	7.86E+05
100.0	0.0	1.59E-01	3.51E+02	6.40E+05
100.0	10.0	1.06E-01	3.56E+02	6.46E+05

a. Wet V sequence base case DF is a sampled distribution as given in Table G.5.

Figure G.1 Mean early fatality consequence results as a function of decontamination factor for the dry V sequences.

Figure G.2 Mean latent fatality consequence results as a function of decontamination factor for the dry V sequences.

Figure G.3 Mean 50-mile population dose consequence results as a function of decontamination factor for the dry V sequences.

Figure G.4 Mean early fatality consequence results as a function of decontamination factor for the wet V sequences (all release elevations are 0.0 m).

Figure G.5 Mean latent fatality consequence results as a function of decontamination factor for the wet V sequences (all release elevations are 0.0 m).

Figure G.6 Mean 50-mile population dose consequence results as a function of decontamination factor for the wet V sequences (all release elevations = 0.0 m).

Figure G.7 Comparison of the dry and wet V sequences mean early fatality consequence results as a function of decontamination factor (release elevation = 10.0 m for DFs of 50.0 and 100.0, otherwise the elevation = 0.0 m).

Figure G.8 Comparison of the dry and wet V sequences mean latent fatality consequence results as a function of decontamination factor (release elevation = 10.0 m for DFs of 50.0 and 100.0, otherwise elevation = 0.0 m).

Figure G.9 Comparison of the dry and wet V sequences mean 50-mile population dose consequence results as a function of decontamination factor (release elevation = 10.0 m for DFs of 50.0 and 100.0, otherwise elevation = 0.0 m).

Appendix H

Uncertainty Analysis

Appendix H

Uncertainty Analysis

A simplified analysis was performed to estimate the range of uncertainty in the ISLOCA core damage frequency (CDF). The analysis was limited to the uncertainty in the dominant ISLOCA scenario (scenario 3), which involved failure of the discharge check valves at the interface between the residual heat removal (RHR or ND) system and the RCS cold legs. Some scenarios (for this analysis, those with a CDF of $10^{-8}/\text{y}$ or less) were omitted from the uncertainty analysis on the basis that they were insignificant contributors to core damage frequency.

Figure I.1 shows the event tree used to quantify the dominant core damage scenario. Note that there are three REL-1g end states (sequence cutsets) on this event tree that have a frequency $> 10^{-7}/\text{y}$ and three REL-1g sequence cutsets with a frequency $\leq 10^{-9}/\text{y}$. To simplify the analysis, only the three cutsets with a frequency $> 10^{-7}/\text{y}$ were included in the uncertainty calculation. These three cutsets and their estimated frequency of occurrence are listed below.

Sequence cutset	Frequency (per reactor-year)
CV-L*OP-FTD*ND-RUPT*OP-FTC-2*REL-MIT	2.75×10^{-7}
CV-L*OP-FTD*ND-RUPT*DIAG-LOCA*REL-MIT	1.01×10^{-6}
CV-L*OP-FTD*ND-RUPT*FTD-LOCA*REL-MIT	9.75×10^{-7}

These sequence cutsets were loaded into the Integrated Reliability and Risk Assessment System (IRRAS) software package, Version 2.5. The uncertainty in the basic events that make up these cutsets was calculated as follows.

CV-L. The fault tree shown below was used to quantify event CV-L, which represents failure of a pair of discharge check valves in one of the four lines connecting the RHR system to the RCS cold legs.

Figure H.1 Event Tree for ISLOCA at RHR/RCS Cold Leg Interface

Each of the basic event inputs to the AND gates in this fault tree represents failure of a discharge check valve. Failures are assumed to be independent, that is, either valve in a pair can fail at any time, regardless of whether the other valve in the pair has failed. A constant mean failure rate, λ , of $8.7 \times 10^{-8}/\text{h}$ was used, with an error factor of 10. The distribution of λ is lognormal and the failure rates are all correlated, because the same failure data are used for each of the discharge check valves. The failure probabilities were calculated using the λr approximation to the exponential distribution, with $r = 8760$ h. The hand-calculated mean value (which was used in the event tree as a best estimate) is 1.64×10^{-5} . Using 5000 Latin hypercube samples (LHS), with a pseudorandom seed of 3571, IRRAS calculates the following parameters for the failure probability of event CV-L.

Mean	1.47×10^{-5}
Median	3.28×10^{-7}
5 th percentile	3.28×10^{-9}

95th percentile 3.28×10^{-5}

IRRAS 2.5 has the ability to generate a histogram for the calculated uncertainty data. This histogram can then be used in the calculation of sequence cutset uncertainty, obviating the need to assume a distribution for event CV-L (the distribution can be shown to be lognormal in the case being analyzed here).

OP-FTD. This event represents failure of the operator to detect overpressurization of the RHR system prior to system rupture. Given the rapid system pressurization that occurs when the discharge check valves fail, this event was assumed to have a failure probability of 1.0 with an error factor of 1 (i.e., it was treated as a point estimate).

ND-RUPT. Rupture of the RHR system was quantified in a separate analysis (see Appendix F). The rupture probability was estimated to be 0.98, with this value being used as a point estimate. This is thought to be a valid treatment, because the uncertainty that arises from the lack of knowledge of the composite logarithmic standard deviation of the failure pressure, β_c , is expected to be small, especially when compared to the large uncertainty in the initiating event, CV-L (see above).

OP-FTC-2. This event models operator failure to isolate the RHR system from the RCS following rupture of the RHR system. It was estimated by a detailed HRA task analysis (see Appendix C) to have a mean failure probability of 1.95×10^{-2} . For purposes of this uncertainty analysis, this value is interpreted as being the mean of a lognormal distribution with an error factor of 5.¹

¹As discussed in Appendix C, the human error rates obtained from THERP are customarily interpreted as being median values of lognormal distributions, with a recommendation that they be used as point estimates. For this report, these median values were converted to means, with the assumption that the underlying distribution was lognormal. This allows uncertainties in human error to be included in the estimation of the uncertainty in core damage frequency. If these median values from THERP are treated as point estimates, the uncertainty in core damage frequency changes to:

REL-MIT. This event models the mitigation of the fission product release by removal mechanisms, such as scrubbing through an overlying water pool or by auxiliary building fire protection sprays. Walkdowns of the auxiliary building and discussions with plant personnel indicate that the break is not likely to be submerged and no general area fire sprays are present in the auxiliary building. Therefore, this event was judged to have a failure probability of 1.0 with an error factor of 1.

DIAG-LOCA. This event models operator failure to correctly diagnose that an ISLOCA has occurred, given that the operators detect the LOCA successfully. It was estimated by a detailed HRA task analysis (see Appendix C) to have a mean failure probability of 0.067. This is interpreted as being the mean of a lognormal distribution with an error factor of 5.

FTD-LOCA. This event models operator failure to detect the ISLOCA, given a rupture in the RHR system. It was estimated by a detailed HRA task analysis (see Appendix C) to have a mean failure probability of 0.0607. This is interpreted as being the mean of a lognormal distribution with an error factor of 5.

The final step in the analysis is to use IRRAS to calculate the uncertainty in the overall sequence, which is described by the Boolean union of the three sequence cutsets listed above. Again, this calculation was done using 5000 LHS with a pseudorandom seed of 3571. The results are listed below.

Mean:	$1.33 \times 10^{-6}/y$
Median:	$5.52 \times 10^{-8}/y$
5 th percentile:	$7.51 \times 10^{-10}/y$
95 th percentile:	$6.62 \times 10^{-6}/y$

The uncertainty in core damage frequency is still approximately four orders of magnitude, but the mean value has been shifted to the left by a factor of approximately two. The new distribution of core damage frequency is less skewed to the right and flatter (as measured by the IRRAS-calculated coefficients of skewness and kurtosis), that is, it is more nearly symmetric about the mean value. If the point estimate values from THERP were interpreted as best estimates of the HEPs, the core damage frequency then could be interpreted as a best estimate, also.

Mean	$2.18 \times 10^{-6}/y$
Median	$5.11 \times 10^{-8}/y$
5 th percentile	$5.30 \times 10^{-10}/y$
95 th percentile	$6.88 \times 10^{-6}/y$

Appendix I

Component Failure Analysis

Pressure-Dependent Fragilities for Piping Components - Reference Westinghouse Four-Loop Plant

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1. INTRODUCTION

The work presented in this report is described in Revision 4 of the Task Action Plan for Generic Issue 105, *Interfacing System LOCA in LWRs*, dated February 13, 1990. It is anticipated that the results documented here will facilitate the resolution of this high priority generic issue in a timely manner as well as the effort being undertaken by EPRI/NUMARC.

A probabilistic risk assessment of a reference Westinghouse four-loop plant is being conducted by EG&G Idaho, Inc., to evaluate the probability of plant damage from Interfacing System Loss of Coolant Accidents (ISLOCA). Impell Corporation is under sub-contract to EG&G Idaho, Inc., to establish the pressure capacities of several low pressure systems to withstand pressures and temperatures above the design levels. The probability of failure as a function of internal pressure has been developed for the critical modes of failure of two fluid systems subject to potential overpressurization. The two systems identified by EG&G for the Reference Westinghouse four-loop Plant (RWP) are the residual heat removal system up to the pressure isolation check valves in the injection lines to the reactor coolant system, and the safety injection system on the suction side of the safety injection pumps. Included in this evaluation are the residual heat exchanger, and pumps, valves, and flanged connections for each system. The variability in the probability of failure is included, and the estimated leak rates or leak areas are given for the controlling modes of failure. For this evaluation, all failures are based on quasistatic pressures since the probability of dynamic effects resulting from such causes as water hammer have been initially judged to be negligible for the RWP ISLOCA.

The pressure capacities of the pipes and vessels are evaluated using limit state analyses for the various failure modes considered. The capacities are dependent on several factors, including the material properties, modeling assumptions, and the postulated failure criteria. A major source of uncertainty in the failure criteria is the expected strain resulting in failure. All welds are full penetration and the probability of failure at membrane strains below yield is considered to be quite low. On the other hand, biaxial strains and gage length effects as well as strain concentrations and bending significantly reduces the expected hoop strain at failure when compared to elongation data developed from standard specimen ultimate tests. Since test data from vessel tests are extremely limited, considerable variability is introduced, not only in the failure criteria but in analytical modeling and other assumptions. In particular the limited data that do exist are related to finite length cylinders with internal pressure loading only, and no test results are available for such effects as thermal or bending strains in pipe,

strain concentrations at branch connections, or nozzle loads on tanks. Since many of the base parameters are random and the methods used to evaluate the capacities are subject to some uncertainty, the pressure capacity for any failure mode is also considered to be a random variable.

A different approach must be used to evaluate the pressure capacities for gasketed flange connections, valves, and pumps. Unlike the failure modes for piping, vessels, and heat exchangers, which lend themselves to evaluation by general structural mechanics techniques, the failure modes for gasketed flange connections, valves, and pumps are very complex and evaluation must rely primarily on the results from ongoing gasket research test programs and available vendor information and test data.

It is assumed that the pressure capacities have a lognormal distribution. This assumption is made because a lognormal distribution has been shown to be a valid description of the variability in material strengths. In addition, for a random variable that can be expressed as the product and quotient of several random variables, the distribution of the dependent variable tends to be lognormal regardless of the distributions of the independent base variables.

With the pressure capacity assumed to be a lognormal random variable and denoting it as P , the probability of failure occurring at a pressure less than or equal to a specific value p is expressed as:

$$P_f = \text{Prob}(P \leq p) = \Phi \left[\frac{\ln(p/\hat{P})}{\beta_c} \right] \quad (1-1)$$

where: P_f = probability that failure occurs at a pressure $p \leq P$

P = random pressure capacity

β_c = logarithmic standard deviation of P

$\hat{P}$ = median pressure capacity

$\Phi(\cdot)$ = cumulative distribution function for a standard normal random variable

In equation (1-1), the pressure capacity for a given failure mode is probabilistically described by the following expression,

$$P = \hat{P} \cdot M \cdot S \quad (1-2)$$

in which $\hat{P}$ is the median pressure capacity, M is a lognormally distributed random variable having a unit median and a logarithmic standard deviation β_M representing the uncertainty in modeling, and S is also a lognormally distributed random variable with a unit median value and a logarithmic standard deviation β_S representing the uncertainty in the material properties. The overall uncertainty in the median capacity is obtained by taking the square root of the sum of the squares of β_M and β_S .

The median pressure capacity represents the internal pressure level for which there is a 50% probability of failure (leakage or burst) for a given failure mode. The median values are evaluated from limit state analyses for the different failure modes. The uncertainties, β_M and β_S , are associated with variability due to a lack of knowledge related to differences between the analytical model and the real structure. Modeling uncertainties are associated with the assumptions used to develop analytical models and their ability to properly represent the failure condition. The strength uncertainties are associated with variabilities related to the material resistance. Examples of the sources of strength uncertainties include: variability in steel yield and ultimate strengths, stress-strain relationships, and the influence of elevated temperatures on material strength.

Uncertainties will exist in the estimated pressure capacities due to differences between the analytical idealization of the structure and the real conditions. There are numerous possible sources of modeling uncertainties. Examples of the sources of modeling uncertainties include: assumptions used to develop the internal force distributions, failure criteria, and the use of empirical formulae. Moreover, since the uncertainties are dependent on the particular failure mode under consideration, they must be evaluated on a case-by-case basis. However, in many instances, the evaluation of these uncertainties would require very detailed analysis and/or extensive data which may not be available. As a result, it was necessary to use subjective evaluation and engineering judgment to estimate these uncertainties.

The evaluation of the median capacities and the associated variabilities for the postulated failure modes for tanks, heat exchangers and piping is discussed in Section 2 while the evaluation of median capacities and variabilities for flanged connections, valves, and pumps are discussed in Sections 3, 4, and 5, respectively.

Although the results presented here are for a reference Westinghouse four-loop plant, it should be recognized that these results must be considered site-specific based on the detail design of the heat exchanger, flange bolt material and installation torques, and plant-specific valves.

2. TANKS, HEAT EXCHANGERS, AND PIPE

The piping of pertinence for ISLOCA consideration at the RWP are fabricated from 304 Stainless Steel (i.e., SA 312, TP 304). The only vessel of importance for ISLOCA consideration at the RWP is the residual heat exchanger RS-707. The heat exchanger is fabricated from SA 285 Grade C, SA 515 Grade 70, SA 249 TP 304, and SA 240 TP 304. The material properties for SA 106 Grade B and SA 516 Grade 70 from Reference 1 were used for the first two of the above four materials. The material properties for 304 Stainless Steel from Reference 1 were used for the last two of the four materials. Ultimate strength values at room temperature up to 800°F (References 2 and 3) available in the literature were used to establish expected median strengths for the 304 SS. Table 2-1 shows the expected median ultimate strengths for room temperature, 400°F, 600°F, and 800°F, as well as the corresponding values for yield strength and elongation. Also shown for comparison are the corresponding ASME Code values. Tables 2-2 through 2-4 show similar expected median and code values for 316 SS (SA 312, TP 316, and TP 316 H for pipe), and Carbon steel (SA 106 Grade B for pipe and SA 516 Grade 70 for plate).

As is apparent from these tables, a significant margin of safety often exists when the median material properties are compared with the corresponding code (lower bound) values. In a few instances, however, the median values available in the literature are close to, or even slightly lower than, the Code value; particularly for the 800°F temperature range. In all cases, however, since the systems and components being evaluated here were designed, for the most part, for relatively low pressures and temperature, ultimate pressure capacities for these components, governed by stresses near median ultimate values rather than the code design stress intensity, S_m , or allowable stress, S , values, are expected to be well in excess of the system design pressures.

2.1 Median Cylinder and Pipe Failure Pressure Criteria

Design stresses in piping system pressure vessels include provision for stresses resulting from deadweight, thermal expansion, nozzle loads, earthquake and other loads as well as internal pressure. Stresses from other than internal pressure may constitute a major or even the controlling portion of the design allowable stress. At overpressure conditions, however, the percentage of available strength required to resist the nonpressure loads may be expected to decrease (i.e., the deadweight stress in a piping system does not increase with an increase in pressure, and while thermal stresses may increase above the design case, they are not expected to be the controlling load). Thus, the failure criteria developed for pipe

and pressure vessel burst is concentrated on the internal pressure effects, as reflected in the hoop stress in a cylinder, while still retaining some consideration for other loads such as bending or branch connections in pipe, or nozzle loads in tanks. The goal was to develop criteria which could reasonably include these additional effects without requiring a detailed evaluation in order to obtain the actual magnitudes of the bending stresses, etc., at every location and temperature.

Failure due to hoop stress in a basically unflawed cylinder can be expected when the failure strain is reached. Several factors influence the failure strain when compared to the elongation predicted from a simple uniaxial 2-inch gage specimen, however. First, an unrestrained cylinder is in a state of biaxial stress and it is known that failure strains are reduced for multiaxial stress states. Manjoine (Reference 4) suggests a triaxial reduction factor of the form:

$$T.F. = \frac{\sqrt{2}(\sigma_1 + \sigma_2 + \sigma_3)}{[(\sigma_1 - \sigma_2)^2 + (\sigma_1 - \sigma_3)^2 + (\sigma_2 - \sigma_3)^2]^{1/2}} \quad (2-1)$$

where σ_i ($i = 1,2,3$) are the principal stresses.

For an unrestrained cylinder with no bending subjected to internal pressure, the hoop stress is twice the axial stress and the maximum radial stress equal to the internal pressure is small, compared to the hoop and axial stresses. For this condition,

$$\sigma_1 = 2\sigma_2 \quad \text{and} \quad \sigma_3 \approx 0$$

and $T.F. = 1.73$

Including provision for bending such that:

$$\sigma_1 = \sigma_2 \quad \text{and} \quad \sigma_3 \approx 0$$

results in $T.F. \approx 2$.

Thus, failure strain in a cylinder fabricated from a material with uniaxial elongation, ϵ , due to hoop stress could be expected in the range of:

$$\epsilon_{T.F.} = \frac{\epsilon}{1.73} \text{ to } \frac{\epsilon}{2}$$

Another factor which can influence the failure strain in a pipe or vessel is the effective gage length. Part of the elongation reported for a 2-inch uniaxial test specimen results from necking. For longer specimens, the necking portion of the elongation remains essentially

constant so that the total elongation for longer length specimen is less (i.e., the elongation for a uniaxial 8-inch test specimen is less than for a 2-inch specimen). For a segment of pipe, the effective gage length may be expected to be of the order of the pipe circumference, and a further reduction on failure strain of the order of 1.5 to 2 is estimated. For pressure vessels with strain concentrations due to under- or over-reinforced nozzles and other discontinuities, failure is expected at somewhat lower average strains. Based on failure strain values which have been reduced to account for the triaxial and gage length effects and based on the typical stress-strain relationships for strain hardening carbon and austenitic steels, failure stresses can therefore be expected at about 0.9 and 0.85 of the ultimate uniaxial stress, respectively. Based on the limited available data, these ratios are assumed independent of temperature. Therefore, including the increased radius at failure and using the corrected stress, the median pressure for failure in a thin-walled circular cylinder can be found from the simple relationship:

$$p_f = \frac{\hat{\sigma}_f t}{r(1 + \epsilon_f)} \quad (2-2)$$

where: $\hat{\sigma}_f$ is the median failure stress
 p_f is the median failure pressure
 t is the wall thickness (nominal)
 r is the initial inside radius
 and ϵ_f is the median hoop strain at failure

This approach defines the failure pressure in terms of hoop stress which is an easy parameter to quantify but includes some provision for the biaxial stress-state and strain concentration effects.

2.2 Cylinder and Pipe Failure Pressure Variability

The variability associated with the calculated cylinder failure pressure results from a number of sources including material strength, wall thickness, variability in the stress-strain relationship, uncertainty related to the biaxial stress condition, necking and strain concentration effects on the failure strain and bending or discontinuity stresses. Most important, however, may be the existence, size, and orientation of partial through-wall flaws in the cylinder wall. To conduct a rigorous, probabilistic fracture mechanics evaluation of all the piping and vessels important for ISLOCA requires a knowledge of the crack depth, crack length, crack orientation, and fracture toughness including biaxial load and corrosion effects, both now and at end of plant life. In addition to a mean or median value of the above parameters, the statistical

distribution and coefficient of variation would be required. Current evaluations are being conducted to accumulate and evaluate this type of data, but a rigorous evaluation based on this type of data requires consideration of the stress-state at every pipe weld and component over the entire range of temperatures which is beyond the scope of the current program.

Therefore, for the current investigation, the variability was developed for a log-normal distribution by assuming a probability of failure of 0.001 corresponding to yield in the cylinder. This approach, in essence, assumes the possibility a large flaw may exist which would be required to fail the cylinder at yield but eliminates the need to determine how the flaw was developed. Since the controlling failure condition is based on hoop stress, the above assumption of failure at yield implies the controlling flaw is an axial crack although a larger circumferential crack could also lead to the same failure at yield. Whether the crack was present at fabrication or whether it was initiated and grew due to causes such as thermal fatigue, water hammer, or corrosion, is unknown and is not required with this assumption.

In essence then, the assumption of 0.001 probability of failure at yield implies a 0.001 probability of the existence of a very large flaw or combination of a large flaw with high thermal stress, etc. This is believed to be a very conservative (probably overconservative) assumption. If it is found that a significant contribution to the total risk occurs due to pipe or vessel burst at pressures well below the median burst pressures, then this assumption should be reevaluated. This would require that burst, as opposed to flange leakage, is determined to be the dominant failure mode but could easily be evaluated by sensitivity studies where the variability is based on probability of failure at yield is assumed to be 10^{-4} or 10^{-5} , for instance, or by truncating the tails of the distribution in the range of .02 to .04. If the overall risk is found to be sensitive to these assumptions, further research into the probability of expected flaw size may be necessary.

2.3 Cylinder and Pipe Failure Stress and Variability

Using the methods described in Sections 2.1 and 2.2 above, median hoop failure stresses and variabilities were developed for the materials and temperature range for both pipes and vessels. Due to the possibility of strain concentrations due to nozzles and other discontinuity stresses in tanks, somewhat lower median failure stresses, along with decreased variabilities, were used for tanks and vessels compared to pipe. Tables 2-5 through 2-7 show the failure stresses and corresponding lognormal standard deviations for the materials used in the RWP.

In general, the failure stresses decrease with temperature, but due primarily to change in the ratio of yield-to-ultimate with temperature, the lognormal standard deviation tends to increase with temperature. Note, however, the tendency of the failure stress for the low carbon steels to increase in the 400° to 600°F range. This characteristic results from the corresponding increase in ultimate strength in same temperature range, although the yield strength of the same material shows essentially a monotonic decrease with temperature (c.f., Reference 5). The lognormal standard deviations shown in Tables 2-5 through 2-7 are considered to be representative (although probably conservative) for cylinders with median pressure determined essentially for unflawed vessels but admitting the possibility of the presence of very large flaws. As such, a lognormal distribution is considered reasonable for failure pressure of the order of plus one standard deviation and below. However, at failure pressures in the high end of the distribution, the use of the lognormal distribution is inappropriate and some upper bound cutoff is required. This occurs physically since the presence of flaws can significantly reduce the failure pressure below the median unflawed cylinder, but the absence of flaws cannot further strengthen the cylinder above the assumed median unflawed cylinder capacity. This upper bound cutoff is controlled essentially by the ultimate strength of the material without including the biaxial load and strain concentration reduction effects. In essence, the upper bound cutoff is expected to correspond more closely to the failure of a cylinder with no bending, nozzles, branch connections, or flaws, and is representative of the results obtained from finite length unflawed cylinder test results. The cutoff is also not a discrete value but has a median with associated uncertainty governed by the material strength properties. These values are shown for the individual vessels in a subsequent section.

2.4 Pressure Vessel Head Failure

Buckling of pressure vessel heads is also a potential mode of failure for the RWP RS 707 residual heat exchanger. Semi-ellipsoidal heads are used for this vessel. Two modes of buckling have been identified for pressure vessels. The first is asymmetric buckling denoted by P_{cr} , and the second is plastic collapse denoted by P_o . Buckling pressure capacities for these modes of failure were calculated using the equations obtained by Galletly and his co-workers (References 6 through 9). These equations were developed from analyses results using the BOSOR-5 computer program (Reference 10). For 2:1 semi-ellipsoidal steel heads, these results may be summarized as:

$$P_{cr} = 10.4 \sigma_y \left(\frac{t}{2r} \right)^{1.25} \quad (2-3)$$

and

$$P_o = \frac{\sigma_y t (1 + 50 \epsilon_y)}{r} \quad (2-4)$$

where: σ_y is the yield stress
 t is the head thickness
 r is the radius of the attached cylinder
 and ϵ_y is the yield point strain

The ranges of parameters investigated were:

$$200 < r/t < 750$$

$$30 \text{ ksi} < \sigma_y < 60 \text{ ksi}$$

and the strain hardening slope, S , = 0, 5, and 10 percent.

The above analysis results are sensitive to the analytical modeling assumptions. For instance, for strain hardening steel heads, buckling is predicted for semi-ellipsoidal heads only when deformation theory is used. No buckling is analytically predicted using flow theory (Reference 7). Equations 2-3 through 2-4 are also conservative by about 10 percent based on analysis results. Results of tests for steel head buckling pressure capacities are very limited. Reference 6 shows a ratio of experiment to theory pressures of 1.59 to 1.97. Thus, while the above equations are considered an adequate basis for design, they appear to be excessively conservative for use in establishing median-centered buckling capacity. For use in this investigation, the analytically predicted (Equations 2-3 through 2-4) buckling pressures were increased by a factor of 1.78 to determine the expected median capacities. In all cases, the P_o plastic collapse capacity controls for steel heads.

2.5 Pressure Vessel Head Variability

The principal sources of uncertainty in determining the dished-head buckling capacities are the modeling uncertainty associated with the BOSOR-5 analyses, the variability in the yield strength over the temperature range of interest, and the expected variability in the actual buckling capacity compared to the analysis results. Reference 11 recommends a coefficient of variation 0.1 for plastic collapse of semi-ellipsoidal heads, and typical coefficients of variation on the yield strength of stainless steel at room temperature are about 0.12. From the limited test results for buckled heads, a coefficient of variation of about 0.11 was developed to account for the uncertainty in actual buckling capacity. Assuming a lognormal distribution, the overall lognormal standard deviation at room temperature was determined as:

$$(0.1^2 + 0.12^2 + 0.11^2)^{1/2} = 0.19$$

At higher temperatures, increased variability in the yield strength coefficient of variation (Reference 3) results in increased overall variability. Table 2-8 shows the lognormal standard deviations developed over the temperature range of interest.

It should be noted that buckling does not necessarily result in the formation of a leak in a ductile, thin shell. The dished-heads under consideration here are in a net compression membrane stress-state which tends to substantially reduce the probability of a leak occurring, even in the post-buckled condition. It is estimated that the conditional probability of crack formation is about 0.2, given buckling occurs.

2.6 Pipe and Vessel Leak Areas

Some conditions exist where the presence of large flaws could result in leaks rather than uncontrolled bursts at the failure pressures being considered here. For the most part, however, once the pressure exceeds yield, uncontrolled leaks are expected for tanks and pipes. Even if the flaw configuration is such that theoretically a "leak" occurs before "break", the leak area is large in the context of ISLOCA.

One exception is the asymmetric buckling of the dished-heads. As previously noted, the head is in a state of membrane compression such that crack formation is expected only in regions of high local bending in the post-buckled condition. To the writers' knowledge, there is essentially no experimental data on crack size for this mode of failure. Consequently, it was estimated that the crack width for this condition is expected to be in the range of ten times the computed crack opening displacement (COD) computed using methods such as those described in Reference 12. Crack lengths were estimated to range from a few inches up to about the radius of the cylinder to which the head is attached. Leak areas were taken as the product of the crack width and the crack length. As a result, the lower and upper bound leak areas corresponded to the estimated minimum and maximum crack lengths. The median leak area was evaluated based on the upper and lower bound leak areas representing a $\pm 3.09\beta$ variation. These assumptions lead to median crack leak areas of the order of a square inch or less for this mode of failure. It should be noted, however, that if the crack progresses to a region of net membrane tension such as the cylinder, large, uncontrolled leak areas may be expected to occur, even if the crack initiates in the knuckle region of the head. However, this effect was not included in the estimation of the leak areas associated with the asymmetric buckling of the dished heads.

2.7 RS 707 Residual Heat Exchanger Capacity

In this section, the pressure capacity of the RWP RS 707 heat exchanger is presented for the important failure modes, together with the expected variabilities in terms of the lognormal standard deviations. Failure pressures are given for metal temperatures ranging from room temperature to 800°F. Interpolation may be used to determine failure pressure and variability at intermediate temperatures. Also shown are upper bound cutoffs and associated variabilities, as well as expected leak areas. In general, leak areas are considered independent of temperature. Both shell and tube side failure pressures are presented.

The RS 707 residual heat exchanger is a U-tube heat exchanger with 43.75-inch outside diameter. The shell side cylinder is fabricated from 1/2-inch SA 285 Grade C plate with a 1/2-inch 2:1 semi-ellipsoidal head fabricated from SA 515 Grade 70. The tube side cylinder is fabricated from 0.875-inch SA 240 TP 304 plate and the head is of the same thickness and material. The tubes have outside diameter of 3/4 inch and are fabricated from SA 249 TP 304 material. Two 18-inch, 150# flanges are located on the shell side. A 44-inch tube sheet flange is located between the tube and shell sides. Median pressure capacities and expected variabilities for both the shell and tube side cylinders and dished-heads are shown in Table 2-9.

2.8 Pipe

All pipe of concern for the RWP ISLOCA investigation is fabricated from SA 312 TP 304 stainless steel. Median burst pressures for the temperature range of interest are presented for various assumed levels of corrosion in Tables 2-10 through 2-12. The associated variabilities at these temperatures are given in Table 2-5. Note that the variabilities given in Table 2-5 are for failure stress but can also be used for failure pressure since the contribution of other elements to the total variability is expected to be negligible.

Table 2-1. 304 Stainless Steel Material Properties

Temperature (°F)	Ultimate Strength (ksi)		Yield Strength (ksi)		Elongation (%) (2-inch gage)	
	Median	Code*	Median	Code	Median	Code
R.T.	86	75	37	30	60	35 (long.) 25 (trans.)
400	74	64.4	23	20.7	53	
600	70	63.5	19.5	18.2	49	
800	65.5	62.7	16.5	16.8	46	

* SA 312, TP 304, and TP 304 H

Table 2-2. 316 Stainless Steel Material Properties

Temperature (°F)	Ultimate Strength (ksi)		Yield Strength (ksi)		Elongation (%) (2-inch gage)	
	Median	Code*	Median	Code	Median	Code
R.T.	86.5	75	42	30	46	35 (long.) 25 (trans.)
400	81.5	71.8	35	21.4	43	
600	77.5	71.8	31	18.8	41.5	
800	73	70.9	28	17.6	40	

* SA 312, TP 316, and TP 316 H

Table 2-3. Carbon Steel Pipe Material Properties

Temperature (°F)	Ultimate Strength (ksi)		Yield Strength (ksi)		Elongation (%) (2-inch gage)	
	Median	Code*	Median	Code	Median	Code**
R.T.	68	60	36	35	36	22 (long.) 12 (trans.)
400	72	60	31	30	25	
600	69	60	27	25.9	33	
800	55	55.1	24	23.3	40	

* SA 106 Grade B

** Standard Specimen

Table 2-4. Carbon Steel Plate Material Properties

Temperature (°F)	Ultimate Strength (ksi)		Yield Strength (ksi)		Elongation (%) (2-inch gage)	
	Median	Code*	Median	Code	Median	Code
R.T.	84	70	46	38	27	21
400	87	70	40	32.6	16	
600	85	70	37	28.1	25	
800	71	64.3	34	25.3	34	

* SA 516 Grade 70

Table 2-5. Failure Stresses and Variability for 304 SS

Temperature (°F)	Pipe		Vessel	
	$\hat{\sigma}_f$ (ksi)	β	σ_f (ksi)	β
R.T.	74	0.22	67	0.19
400	62.9	0.33	57.9	0.30
600	59.5	0.36	54.6	0.33
800	55.7	0.39	51.2	0.37

where $\hat{\sigma}_f$ denotes the median hoop failure stress

Table 2-6. Failure Stress and Variability for SA 106 B Pipe

Temperature (°F)	$\hat{\sigma}_f$ (ksi)	β
R.T.	61.2	0.17
400	64.8	0.24
600	62.1	0.27
800	49.5	0.23

where $\hat{\sigma}_f$ denotes the median hoop failure stress

Table 2-7. Failure Stress and Variability for SA 516 Grade 70 Vessels

Temperature (°F)	$\hat{\sigma}_f$ (ksi)	β
R.T.	75.6	0.16
400	78.3	0.22
600	76.5	0.24
800	63.9	0.20

Table 2-8. Lognormal Standard Deviations for Dished-Head Buckling

Temperature (°F)	Lognormal Standard Deviation, β
R.T.	0.19
400	0.23
600	0.23
800	0.24

**Table 2-9. RS 707 Residual Heat Exchanger
Median Failure Pressures and Variabilities**

Shell Side Cylinder					Shell Side Head Buckling			
Temp.	$\bar{P}$	β	P_U	β	$\bar{P}_{CR}$	β	$\bar{P}_o$	β
R.T.	1350	0.17	1500	0.07	3280	0.22	2260	0.22
400°F	1460	0.24	1620	0.07	2850	0.25	1950	0.25
600°F	1370	0.27	1530	0.08	2630	0.27	1800	0.27
800°F	1070	0.23	1190	0.08	2420	0.30	1660	0.30

Tube Buckling Pressure >> Shell Side Cylinder

(2) 18" 150# Flanges on Shell Side Cylinder

Tube Sheet Flange 44" Diameter

Tube Side Cylinder					Tube Side Head Buckling			
Temp.	$\bar{P}$	β	P_U	β	$\bar{P}_{CR}$	β	$\bar{P}_o$	β
R.T.	2330	0.19	2990	0.07	5420	0.19	3200	0.19
400°F	2140	0.30	2730	0.07	3370	0.23	1950	0.23
600°F	2040	0.33	2610	0.08	2860	0.23	1650	0.23
800°F	1910	0.37	2450	0.08	2420	0.24	1390	0.24

Table 2-9. RS 707 Residual Heat Exchanger (Continued)
Median Leak Areas and Variabilities

TUBE FAILURE PRESSURE >> TUBE SIDE CYLINDER

	Median Leak Area	β
	$\hat{A}$ (inch ²)	
Cylinder	Uncontrolled	—
Head*, P_{CR}	0.7	0.5
Head*, P_o	Uncontrolled	—

where $\hat{P}$ is median cylinder rupture pressure (psig)
 P_U is upper bound cylinder rupture pressure (psig)
 P_{CR} is median asymmetric head buckling pressure (psig)
 $\hat{P}_o$ is median plastic collapse head buckling pressure (psig)

*Assume: 0.2 probability of crack formation given head buckling occurs.

Table 2-10. 304 Stainless Steel Pipe Failure Pressures

CORROSION ALLOWANCE = 0.000

Pipe Size (in)	Schedule	OD (in)	ID (in)	FAILURE PRESSURES			
				70°F	400°F	600°F	800°F
1-1/2	40S	1.900	1.610	9968	8556	7963	7525
	80	1.900	1.500	14757	12666	11788	11140
	160	1.900	1.337	23303	20001	18615	17591
2	40S	2.375	2.067	8246	7078	6587	6225
	80	2.375	1.939	12443	10680	9940	9393
	160	2.375	1.689	22476	19292	17955	16967
3	10S	3.500	3.260	4074	3497	3254	3075
	40S	3.500	3.068	7792	6688	6225	5882
	80	3.500	2.900	11449	9827	9146	8643
	160	3.500	2.624	18474	15857	14758	13946
4	10S	4.500	4.260	3118	2676	2490	2353
	40S	4.500	4.026	6515	5592	5205	4918
	80	4.500	3.826	9749	8367	7787	7359
	160	4.500	3.438	17094	14672	13655	12904
6	10S	6.625	6.357	2333	2002	1864	1761
	40S	6.625	6.065	5110	4386	4082	3857
	80	6.625	5.761	8299	7123	6630	6265
	120	6.625	5.501	11307	9705	9032	8536
	160	6.625	5.189	15314	13145	12234	11561
8	10S	8.625	8.329	1967	1688	1571	1485
	20	8.625	8.125	3405	2923	2720	2571
	40S	8.625	7.981	4465	3833	3567	3371
	80	8.625	7.625	7258	6229	5798	5479
	120	8.625	7.189	11054	9488	8830	8344
	140	8.625	7.001	12837	11018	10254	9690
	160	8.625	6.813	14718	12633	11757	11110
10	10S	10.750	10.420	1753	1504	1400	1323
	20	10.750	10.250	2699	2317	2156	2038
	40S	10.750	10.020	4032	3460	3221	3043
	80	10.750	9.564	6862	5890	5482	5180
	120	10.750	9.064	10294	8835	8223	7770
	140	10.750	8.750	12649	10857	10104	9548
	160	10.750	8.500	14649	12573	11702	11058
12	10S	12.750	12.390	1608	1380	1284	1214
	20	12.750	12.250	2259	1939	1804	1705
	Std	12.750	12.000	3459	2969	2763	2611
	40	12.750	11.938	3764	3231	3007	2841
	80	12.750	11.376	6684	5737	5339	5046
	120	12.750	10.750	10296	8837	8224	7772
	140	12.750	10.500	11858	10178	9473	8952
	160	12.750	10.126	14340	12308	11455	10825
14	10S	14.000	13.624	1527	1311	1220	1153
	20	14.000	13.375	2586	2220	2066	1952
	Std	14.000	13.250	3132	2689	2502	2365
	40	14.000	13.125	3689	3167	2947	2785
	80	14.000	12.500	6641	5700	5305	5013
	120	14.000	11.814	10240	8789	8180	7730
	140	14.000	11.500	12030	10326	9610	9081
	160	14.000	11.188	13909	11938	11111	10500

Calculation 304P000

Table 2-10. 304 Stainless Steel Pipe Failure Pressures (Continued)

CORROSION ALLOWANCE = 0.000

Pipe Size (in)	Schedule	OD (in)	ID (in)	FAILURE PRESSURES			
				70°F	400°F	600°F	800°F
16	10S	16.000	15.624	1332	1143	1064	1005
	20	16.000	15.375	2250	1931	1797	1698
	Std	16.000	15.250	2722	2336	2174	2054
	40	16.000	15.000	3689	3167	2947	2785
	80	16.000	14.314	6518	5595	5207	4920
	120	16.000	13.564	9939	8530	7939	7502
	140	16.000	13.124	12127	10409	9687	9154
	160	16.000	12.814	13759	11810	10991	10387
18	10S	18.000	17.624	1181	1013	943	891
	20	18.000	17.375	1991	1709	1590	1503
	Std	18.000	17.250	2406	2065	1922	1816
	40	18.000	16.876	3686	3164	2944	2782
	80	18.000	16.126	6431	5520	5137	4855
	120	18.000	15.250	9979	8565	7972	7533
	140	18.000	14.876	11621	9975	9283	8773
	160	18.000	14.433	13677	11739	10925	10324
20	10S	20.000	19.564	1233	1059	985	931
	20S	20.000	19.250	2156	1851	1722	1628
	40	20.000	18.814	3488	2994	2787	2633
	80	20.000	17.938	6361	5460	5082	4802
	120	20.000	17.000	9766	8382	7801	7372
	140	20.000	16.500	11739	10075	9377	8861
	160	20.000	16.064	13559	11638	10831	10236
24	10S	24.000	23.500	1177	1011	941	889
	20S	24.000	23.250	1785	1532	1426	1348
	40	24.000	22.626	3361	2884	2684	2537
	80	24.000	21.564	6251	5366	4994	4719
	120	24.000	20.376	9842	8448	7862	7430
	140	24.000	19.876	11482	9855	9172	8668
	160	24.000	19.314	13426	11524	10725	10135

Calculation 304P000

Table 2-11. 304 Stainless Steel Pipe Failure Pressures

CORROSION ALLOWANCE = 0.020

Pipe Size (in)	Schedule	OD (in)	ID (in)	FAILURE PRESSURES			
				70°F	400°F	600°F	800°F
1-1/2	40S	1.900	1.610	8593	7375	6864	6487
	80	1.900	1.500	13281	11399	10609	10026
	160	1.900	1.337	21647	18580	17292	16341
2	40S	2.375	2.067	7175	6158	5732	5416
	80	2.375	1.939	11302	9700	9028	8532
	160	2.375	1.689	21166	18167	16908	15978
3	10S	3.500	3.260	3395	2914	2712	2563
	40S	3.500	3.068	7071	6069	5648	5338
	80	3.500	2.900	10686	9172	8536	8067
	160	3.500	2.624	17631	15133	14084	13309
4	10S	4.500	4.260	2598	2230	2075	1961
	40S	4.500	4.026	5966	5120	4765	4503
	80	4.500	3.826	9170	7871	7325	6922
	160	4.500	3.438	16450	14119	13141	12418
6	10S	6.625	6.357	1985	1704	1585	1498
	40S	6.625	6.065	4745	4072	3790	3582
	80	6.625	5.761	7915	6794	6323	5975
	120	6.625	5.501	10905	9360	8711	8232
	160	6.625	5.189	14888	12778	11893	11239
8	10S	8.625	8.329	1701	1460	1359	1284
	20	8.625	8.125	3133	2689	2503	2365
	40S	8.625	7.981	4188	3595	3345	3161
	80	8.625	7.625	6967	5980	5566	5259
	120	8.625	7.189	10746	9223	8584	8112
	140	8.625	7.001	12521	10747	10002	9452
	160	8.625	6.813	14393	12354	11498	10865
10	10S	10.750	10.420	1540	1322	1230	1163
	20	10.750	10.250	2484	2132	1984	1875
	40S	10.750	10.020	3811	3271	3044	2877
	80	10.750	9.564	6631	5691	5297	5006
	120	10.750	9.064	10049	8625	8028	7586
	140	10.750	8.750	12396	10640	9902	9357
	160	10.750	8.500	14388	12349	11494	10861
12	10S	12.750	12.390	1429	1227	1142	1079
	20	12.750	12.250	2078	1784	1660	1569
	Std	12.750	12.000	3274	2810	2616	2472
	40	12.750	11.938	3579	3072	2859	2701
	80	12.750	11.376	6489	5570	5184	4899
	120	12.750	10.750	10090	8660	8060	7617
	140	12.750	10.500	11648	9997	9304	8793
	160	12.750	10.126	14122	12121	11281	10660
14	10S	14.000	13.624	1365	1171	1090	1030
	20	14.000	13.375	2420	2077	1933	1827
	Std	14.000	13.250	2965	2545	2369	2238
	40	14.000	13.125	3521	3022	2812	2658
	80	14.000	12.500	6464	5548	5163	4879
	120	14.000	11.814	10052	8628	8030	7588
	140	14.000	11.500	11838	10160	9456	8936
	160	14.000	11.188	13711	11768	10953	10350

Calculation 304P020

Table 2-11. 304 Stainless Steel Pipe Failure Pressures (Continued)

CORROSION ALLOWANCE = 0.020

Pipe Size (in)	Schedule	OD (in)	ID (in)	FAILURE PRESSURES			
				70°F	400°F	600°F	800°F
16	10S	16.000	15.624	1190	1021	951	898
	20	16.000	15.375	2106	1807	1682	1589
	Std	16.000	15.250	2576	2211	2058	1945
	40	16.000	15.000	3542	3040	2829	2674
	80	16.000	14.314	6364	5462	5083	4804
	120	16.000	13.564	9775	8390	7809	7379
	140	16.000	13.124	11958	10264	9553	9027
	160	16.000	12.814	13586	11661	10853	10256
18	10S	18.000	17.624	1055	906	843	796
	20	18.000	17.375	1863	1599	1488	1407
	Std	18.000	17.250	2278	1955	1819	1719
	40	18.000	16.876	3555	3051	2839	2683
	80	18.000	16.126	6294	5402	5028	4751
	120	18.000	15.250	9834	8441	7856	7424
	140	18.000	14.876	11473	9847	9165	8660
	160	18.000	14.433	13523	11607	10803	10208
20	10S	20.000	19.564	1120	961	895	846
	20S	20.000	19.250	2041	1752	1630	1541
	40	20.000	18.814	3371	2893	2693	2545
	80	20.000	17.938	6238	5354	4983	4709
	120	20.000	17.000	9636	8270	7697	7274
	140	20.000	16.500	11604	9960	9270	8760
	160	20.000	16.064	13421	11520	10721	10132
24	10S	24.000	23.500	1083	930	865	818
	20S	24.000	23.250	1690	1450	1350	1276
	40	24.000	22.626	3263	2800	2606	2463
	80	24.000	21.564	6149	5278	4912	4642
	120	24.000	20.376	9734	8355	7776	7348
	140	24.000	19.876	11371	9760	9083	8584
	160	24.000	19.314	13312	11426	10634	10049

Calculation 304P020

Table 2-12. 304 Stainless Steel Pipe Failure Pressures

CORROSION ALLOWANCE = 0.040

Pipe Size (in)	Schedule	OD (in)	ID (in)	FAILURE PRESSURES			
				70°F	400°F	600°F	800°F
1-1/2	40S	1.900	1.610	7218	6195	5766	5449
	80	1.900	1.500	11806	10133	9431	8912
	160	1.900	1.337	19992	17159	15970	15091
2	40S	2.375	2.067	6104	5239	4876	4608
	80	2.375	1.939	10160	8721	8116	7670
	160	2.375	1.689	19855	17042	15861	14988
3	10S	3.500	3.260	2716	2331	2170	2050
	40S	3.500	3.068	6349	5450	5072	4793
	80	3.500	2.900	9923	8517	7927	7491
	160	3.500	2.624	16787	14409	13410	12672
4	10S	4.500	4.260	2078	1784	1660	1569
	40S	4.500	4.026	5416	4648	4326	4088
	80	4.500	3.826	8592	7374	6863	6486
	160	4.500	3.438	15807	13567	12627	11932
6	10S	6.625	6.357	1637	1405	1307	1235
	40S	6.625	6.065	4380	3759	3499	3306
	80	6.625	5.761	7531	6464	6016	5685
	120	6.625	5.501	10502	9014	8390	7928
	160	6.625	5.189	14461	12412	11552	10917
8	10S	8.625	8.329	1435	1232	1146	1083
	20	8.625	8.125	2861	2455	2285	2159
	40S	8.625	7.981	3911	3357	3124	2952
	80	8.625	7.625	6677	5731	5334	5040
	120	8.625	7.189	10438	8959	8338	7880
	140	8.625	7.001	12204	10475	9749	9213
	160	8.625	6.813	14068	12075	11238	10620
10	10S	10.750	10.420	1328	1140	1061	1002
	20	10.750	10.250	2268	1946	1811	1712
	40S	10.750	10.020	3590	3081	2868	2710
	80	10.750	9.564	6400	5493	5112	4831
	120	10.750	9.064	9805	8416	7833	7402
	140	10.750	8.750	12143	10422	9700	9166
	160	10.750	8.500	14128	12126	11286	10665
12	10S	12.750	12.390	1251	1073	999	944
	20	12.750	12.250	1897	1628	1516	1432
	Std	12.750	12.000	3090	2652	2468	2332
	40	12.750	11.938	3393	2912	2711	2561
	80	12.750	11.376	6295	5403	5028	4752
	120	12.750	10.750	9884	8483	7895	7461
	140	12.750	10.500	11437	9816	9136	8633
	160	12.750	10.126	13903	11933	11106	10495
14	10S	14.000	13.624	1202	1032	960	908
	20	14.000	13.375	2255	1935	1801	1702
	Std	14.000	13.250	2798	2402	2235	2112
	40	14.000	13.125	3352	2877	2678	2530
	80	14.000	12.500	6287	5396	5022	4746
	120	14.000	11.814	9865	8467	7880	7447
	140	14.000	11.500	11645	9995	9302	8791
	160	14.000	11.188	13513	11599	10795	10201

Calculation 304P040

Table 2-12. 304 Stainless Steel Pipe Failure Pressures (Continued)

CORROSION ALLOWANCE = 0.040

Pipe Size (in)	Schedule	OD (in)	ID (in)	FAILURE PRESSURES			
				70°F	400°F	600°F	800°F
16	10S	16.000	15.624	1048	900	837	791
	20	16.000	15.375	1962	1684	1567	1481
	Std	16.000	15.250	2431	2087	1942	1835
	40	16.000	15.000	3394	2913	2711	2562
	80	16.000	14.314	6209	5329	4960	4687
	120	16.000	13.564	9612	8250	7678	7256
	140	16.000	13.124	11790	10119	9418	8900
	160	16.000	12.814	13414	11513	10715	10126
18	10S	18.000	17.624	929	798	742	702
	20	18.000	17.375	1736	1490	1387	1310
	Std	18.000	17.250	2149	1845	1717	1623
	40	18.000	16.876	3423	2938	2735	2584
	80	18.000	16.126	6156	5284	4918	4647
	120	18.000	15.250	9689	8316	7740	7314
	140	18.000	14.876	11324	9719	9046	8548
	160	18.000	14.433	13370	11475	10680	10093
20	10S	20.000	19.564	1007	864	804	760
	20S	20.000	19.250	1926	1653	1539	1454
	40	20.000	18.814	3253	2792	2599	2456
	80	20.000	17.938	6115	5248	4884	4616
	120	20.000	17.000	9505	8158	7593	7175
	140	20.000	16.500	11470	9845	9163	8659
	160	20.000	16.064	13284	11401	10611	10028
24	10S	24.000	23.500	989	849	790	747
	20S	24.000	23.250	1595	1369	1274	1204
	40	24.000	22.626	3165	2716	2528	2389
	80	24.000	21.564	6046	5189	4830	4564
	120	24.000	20.376	9625	8261	7689	7266
	140	24.000	19.876	11259	9664	8994	8499
	160	24.000	19.314	13197	11327	10542	9962

Calculation 304P040

3. GASKETED FLANGE CONNECTIONS

3.1 Introduction

Although most of the piping joints in the residual heat removal and the safety injection systems are full penetration butt welds, a number of gasketed flange connections are required for the installation and maintenance of flow restricting orifices, flow elements, and major equipment components. The elements of the flanged joints are defined in References 13 and 14 and include standard ANSI B16.5 flanges with asbestos-filled, spiral-wound gaskets.

The lines designed for 150, 300, and 600lb rated service employ raised face flanges fabricated from Type 304 stainless steel and are secured by means of SA 193-B7 bolts or studs with minimum room temperature yield and ultimate strength of 105,000 and 125,000 psi, respectively.

Table 2(d) of the flange gasket removal and replacement specification for the RWP (Reference 14) provides recommended torque values for class B through H stainless steel flanges. These recommended torque values result in the bolts being prestressed in the range of 26,000 to 30,000 psi. The specification also provides the maximum allowable torque values which at 200°F are about twice the recommended torque values and therefore correspond to bolt prestress values of 52,000 to 60,000 psi. Based on this, the 150lb rated flanges were considered to have bolts prestressed in the range of 15,000 to 30,000 psi while 300 to 600lb rated flanges were considered to have bolts prestressed in the range of 30,000 to 40,000 psi.

3.2 Variables Affecting Flanged Joint Leakage

The behavior of gasketed flanges under pressure and temperature conditions is quite complex. The propensity for leakage under a given pressure loading is as much or more dependent upon the previous history of the joint than it is on its state at the time the pressure is applied. As a result, numerous variables are introduced. These include:

- Bolt/Stud Preload
- Bolt/Stud Temperature
- Bolt/Stud 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

3.2.1 Bolt/Stud Preload

As noted above there are a range of preload stress values which satisfy the requirements of the installation specification for the various flange pressure ratings. For 150lb rated flanges the range is 15,000 to 30,000 psi based on area at the thread root. However, it will be shown in results presented later that a bolt preload of 15,000 psi provides an insufficient load to properly seat the gasket for the system operating conditions. In addition, a bolt preload of 20,000 psi provides a sufficient gasket seating load for only small flanges less than about 2-1/2 inches in diameter. For larger flanges, leakage would probably be observable at the operating and hydrotest conditions. Thus, for 150lb flanges, a bolt preload stress of 25,000 psi was taken as a median value with the range from 25,000 to 30,000 psi taken to represent a $+2.33\beta$ variation.

For 300 and 600lb flanges, the range of allowable bolt preload stress is 30,000 to 40,000 psi, again based on thread root area. Since SA193-B7 bolts with a minimum yield stress of 105,000 psi are used in the lower pressure safety systems under investigation, the bolts in these flanges are not initially stressed beyond yield. Since bolt preload stresses in the specified range all provide sufficient load to properly seat the gasket, a preload stress of 35,000 psi was taken as a median value with the range from 30,000 to 35,000 psi taken to represent a -2.33β variation.

It should be remembered that the uncertainty variability is related to the parameter of interest such as leak rate or leak area. Thus, the variability is calculated from the variation in say, leak rate resulting from the variations in initial bolt stress noted above.

3.2.2 Bolt/Stud Temperature

The maximum operating temperatures for the low pressure portions of the safety systems under consideration range from 150 to 280°F. As a result, the bolt temperature is likely less than 200°F during normal operation. However, during the ISLOCA event, the reactor coolant system pressure and temperature conditions of 2250 psi and 650°F can propagate back through the initially cold, non-operating systems. Based on preliminary analyses of the system, it is our understanding that the pressure propagates much more rapidly than the fluid temperature. Specifically, a relatively large leak at or downstream (reverse flow condition) of the flange connection must occur in order for the flange or flange bolting temperatures to rise

substantially. In addition, for higher temperature conditions in the pipe, flow must continue for a relatively long period of time before the bolting temperature will rise substantially. Thus, it was judged that potential flange leakage will most likely occur under high pressure-low temperature conditions and that leak rates and leak areas will increase somewhat as the flange and bolt temperatures increase. Based on these considerations, a bolt temperature of approximately 140°F was taken to represent the median case and the effect of higher bolt temperatures was not considered. Evaluation of higher bolt temperatures, should they occur, can be modeled employing lower elastic modulus, yield strength, and ultimate strength values.

3.2.3 Bolt/Stud Yield Strength

Consistent with the selected 140°F median value for the flange bolt/stud temperature, a value of 111,700 psi was taken as the median bolt yield strength. The ASME Code minimum value of 101,500 psi was taken to represent a -2.33β variation.

3.2.4 Bolt/Stud Stress-Strain Relationship

For the pressure values of interest in this study the bolt stresses are always below the bolt yield stress, hence, the stress-strain quantity needed is the modulus of elasticity for which the ASME Code minimum value of 29,000 ksi at 200°F was conservatively used.

3.2.5 Bolt Relaxation

For stainless steel bolts which are initially torqued to prestress levels exceeding the material yield strength, some relaxation will occur. This relaxation requires substantial time, particularly at the relatively low bolt temperatures experienced during normal plant operation. Relaxation during the course of the ISLOCA event is judged to be negligible. Since the bolts used at the RWP are not initially torqued to prestress levels exceeding the material yield strength, bolt relaxation was not included in this study.

3.2.6 Flange Flexibility

The flanged joint consists of the flange, the flange bolting, and the gasket. To study specific variables most test programs isolate one or more of the joint elements, such as the flexibility of the flange. However, to properly characterize the overall joint behavior, all three elements must function as part of an integral unit. In order to evaluate the joint behavior, three axisymmetric elastic finite element joint models (Reference 15) were developed. The models were for a 4"-300# flange, a 12"-300# flange, and a 4"-150# flange and included the flange and pipe structure, bolt stiffness, and gasket unloading stiffness. Material properties were taken at 200°F. As expected, it was found that flange flexibility affects the way that the

pressure load is carried by the gasket and bolting. As the relative flange stiffness increases, a greater portion of the pressure load goes to loading the bolting and a lesser portion goes to unloading the gasket. Thus, for larger flanges or more flexible flanges (lower pressure rating), the portion of the pressure load going to the unloading of the gasket increases. This can be seen in Figure 3-1 which plots the results of the three analyses. In this figure, the ratio of the load removed from the gasket to the total pressure load is plotted versus nominal flange size. For the elastic case of the 4"-300# flange about 43% of the total pressure load goes to unloading the gasket while the remaining 57% is carried by increased bolt load. The ratio for the larger 12"-300# flange is approximately 86% and the ratio for the lighter weight 4"-150# flange is about 59%. It is likely that the variation of the ratio with flange size is more complex than the linear variation shown but project time constraints limited the scope of this supporting study. Due to the lack of specific data, the maximum value of the incremental gasket load to total pressure load ratio was set at 1.0. However, for flanges which are flexible relative to the gasket, experience has shown that the ratio can be greater than 1.0 such that an increase in pressure results in a decrease in flange bolt tension.

High strength flange bolts (i.e., SA 193-B7) were used at the RWP. Hence, some flange inelastic behavior can be expected at higher pressures. The inelastic distortion of the flange at the gasket is expected to affect the calculated leak areas. It is recommended that further study of the effect of flange flexibility be undertaken and incorporated if leakage through gasketed flange connections is found to be a major contributor to overall plant risk for the ISLOCA initiating event.

3.2.7 Initial Gasket Stress

Over the past several years, the Pressure Vessel Research Committee (PVRC) of the Welding Research Council has sponsored a major ongoing gasket test program as part of its Long Range Flanged Joint Improvement Program. Most of the tests have been conducted with nitrogen or helium as the test fluid; however, a limited data set exists for tests using water. Some of the results of the test program are reported in References 16 through 19. These results clearly indicate that the leak resistance of a gasketed flange joint is a function of the initial level to which the gasket is stressed during the preloading of the flange bolting. The higher the initial gasket stress, the greater the leak resistance. The gasket stress versus deflection curve (Figure 3-2) and the mass leak rate versus gasket stress curve (Figure 3-3) for a typical spiral-wound gasket are both characterized by the presence of a "knee" at a gasket stress of approximately 5,000 psi. Above 5,000 psi the leak rate drops more rapidly with

increasing stress indicating improved sealing performance. Thus, although the controlled variable in assembling a flanged connection is bolt preload, it is the resulting gasket stress which determines the leak resistance of the joint for the pressure loading.

3.2.8 Gasket Loading Stiffness

The loading stiffness parameter is of importance since it determines whether or not the flange raised face bottoms out on the 0.125" thick compression gauge ring due to the bolt preload. Spiral-wound gaskets used for virtually all applications are fabricated to the requirements of Military Specification MIL-G-21032E, including Amendment 2 (Reference 20). This standard specifies the test load and corresponding deflection for each gasket size and service rating. Gaskets with an initial nominal thickness of 0.175" are to be compressed to a thickness of 0.130 ± 0.005 " under the specified test load while gaskets with an initial nominal thickness of 0.125" are to be compressed to a thickness of 0.100 ± 0.005 ". Figure 3-2 shows the gasket stress versus deflection for the loading sequence of a typical spiral-wound gasket and Table 3-1 presents the range of gasket stiffness (expressed as gasket stress per inch of deflection) which meet the specification. All of the gaskets used in the RWP flange connections have a nominal thickness of 0.175". The Reference 20 specified test load is shown together with the resulting bolt stress (based on bolt stress area). Test loads greater than 557000 lbs are not specified since this load is judged to be the practical limit of testing facilities. However, since the specified test load corresponds to the load resulting from prestressing the flange bolts to approximately 30,000 psi (based on bolt thread root area), the stiffness for larger size gaskets can also be computed. 300 and 600lb rated gaskets are interchangeable for sizes 3" and smaller and therefore, the stiffnesses are identical. The stiffnesses associated with the nominal compressed thickness ($T_g=0.130$) were taken to be median centered and the range from a compressed thickness of 0.125 to 0.135" was considered to represent a $\pm 2.33\sigma$ variation. In this study, for the sake of economy in the computing effort, the gasket stiffnesses for 150lb rated flanges were conservatively used for the 300 and 600lb rated gaskets as well.

Based upon the definition of the gasket test load noted above, prestressing the flange bolts to less than 30,000 psi will not result in a lock-up between the flange and the compression gauge ring. However, prestressing the bolts to 30,000 psi or greater may or may not result in lock-up depending on the gasket loading stiffness.

3.2.9 Gasket Unloading/Reloading Stiffness

The unloading stiffness characterizes the recovery of the gasket as the gasket stress is reduced due to increase in pressure, bolt relaxation, or other means. Figure 3-2 depicts the loading and unloading/reloading behavior of a typical spiral-wound gasket. Review of the available test data indicates that the recovery behavior for all spiral wound gaskets is quite similar and can reasonably be expressed as an unloading/reloading stiffness of about 1,000,000 psi/inch. This appears to be a reasonable value for new gaskets such as those used in the test program. However, hardening due to gasket aging may substantially increase the gasket unloading/reloading stiffness, resulting in a decrease of the Gross Leak Pressure and an increase in the joint leak rate. It is important to note that reloading of the gasket due to removal of the pressure load, for example, follows the unloading stiffness curve.

3.2.10 Gasket Creep and Relaxation

It is understood from experience that gaskets behave nonlinearly and that they creep, even at room temperature. Until recently, information available on the creep and relaxation of commonly-used fabricated gaskets was scarce. Such information is vital for the proper understanding of the behavior of bolted flange joints. The previously mentioned PVRC-sponsored test program provided a vehicle for gathering creep and relaxation data for spiral-wound gaskets which is reported in Reference 19. The results indicate that maximum creep occurs at the lower stress levels and is particularly extensive at about 5,000 psi which coincides with the yield plateau in the stress-deflection diagram. For constant stress, most of the creep occurs in the first 10 to 15 minutes, while for cyclic stress, 20 to 25 stress cycles are required. Cyclic creep exhibits nearly the same overall behavior as constant stress creep, but is more extensive. In contrast, gasket relaxation is greatest at a higher initial stress level but is reasonably constant in terms of percent relaxation. Most of the gasket relaxation also occurs in the first 10 to 15 minutes after initial prestressing of the bolts. It is also of interest to note that lock-up of the flange and compression gauge ring significantly limits the gasket relaxation.

In a real life gasketed bolted flange connection, the gasket is subjected to neither pure creep or pure relaxation, even under steady-state operating conditions, since gasket creep causes the bolt load and deflection to change when there is no lock-up between the flange and the compression gauge ring. It is expected that, in many cases, the flange bolting was initially tightened and then retightened some minutes or hours later or possibly during or after the preservice hydrostatic pressure test. Retightening eliminates much of the effect of initial short term creep and relaxation. Thus to account for relaxation and cyclic creep for those cases where lock-up between the flange and compression gauge ring does not occur (all

150# flanges and 300 and 600# flanges with low bolt stress and high gasket loading stiffness), a joint relaxation of 25% was taken to be median-centered while the range from 0 to 25% was taken to represent a $+3.0\beta$ variation.

3.2.11 Pipe Bending Moments

Bending moments in the piping at the flange connection due to deadweight or thermal loads are carried by tension in the bolting. Based on normal practice, the piping supports are placed such that deadweight pipe stresses are relatively low and thus, the additional bolt stress is small. If the flange bolting is elastic, it was felt that the reduced gasket stress on the one side is balanced by an increased gasket stress on the other such that the joint mass leak rate will be about the same whether or not the bending moment is considered. On the other hand, if the flange bolting is inelastic, the bending moment could result in some increase in the calculated leak area but the increase would be limited by redistribution of the bending moment. Thus, pipe bending stress was not considered specifically in the evaluation of the flange joint. It is judged that the variabilities from other sources and conservatisms introduced into the approach cover the potential effect of pipe bending moments.

3.3 Flange Joint Behavior

The behavior of gasketed flange connections due to increasing pressure of the ISLOCA event is characterized in the manner described in the following steps.

1. The flange bolts are initially torqued and retightened to prestress levels satisfying the requirements of Reference 14 resulting in an initial gasket stress.
2. Over the course of normal operation, gaskets sustain cyclic creep and relaxation. If Step 1 produced lock-up between the flange and compression gauge ring, the relaxation reduces the gasket stress with a corresponding increase in the lock-up stress and negligible change in the bolt stress. If Step 1 did not produce lock-up, the creep and relaxation reduces the gasket stress with a corresponding reduction in the bolt stress.
3. At the initiation of the ISLOCA event, the increasing pressure must first overcome the lock-up load, if any, with no reduction of the gasket stress or increase in bolt stress.

4. Further increase in pressure to the Gross Leak Pressure, defined as the point at which the gasket stress and the pressure are equal, is shared by the gasket and bolts in accordance with Figure 3-1 resulting in a decrease in the gasket stress and an increase in the bolt stress.
5. Further increase in pressure above the Gross Leak Pressure results in a corresponding increase in the bolt stress accompanied by increases in bolt length up to the bolt failure strain in accordance with the bolt stress-strain diagram.

These steps were used consistently in the evaluation of the myriad of cases covering the sizes, pressure ratings, and ranges of the variables affecting leakage.

3.4 Calculation Of Leak Rate And Leak Area

The definition of the onset of gross leakage, or the Gross Leak Pressure, as the point at which the gasket stress is equal to the pressure being retained, is used quite generally throughout the gasket industry. This definition has come about, it appears, from gasket tests where some "O"-ring and flat face gaskets have suffered blowout. Although it is doubtful that spiral-wound gaskets are on the verge of catastrophic failure when the gasket stress is reduced to the point that it equals the pressure, the potential certainly exists. For pressures less than the Gross Leak Pressure, the mass leak rate is calculated from the results of the gasket leakage test with water reported in Reference 18. Leakage of this form is related to the presence of seams and crevasses in the flange/seal joint rather than any apparent leak area. In this test, 4"-600# rated gaskets were subjected to both standard and cyclic load pressure sequences. The results are presented in Figure 3-4 which is a plot of Gasket Stress versus the Tightness Parameter (T_P). T_P is defined as:

$$T_P = \frac{P}{P^*} \left[\frac{L_{RM}^*}{L_{RM}} \right]^\alpha \quad (3-1)$$

where P = Internal Fluid Gauge Pressure (psig)

P^* = Reference Atmospheric Pressure (14.7 psia)

L_{RM}^* = Reference Mass Leak Rate (1 mg/sec)

L_{RM} = Total Mass Leak Rate through the Gasket (mg/sec)

and α = Tightness Parameter Exponent (1.0 for water)

Thus, the total mass leak rate for the water case is computed as:

$$L_{RM} = \frac{P}{(14.7 \cdot T_P)} \quad (3-2)$$

Since the leak rate data correspond to the total mass leakage from the 4"-600# rated gasket and not, for example, the leakage per unit mean circumference, a correction must be made to the calculated mass leak rate to account for the various gasket sizes. Since the probability of leakage increases with gasket perimeter, it is reasonable to assume that leakage through a larger diameter gasket will increase in proportion to the gasket diameter. In addition, a correction factor must be introduced to account for variations in the gasket width. It should be noted that the calculation of gasket width and gasket area should not include the outer 1/8" which is ineffective in the sealing process. The leak rate at the Gross Leak Pressure is then determined as:

$$LR_{GLP} = \frac{(D_o + D_i)(P)}{348.6 W_G} \left[\frac{1}{e^{\{6.1872 - 6.2361(9.2103 - \ln SG_o) - .791311(\ln SG_o - \ln SG)\}}} \right] \quad (3-3)$$

where D_o = Gasket Outside Diameter (in)

D_i = Gasket Inside Diameter (in)

W_G = Gasket Width (in)

SG_o = Initial Gasket Stress (psi) = Actual Gasket Stress/(1 - JR/100)

SG = Current Gasket Stress (psi)

and JR = Joint Relaxation expressed in percent of SG_o

Note that in Equation 3-3 the quantity, 348.6, is the product of 14.7 from Equation 3-2 and 23.714 which is the value of $(D_o + D_i)/W_G$ for a 4" - 600# rated gasket. The quantity in the denominator of the term within the brackets in Equation 3-3 represents the Tightness Parameter (T_P) which is obtained by curve-fitting using the curves shown in Figure 3-4.

It is difficult to determine the significance of a leak rate in milligrams per second. However, if a drop of water is idealized as a 1/8 inch diameter solid sphere, a leak rate of 1 mg/sec would correspond to 3.5 drops per minute or about one drop every 17 seconds. By the time the leak rate increased to 17 mg/sec, the joint leakage would be about 1 drop per second which could be of concern for nuclear operation depending on the location of nearby electrical equipment. Leak rates of 200 to 500 mg/sec would constitute a spray of water which could possibly inhibit some operator actions in the vicinity.

For pressures above the Gross Leak Pressure, it was judged that the leakage is no longer due to seams and crevasses in the flange/seal joint but due to actual separation of the flange and gasket. Thus, a leak area is calculated which is intended to be in addition to the leak rate calculated at Gross Leak Pressure. The leak area is calculated as the mean gasket perimeter times the separation distance at the gasket. The separation distance is affected by bolt extension, gasket recovery, and flange flexibility. Of these, the contribution of bolt extension is by far the most dominant one. Therefore, the separation distance calculated in this study includes the effect of bolt extension only. Note that excluding the effect of gasket recovery from the leak area calculation is conservative and leads to slightly higher leak area values. The leak area at pressures above the Gross Leak Pressure is equated as shown in Equations 3-4 and 3-5, respectively, for the case where the bolt stress is less than or equal to the bolt material yield stress and for the case where the bolt stress exceeds the material yield. The term, $f(p_{GL}, JR, SG_o, SG_{pGL}, K_G)$, represents the remaining recoverable gasket deflection beyond GLP. It should be noted that some gasket recovery occurs prior to GLP. Due to the difficulties involved in arriving at a reasonably accurate estimate of the term, $f(p_{GL}, JR, SG_o, SG_{pGL}, K_G)$, and recognizing the fact that the effect of bolt extension on the leak area far exceeds that of the gasket recovery, this term was conservatively neglected.

For bolt stress $\leq$ yield

$$A_L = \frac{\pi(D_o + D_i)}{2} \left[L_B \left\{ \frac{(p - p_{GL})(A_p)}{N_B(A_B)(E_B)} \right\} - f(p_{GL}, JR, SG_o, SG_{pGL}, K_G) \right] \quad (3-4)$$

For bolt stress $>$ yield

$$A_L = \frac{\pi(D_o + D_i)}{2} \left[L_B \left\{ \frac{(S_{By} - S_{Bo})}{E_B} + \left[S_{Bo} + \frac{(p - p_{GL})(A_p)}{N_B(A_B)} - S_{By} \right] / E'_B \right\} - f(p_{GL}, JR, SG_o, SG_{pGL}, K_G) \right] \quad (3-5)$$

where L_B = Bolt/Stud Length (in)

P_{GL} = Gross Leak Pressure (psi)

A_P = Pressure Area (in²) - based on gasket inside diameter

N_B = Number of Flange Bolts

A_B = Bolt Tensile Stress Area (in²) - per bolt

E_B = Bolt Material Elastic Modulus (psi)

SG_{PGL} = Gasket Stress at Gross Leak Pressure (psi)

$f(P_{GL}, JR, SG_o, SG_{PGL}, K_G)$ = Recoverable Gasket Deflection (in)

K_G = Gasket Unloading/Reloading Stiffness (psi/in)

S_{By} = Bolt Material Yield Stress (psi)

S_{Ba} = Actual Bolt Stress (psi) = $(1 - JR/100)S_{Bo}$ for no lockup case
= S_{Bo} for lockup case

S_{Bo} = Initial Bolt Stress (psi)

and E'_B = Bolt Modulus for appropriate inelastic portion of the stress-strain diagram (psi)

3.5 Gasketed Flange Connection Capacities and Variabilities

Based on gasket loading stiffness of Table 3-1 (for median case, $T_g=0.130$) and gasket load to total load ratios of Figure 3-1, the gasketed flange connection capacities and variabilities of 150, 300, and 600lb rated flanges were evaluated and are discussed in the following sections. As mentioned in section 3.2.8, for the sake of savings in the computation effort, the gasket stiffnesses for the 150lb flanges were conservatively used for the 300 and 600lb rated flanges as well.

3.5.1 150lb Flanges

The results of the analyses for the 150lb rated flanges are presented in Tables 3-2 through 3-14. Table 3-2 shows the dimensional data for the gaskets and flange bolting which are standard for 150lb ANSI flanges while Tables 3-3 through 3-14 present the calculated leak

rates (pressure $\leq$ Gross Leak Pressure) and leak areas (pressure $>$ Gross Leak Pressure) for the various cases studied. The cases include variation in initial bolt/stud stress (Tables 3-3 through 3-9), variation in joint relaxation (Tables 3-5 and 3-10 through 3-13), and variation in bolt yield strength (Tables 3-5 and 3-14).

Each table includes the effective gasket stress, defined here as the gasket stress due to bolt preload unaffected by the presence of the compression gauge ring, the actual gasket stress, limited by the presence of the compression gauge ring, and the resulting gasket loading deflection. For the 150lb flanges, the effective and actual gasket stress values are equal indicating that the bolt preload is insufficient to bottom out the flange on the compression ring and thus the gasket deflection is less than the maximum deflection of the nominal 0.175" gasket. As noted above, the Gross Leak Pressure (GLP) is computed assuming that the pressure load results in a direct and corresponding reduction in the gasket load. As expected, the computed Gross Leak Pressure is greater for small flanges and trends lower as flange size increases. The mass leak rates and leak areas are calculated as described above. In Table 3-3, which presents the results for an initial bolt stress of 15,000 psi, it can be seen that the mass leak rate at the Gross Leak Pressure and even at low fractions of the GLP are high indicating that the preload is insufficient to properly seat the gasket. It can also be seen that the gasket stress is well below the 5000 psi "knee" of the stress-deflection and stress-leak rate curves for typical spiral wound gaskets (Figures 3-2 and 3-3). Thus, an initial bolt stress of 15,000 psi was not considered a realistic plant condition.

Comparing Tables 3-3 through 3-9, the Gross Leak Pressure increases and the mass leak rate decreases with increasing initial bolt stress. The leak area also decreases with increase in initial bolt stress due to the fact that the increased bolt stress does not exceed the material yield strength. It should be noted, however, that the leak areas are reported in terms of multiples of the Gross Leak Pressure and not constant pressures. Comparing Tables 3-10 through 3-13, it can be seen that Gross Leak Pressure decreases and leak rate increases slightly with increase in joint relaxation. It was also found that at a given pressure value above the Gross Leak Pressure the leak area increases with increasing joint relaxation. As expected, the variation in bolt yield stress (as long as the bolt material yield strength is not exceeded) does not affect the results as can be seen by comparing Tables 3-5 and 3-14.

The estimation of the leak rate and leak area variabilities was based upon the results of the 8"-150# flange cases. It was judged that calculated uncertainties would not vary substantially with flange size and therefore a constant uncertainty variability was used for all flanges. Mass leak rate is plotted versus pressure in Figure 3-5 for a selected number of the 8"-150#

flange cases investigated. It can be seen that the variation of leak rate due to variation in the initial bolt stress is greater than that due to variation in percent joint relaxation. The uncertainty variability for joint relaxation was computed judging the range from 25% to 0% relaxation to reasonably represent a -3β variation. Although there was a slight difference in the variability calculated for each pressure, an uncertainty variability of $\beta_{JR} = 0.16$ was selected to model the variation over the entire pressure range. Similarly, the uncertainty variability due to variation in initial bolt preload stress was computed judging the range from 25,000 to 30,000 psi to reasonably represent a $+2.33\beta$ variation. Accordingly, an uncertainty variability of $\beta_{BP} = 0.52$ was selected to model the variation over the entire pressure range. As noted above, there was no variation in leak rate due to variation in bolt yield stress and therefore, the combined mass leak rate uncertainty variability for 150lb rated flanges is:

$$\beta_{LR} = (0.16^2 + 0.52^2)^{1/2} = 0.54 \quad (3-6)$$

The variability in leak area is treated similarly, and a $\beta_{JR} = 0.06$ together with an uncertainty variability of 0.10 due to the effect of initial bolt stress were estimated over the entire pressure range. As noted above, there was no variation in leak area due to variation in bolt yield stress and therefore, the combined leak area uncertainty variability for 150lb rated flanges is:

$$\beta_{LA} = (0.06^2 + 0.10^2)^{1/2} = 0.12 \quad (3-7)$$

3.5.2 300lb Flanges

The results of the analyses for the 300lb rated flanges are presented in Tables 3-15 through 3-27. Table 3-15 shows the dimensional data for the gaskets and flange bolting which are standard for 300lb ANSI flanges while Tables 3-16 through 3-27 present the calculated leak rates (pressure $\leq$ Gross Leak Pressure) and leak areas (pressure $>$ Gross Leak Pressure) for the various cases studied. The cases include variation in initial bolt/stud stress (Tables 3-16 through 3-22), variation in joint relaxation (Tables 3-19 and 3-23 through 3-26), and variation in bolt yield strength (Tables 3-19 and 3-27).

The effective gasket stress exceeds the actual gasket stress for the 300lb rated flanges indicating that the bolt preload is sufficient to bottom out the flange on the compression ring. Thus, the gasket deflection is equal to the maximum deflection of the nominal 0.175" gasket. The Gross Leak Pressure (GLP) is again computed assuming that the pressure load results in a direct and corresponding reduction in the gasket load.

Comparing Tables 3-16 through 3-22, it can be seen that the 300lb rated flanges behave similar to the 150lb rated flanges with regard to the effect of the initial bolt preload stress; the Gross Leak Pressure increases and the mass leak rate decreases with increasing initial bolt stress. The leak area decreases with increase in initial bolt stress due to the fact that the increased bolt stress creates a greater lock-up force which must be overcome before the gasket begins to unload. Comparing Tables 3-23 through 3-26, it can be seen that joint relaxation has little effect on the flange leak resistance. This is again due to the lock-up between the flange and the compression gauge ring. As the gasket relaxes, the reduced gasket force results in a corresponding increase in the lock-up force with negligible change in the bolt force. The effect of variation in bolt yield stress can be seen by comparing Tables 3-19 and 3-27. Again since the bolt material yield stress is not exceeded, the variation of bolt yield stress has no effect on the results.

The mass leak rates for pressures less than the Gross Leak Pressure are relatively low for the 300lb rated flanges and the Gross Leak Pressures are relatively high for the median case (Table 3-24). Thus, the specific value for the uncertainty variability related to leak rate is of little importance. Therefore, the combined uncertainty variability of 0.54 computed for the 150lb rated flanges was also used to represent the combined variability for the 300lb rated flanges accounting for variation in initial bolt preload stress and bolt relaxation.

Similarly, the leak area values for the 300lb flanges are relatively low, and the specific value for the uncertainty variability related to leak area is of little importance. Therefore, the combined uncertainty variability of 0.12 computed for the 150lb rated flanges was also used to represent the combined variability for the 300lb rated flanges accounting for variation in initial bolt preload stress and bolt relaxation.

3.5.3 600lb Flanges

The results of the analyses for the 600lb rated flanges are presented in Tables 3-28 through 3-31. Table 3-28 shows the dimensional data for the gasket and flange bolting which are standard for 600lb ANSI flanges. Table 3-29 presents the calculated leak rates and leak areas for the median case. It can be seen from Table 3-29 that the Gross Leak Pressures for

all sizes of 600lb rated flanges are very high and greater than the range of interest. Thus, 600lb flanges are not expected to leak when subjected to pressures as high as the Reactor Coolant System operating pressure.

The results presented for the 44" tubesheet correspond to the tubesheet used for the residual heat exchanger RS-707 based on 39 1-1/8" bolts. The recommended torque of 295 ft-lbs for these bolts would produce an initial bolt stress value of about 20,000 psi for which the results are presented in Table 3-30. However, the Gross Leak Pressure of 334 psi at this bolt initial stress value is less than the hydrostatic test pressure of 803 psi. Table 3-31 indicates that an initial bolt stress of 60,000 psi which would be produced by a torque of 885 ft-lbs would result in the Gross Leak Pressure of 1002 psi. The specification recommended that the bolts be torqued upto a maximum torque value of 800 ft-lbs if needed, which results in a Gross Leak Pressure value higher than the hydrostatic test pressure of 803 psi.

TABLE 3-1
STIFFNESSES FOR ASBESTOS-FILLED SPIRAL WOUND GASKETS

Flange Diameter (in)	Pressure Rating (#)	*-----GASKET-----*				*-----BOLTS-----*				Gasket Test Load (lbs)	Bolt Stress at GTL (psi)	*-----GASKET STIFFNESS-----*		
		OD (in)	ID (in)	Width (in)	Area (sq in)	Number	Diameter (in)	Area (sq in)	Length (in)			T _g =0.125 (psi/in)	T _g =0.130 (psi/in)	T _g =0.135 (psi/in)
1-1/2	150	2.750	2.125	0.2500	1.865	4	1/2	0.1416	1.625	15100	26660	161902	179892	202378
	300	2.750	2.125	0.2500	1.865	4	3/4	0.3340	1.875	36200	27096	388137	431263	485171
	600	2.750	2.125	0.2500	1.865	4	3/4	0.3340	2.375	36200	27096	388137	431263	485171
2	150	3.375	2.750	0.2500	2.356	4	5/8	0.2256	1.750	24200	26817	205416	228240	256770
	300	3.375	2.750	0.2500	2.356	8	5/8	0.2256	2.000	48400	26817	410832	456480	513540
	600	3.375	2.750	0.2500	2.356	8	5/8	0.2256	2.625	48400	26817	410832	456480	513540
2-1/2	150	3.875	3.250	0.2500	2.749	4	5/8	0.2256	2.000	24200	26817	176071	195634	220089
	300	3.875	3.250	0.2500	2.749	8	3/4	0.3340	2.250	72500	27133	527485	586094	659356
	600	3.875	3.250	0.2500	2.749	8	3/4	0.3340	2.875	72500	27133	527485	586094	659356
3	150	4.750	4.000	0.3125	4.234	4	5/8	0.2256	2.125	24200	26817	114318	127021	142898
	300	4.750	4.000	0.3125	4.234	8	3/4	0.3340	2.500	72500	27133	342483	380537	428104
	600	4.750	4.000	0.3125	4.234	8	3/4	0.3340	3.125	72500	27133	342483	380537	428104
4	150	5.875	5.000	0.3750	6.332	8	5/8	0.2256	2.125	48500	26873	153184	170204	191479
	300	5.875	5.000	0.3750	6.332	8	3/4	0.3340	2.750	72500	27133	228986	254429	286232
	600	5.875	4.750	0.5000	8.247	8	7/8	0.4612	3.625	101000	27374	244947	272163	306184
6	150	8.250	7.188	0.4688	11.275	8	3/4	0.3340	2.250	72500	27133	128606	142895	160757
	300	8.250	7.188	0.4688	11.275	12	3/4	0.3340	3.125	100700	25125	178629	198477	223286
	600	8.250	6.875	0.6250	14.726	12	1	0.6051	4.375	190000	26166	258043	286715	322554
8	150	10.375	9.188	0.5313	16.220	8	3/4	0.3340	2.500	72500	27133	89394	99327	111743
	300	10.375	9.188	0.5313	16.220	12	7/8	0.4612	3.500	150800	27248	185940	206600	232425
	600	10.375	8.875	0.6875	20.654	12	1-1/8	0.7627	5.000	262000	28626	253710	281900	317137
10	150	12.500	11.313	0.5313	19.767	12	7/8	0.4612	2.625	151000	27284	152781	169756	190976
	300	12.500	11.313	0.5313	19.767	16	1	0.6051	4.000	264400	27310	267518	297242	334398
	600	12.500	10.813	0.7813	28.455	16	1-1/4	0.9684	5.625	446000	28785	313474	348304	391842
12	150	14.750	13.375	0.6250	27.489	12	7/8	0.4612	2.750	151000	27284	109862	122069	137328
	300	14.750	13.375	0.6250	27.489	16	1-1/8	0.7627	4.250	349400	28632	254211	282457	317764
	600	14.750	12.875	0.8750	37.797	20	1-1/4	0.9684	5.875	557000	28759	294730	327478	368413
14	150	16.000	14.625	0.6250	29.943	12	1	0.6051	3.000	198000	27268	132250	146944	165312
	300	16.000	14.625	0.6250	29.943	20	1-1/8	0.7627	4.500	436000	28583	291217	323574	364021
	600	16.000	14.250	0.8125	38.448	20	1-3/8	1.1538	6.125	---	27408	328998	365553	411247
16	150	18.250	16.625	0.7500	40.939	16	1	0.6051	3.125	264000	27268	128973	143303	161216
	300	18.250	16.625	0.7500	40.939	20	1-1/4	0.9684	4.750	557000	28759	272113	302348	340141
	600	18.250	16.250	0.9375	50.621	20	1-1/2	1.4041	6.625	---	27643	306701	340778	383376
18	150	20.750	18.688	0.9688	59.822	16	1-1/8	0.7627	3.375	---	27262	111226	123584	139032
	300	20.750	18.688	0.9688	59.822	24	1-1/4	0.9684	5.000	---	27565	214187	237985	267733
	600	20.750	18.500	1.0625	65.298	20	1-5/8	1.7723	7.125	---	28439	308754	343060	385943
20	150	22.750	20.688	0.9688	65.909	20	1-1/8	0.7627	3.625	---	27262	126192	140213	157740
	300	22.750	20.688	0.9688	65.909	24	1-1/4	0.9684	5.250	---	27565	194406	216007	243008
	600	22.750	20.500	1.0625	71.974	24	1-5/8	1.7723	7.625	---	28439	336140	373488	420175
24	150	27.000	24.750	1.0625	86.161	20	1-1/4	0.9684	4.000	---	27565	123927	137696	154908
	300	27.000	24.750	1.0625	86.161	24	1-1/2	1.4041	5.750	---	27643	216232	240258	270290
	600	27.000	24.750	1.0625	86.161	24	1-7/8	2.4107	8.625	---	28667	385000	427778	481250
44	300	47.625	45.625	0.9375	137.138	40	1-3/8	1.1538	10.215	---	27408	307457	368948	461185

Reference: MIL-G-21032E, Amendment 2

Calculation GASSTIF

GTL = Gasket Test Load

T_g = Compressed Gasket Thickness

TABLE 3-2
150# ANSI FLANGE AND GASKET DATA

Flange Diameter (in)	*-----GASKET-----*					*-----BOLTS-----*			
	OD (in)	ID (in)	Width (in)	Area (sq in)	Pressure Area (sq in)	Number	Diameter (in)	Area (sq in)	Length (in)
Flanges									
1-1/2	2.750	2.125	0.3125	2.393	3.547	4	1/2	0.1416	1.625
2	3.375	2.750	0.3125	3.007	5.940	4	5/8	0.2256	1.750
2-1/2	3.875	3.250	0.3125	3.497	8.296	4	5/8	0.2256	2.000
3	4.750	4.000	0.3750	5.154	12.566	4	5/8	0.2256	2.125
4	5.875	5.000	0.4375	7.474	19.635	8	5/8	0.2256	2.125
6	8.250	7.188	0.5313	12.882	40.574	8	3/4	0.3340	2.250
8	10.375	9.188	0.5938	18.245	66.296	8	3/4	0.3340	2.500
10	12.500	11.313	0.5938	22.209	100.509	12	7/8	0.4612	2.625
12	14.750	13.375	0.6875	30.373	140.500	12	7/8	0.4612	2.750
14	16.000	14.625	0.6875	33.073	167.989	12	1	0.6051	3.000
16	18.250	16.625	0.8125	44.510	217.077	16	1	0.6051	3.125
18	20.750	18.688	1.0313	63.884	274.279	16	1-1/8	0.7627	3.375
20	22.750	20.688	1.0313	70.364	336.129	20	1-1/8	0.7627	3.625
24	27.000	24.750	1.1250	91.450	481.105	20	1-1/4	0.9684	4.000
Tubesheet									
6	8.250	7.250	0.5000	12.174	41.282	8	3/4	0.3340	3.500

Bolt areas correspond to tensile stress area and not thread root area which is the basis for the bolt torque values

TABLE 3-3
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 15000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	4555	4555	0 025	2455	62	10	23	40	0 00	0 00	0 00	0 01	38823
2	5745	5745	0 025	2570	22	3	8	13	0 00	0 00	0 01	0 01	40544
2-1/2	4924	4924	0 025	1963	57	8	18	33	0 00	0 01	0 01	0 01	42064
3	3197	3197	0 025	1243	531	72	168	306	0 00	0 01	0 01	0 02	40435
4	4275	4275	0 025	1511	118	15	35	66	0 01	0 01	0 02	0 02	38178
6	3555	3555	0 025	1010	336	36	88	171	0 01	0 01	0 02	0 03	34941
8	2471	2471	0 025	568	2412	223	553	1118	0 01	0 02	0 03	0 04	31625
10	4200	4200	0 025	753	173	13	34	71	0 01	0 02	0 03	0 05	30046
12	3020	3020	0 025	494	960	69	176	375	0 01	0 03	0 04	0 05	27545
14	3637	3637	0 025	550	388	26	67	146	0 02	0 03	0 05	0 06	27731
16	3547	3547	0 025	563	424	30	76	163	0 02	0 04	0 06	0 07	27620
18	3060	3060	0 025	548	826	64	161	339	0 02	0 04	0 07	0 09	27314
20	3472	3472	0 025	569	461	33	85	181	0 03	0 05	0 08	0 11	27541
24	3372	3372	0 024	512	575	39	100	216	0 04	0 07	0 11	0 14	27722
 Tubesheet													
6	3793	3793	0 027	1016	53	26	64	126	0 01	0 02	0 03	0 05	35399

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1501500

TABLE 3-4
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 20000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	6073	6073	0 034	3273	14	2	5	9	0 00	0 00	0 01	0 01	51764
2	7660	7660	0 034	3427	5	1	2	3	0 00	0 01	0 01	0 01	54059
2-1/2	6566	6566	0 033	2617	13	2	4	7	0 00	0 01	0 01	0 02	56085
3	4263	4263	0 034	1657	118	16	37	68	0 01	0 01	0 02	0 02	53913
4	5700	5700	0 034	2015	26	3	8	15	0 01	0 01	0 02	0 03	50904
6	4740	4740	0 033	1347	74	8	20	38	0 01	0 02	0 03	0 04	46588
8	3295	3295	0 033	757	535	49	123	248	0 01	0 02	0 04	0 05	42167
10	5600	5600	0 033	1004	38	3	7	16	0 02	0 03	0 05	0 06	40061
12	4027	4027	0 033	659	213	15	39	83	0 02	0 04	0 05	0 07	36727
14	4850	4850	0 033	734	86	6	15	32	0 02	0 04	0 06	0 08	36974
16	4730	4730	0 033	750	94	7	17	36	0 02	0 05	0 07	0 10	36827
18	4080	4080	0 033	731	183	14	36	75	0 03	0 06	0 09	0 12	36419
20	4629	4629	0 033	759	102	7	19	40	0 04	0 07	0 11	0 14	36721
24	4496	4496	0 033	683	127	9	22	48	0 05	0 10	0 14	0 19	36962
Tubesheet													
6	5058	5058	0 035	1354	12	6	14	28	0 02	0 03	0 05	0 06	47198

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1502000

TABLE 3-5
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 25000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	7591	7591	0 042	4091	4	1	2	3	0 00	0 01	0 01	0 01	64705
2	9575	9575	0 042	4284	2	0	1	1	0 00	0 01	0 01	0 02	67574
2-1/2	8207	8207	0 042	3271	4	1	1	2	0 01	0 01	0 02	0 02	70107
3	5329	5329	0 042	2071	37	5	12	21	0 01	0 01	0 02	0 03	67392
4	7125	7125	0 042	2518	8	1	2	5	0 01	0 02	0 03	0 03	63630
6	5925	5925	0 041	1684	23	2	6	12	0 01	0 02	0 04	0 05	58235
8	4118	4118	0 041	946	166	15	38	77	0 02	0 03	0 05	0 06	52708
10	7000	7000	0 041	1255	12	1	2	5	0 02	0 04	0 06	0 08	50076
12	5033	5033	0 041	824	66	5	12	26	0 02	0 04	0 07	0 09	45909
14	6062	6062	0 041	917	27	2	5	10	0 03	0 05	0 08	0 11	46218
16	5912	5912	0 041	938	29	2	5	11	0 03	0 06	0 09	0 12	46033
18	5100	5100	0 041	913	57	4	11	23	0 04	0 07	0 11	0 15	45524
20	5786	5786	0 041	949	32	2	6	12	0 04	0 09	0 13	0 18	45902
24	5620	5620	0 041	854	40	3	7	15	0 06	0 12	0 18	0 24	46203
Tubesheet													
6	6322	6322	0 044	1693	4	2	4	9	0 02	0 04	0 06	0 08	58998

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1502500

TABLE 3-6
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 30000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	9109	9000	0 050	4877	2	0	1	1	0 00	0 01	0 01	0 01	77133
2	11490	11400	0 050	5120	1	0	0	0	0 00	0 01	0 01	0 02	80766
2-1/2	9848	9800	0 050	3916	2	0	0	1	0 01	0 01	0 02	0 03	83922
3	6394	6350	0 050	2477	15	2	5	8	0 01	0 02	0 03	0 03	80607
4	8550	8500	0 050	3015	3	0	1	2	0 01	0 02	0 03	0 04	76171
6	7110	7110	0 050	2020	9	1	2	5	0 01	0 03	0 04	0 06	69882
8	4942	4942	0 050	1136	64	6	15	30	0 02	0 04	0 06	0 07	63250
10	8400	8400	0 049	1506	5	0	1	2	0 02	0 05	0 07	0 09	60091
12	6040	6040	0 050	988	25	2	5	10	0 03	0 05	0 08	0 11	55091
14	7275	7275	0 049	1101	10	1	2	4	0 03	0 06	0 10	0 13	55462
16	7095	7095	0 050	1126	11	1	2	4	0 04	0 07	0 11	0 15	55240
18	6120	6120	0 049	1096	22	2	4	9	0 04	0 09	0 13	0 18	54628
20	6943	6943	0 050	1138	12	1	2	5	0 05	0 11	0 16	0 21	55082
24	6744	6744	0 049	1024	15	1	3	6	0 07	0 14	0 21	0 29	55443
Tubesheet													
6	7587	7150	0 050	1996	2	1	2	5	0 02	0 05	0 07	0 09	69575

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1503000

TABLE 3-7
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect. (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at .25GLP (mg/sec)	Leak Rate at .50GLP (mg/sec)	Leak Rate at .75GLP (mg/sec)	Leak Area at 1.25GLP (sq in)	Leak Area at 1.5GLP (sq in)	Leak Area at 1.75GLP (sq in)	Leak Area at 2.0GLP (sq in)	Bolt Stress at 2.0GLP (psi)
Flanges													
1-1/2	10628	9000	0.050	5245	2	0	1	1	0.00	0.01	0.01	0.01	82956
2	13405	11400	0.050	5540	1	0	0	0	0.01	0.01	0.02	0.02	87388
2-1/2	11490	9800	0.050	4243	2	0	1	1	0.01	0.02	0.02	0.03	90932
3	7460	6350	0.050	2696	15	2	5	9	0.01	0.02	0.03	0.04	87750
4	9976	8500	0.050	3312	3	0	1	2	0.01	0.02	0.03	0.05	83680
6	8295	7150	0.050	2259	9	1	2	5	0.02	0.03	0.05	0.06	78154
8	5766	4965	0.050	1292	64	6	16	31	0.02	0.04	0.06	0.08	71947
10	9799	8500	0.050	1734	4	0	1	2	0.03	0.05	0.08	0.11	69177
12	7047	6100	0.050	1153	25	2	5	11	0.03	0.06	0.09	0.12	64273
14	8487	7350	0.050	1284	10	1	2	4	0.04	0.07	0.11	0.15	64705
16	8277	7150	0.050	1313	11	1	2	5	0.04	0.09	0.13	0.17	64447
18	7140	6200	0.050	1278	21	2	5	9	0.05	0.10	0.16	0.21	63733
20	8100	7000	0.050	1328	12	1	2	5	0.06	0.12	0.19	0.25	64262
24	7868	6900	0.050	1195	14	1	3	6	0.08	0.17	0.25	0.33	64684
Tubesheet													
6	8851	7150	0.050	2233	2	1	3	5	0.03	0.05	0.08	0.10	77835

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0.050 in

Calculation 1503500

TABLE 3-8
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 40000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	12146	9000	0 050	5613	2	0	1	1	0 00	0 01	0 01	0 02	88780
2	15320	11400	0 050	5959	1	0	0	0	0 01	0 01	0 02	0 02	94010
2-1/2	13131	9800	0 050	4570	2	0	1	1	0 01	0 02	0 02	0 03	97943
3	8526	6350	0 050	2916	15	2	5	9	0 01	0 02	0 03	0 04	94894
4	11401	8500	0 050	3509	3	0	1	2	0 01	0 02	0 04	0 05	91188
6	9480	7150	0 050	2495	9	1	3	5	0 02	0 04	0 05	0 07	86307
8	6589	4965	0 050	1447	66	7	17	34	0 02	0 05	0 07	0 10	80591
10	11199	8500	0 050	1960	5	0	1	2	0 03	0 06	0 09	0 12	78190
12	8053	6100	0 050	1318	26	2	6	12	0 04	0 07	0 11	0 14	73455
14	9700	7350	0 050	1467	10	1	2	4	0 04	0 08	0 13	0 17	73949
16	9460	7150	0 050	1501	11	1	2	5	0 05	0 10	0 15	0 20	73653
18	8160	6200	0 050	1461	22	2	5	10	0 06	0 12	0 18	0 24	72838
20	9258	7000	0 050	1518	12	1	3	6	0 07	0 14	0 21	0 29	73443
24	8992	6900	0 050	1366	14	1	3	6	0 10	0 19	0 29	0 38	73924
Tubesheet													
6	10115	7150	0 050	2470	2	1	3	5	0 03	0 06	0 08	0 11	86095

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1504000

TABLE 3-9
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 50000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	15182	9000	0 050	6349	2	0	1	1	0 00	0 01	0 01	0 02	100427
2	19150	11400	0 050	6799	1	0	0	0	0 01	0 01	0 02	0 03	107255
2-1/2	16414	9800	0 050	5224	2	0	1	1	0 01	0 02	0 03	0 04	111964
3	10657	6350	0 050	3355	16	3	6	10	0 01	0 02	0 04	0 05	109181
4	14251	8500	0 050	4203	3	1	1	2	0 01	0 03	0 04	0 06	106205
6	11849	7150	0 050	2967	9	1	3	6	0 02	0 04	0 06	0 08	102613
8	8237	4965	0 050	1757	68	9	20	38	0 03	0 06	0 09	0 12	97880
10	13999	8500	0 050	2412	5	1	1	2	0 04	0 07	0 11	0 15	96218
12	10067	6100	0 050	1647	27	3	7	13	0 04	0 09	0 13	0 18	91818
14	12125	7350	0 050	1834	11	1	3	5	0 05	0 11	0 16	0 21	92436
16	11824	7150	0 050	1876	12	1	3	6	0 06	0 12	0 19	0 25	92067
18	10200	6200	0 050	1826	23	3	6	12	0 07	0 15	0 22	0 30	91047
20	11572	7000	0 050	1897	13	1	3	7	0 09	0 18	0 27	0 36	91803
24	11239	6900	0 050	1707	15	1	4	7	0 12	0 24	0 36	0 48	92406
Tubesheet													
6	12644	7150	0 050	2944	2	1	3	6	0 03	0 07	0 10	0 13	102614

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1505000

TABLE 3-10
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 25000 psi

JOINT RELAXATION = 15%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	6453	6453	0 041	3477	4	1	2	3	0 00	0 00	0 01	0 01	54999
2	8139	8139	0 041	3641	1	0	1	1	0 00	0 01	0 01	0 01	57438
2-1/2	6976	6976	0 041	2780	4	1	1	2	0 00	0 01	0 01	0 02	59591
3	4529	4529	0 041	1760	35	5	11	20	0 01	0 01	0 02	0 02	57283
4	6057	6057	0 041	2141	8	1	2	4	0 01	0 01	0 02	0 03	54086
6	5036	5036	0 041	1431	22	2	6	11	0 01	0 02	0 03	0 04	49500
8	3501	3501	0 041	804	161	15	37	74	0 01	0 03	0 04	0 05	44802
10	5950	5950	0 040	1067	12	1	2	5	0 02	0 03	0 05	0 07	42565
12	4278	4278	0 041	700	64	5	12	25	0 02	0 04	0 06	0 07	39023
14	5153	5153	0 040	780	26	2	4	10	0 02	0 04	0 07	0 09	39285
16	5025	5025	0 040	797	28	2	5	11	0 03	0 05	0 08	0 11	39128
18	4335	4335	0 040	776	55	4	11	23	0 03	0 06	0 09	0 13	38695
20	4918	4918	0 040	806	31	2	6	12	0 04	0 08	0 11	0 15	39016
24	4777	4777	0 040	726	38	3	7	14	0 05	0 10	0 15	0 20	39272
Tubesheet													
6	5374	5374	0 043	1439	17	2	4	8	0 02	0 03	0 05	0 07	50148

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1502515

TABLE 3-11
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 25000 psi

JOINT RELAXATION = 25%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect. (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at .25GLP (mg/sec)	Leak Rate at .50GLP (mg/sec)	Leak Rate at .75GLP (mg/sec)	Leak Area at 1.25GLP (sq in)	Leak Area at 1.5GLP (sq in)	Leak Area at 1.75GLP (sq in)	Leak Area at 2.0GLP (sq in)	Bolt Stress at 2.0GLP (psi)
Flanges													
1-1/2	5693	5693	0.040	3068	4	1	2	3	0.00	0.00	0.01	0.01	48529
2	7181	7181	0.040	3213	1	0	0	1	0.00	0.01	0.01	0.01	50680
2-1/2	6155	6155	0.040	2453	4	1	1	2	0.00	0.01	0.01	0.02	52580
3	3996	3996	0.041	1553	34	5	11	20	0.01	0.01	0.02	0.02	50544
4	5344	5344	0.040	1889	8	1	2	4	0.01	0.01	0.02	0.03	47723
6	4444	4444	0.040	1263	22	2	6	11	0.01	0.02	0.03	0.04	43676
8	3089	3089	0.040	710	157	14	36	73	0.01	0.02	0.03	0.05	39531
10	5250	5250	0.039	941	11	1	2	5	0.01	0.03	0.04	0.06	37557
12	3775	3775	0.040	618	62	4	11	24	0.02	0.03	0.05	0.07	34432
14	4547	4547	0.040	688	25	2	4	9	0.02	0.04	0.06	0.08	34663
16	4434	4434	0.040	704	28	2	5	11	0.02	0.05	0.07	0.09	34525
18	3825	3825	0.040	685	54	4	10	22	0.03	0.06	0.08	0.11	34143
20	4340	4340	0.040	711	30	2	5	12	0.03	0.07	0.10	0.13	34426
24	4215	4215	0.039	640	37	3	6	14	0.04	0.09	0.13	0.18	34652
Tubesheet													
6	4742	4742	0.043	1270	16	2	4	8	0.01	0.03	0.04	0.06	44248

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0.050 in

Calculation 1502525

TABLE 3-12
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 25000 psi

JOINT RELAXATION = 33%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	5086	5086	0 040	2741	4	1	1	3	0 00	0 00	0 01	0 01	43352
2	6415	6415	0 039	2870	1	0	0	1	0 00	0 01	0 01	0 01	45274
2-1/2	5499	5499	0 039	2192	4	0	1	2	0 00	0 01	0 01	0 02	46972
3	3570	3570	0 040	1387	34	5	11	19	0 00	0 01	0 01	0 02	45152
4	4774	4774	0 040	1687	7	1	2	4	0 01	0 01	0 02	0 02	42632
6	3970	3970	0 039	1128	21	2	6	11	0 01	0 02	0 02	0 03	39017
8	2759	2759	0 040	634	153	14	35	71	0 01	0 02	0 03	0 04	35314
10	4690	4690	0 039	841	11	1	2	5	0 01	0 03	0 04	0 05	33551
12	3372	3372	0 040	552	61	4	11	24	0 01	0 03	0 04	0 06	30759
14	4062	4062	0 039	614	25	2	4	9	0 02	0 04	0 05	0 07	30966
16	3961	3961	0 039	629	27	2	5	10	0 02	0 04	0 06	0 08	30842
18	3417	3417	0 039	612	52	4	10	21	0 02	0 05	0 07	0 10	30501
20	3877	3877	0 039	636	29	2	5	11	0 03	0 06	0 09	0 12	30754
24	3765	3765	0 039	572	36	2	6	14	0 04	0 08	0 12	0 16	30956
Tubesheet													
6	4236	4236	0 042	1134	16	2	4	8	0 01	0 03	0 04	0 05	39528

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1502533

TABLE 3-13
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 25000 psi

JOINT RELAXATION = 50%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	3796	3796	0 038	2045	4	1	1	2	0 00	0 00	0 00	0 01	32352
2	4787	4787	0 037	2142	1	0	0	1	0 00	0 00	0 01	0 01	33787
2-1/2	4103	4103	0 038	1636	3	0	1	2	0 00	0 01	0 01	0 01	35053
3	2664	2664	0 039	1035	32	4	10	18	0 00	0 01	0 01	0 01	33696
4	3563	3563	0 038	1259	7	1	2	4	0 00	0 01	0 01	0 02	31815
6	2962	2962	0 038	842	20	2	5	10	0 01	0 01	0 02	0 02	29118
8	2059	2059	0 039	473	144	13	33	67	0 01	0 02	0 02	0 03	26354
10	3500	3500	0 038	628	10	1	2	4	0 01	0 02	0 03	0 04	25038
12	2517	2517	0 039	412	57	4	10	22	0 01	0 02	0 03	0 04	22955
14	3031	3031	0 038	459	23	2	4	9	0 01	0 03	0 04	0 05	23109
16	2956	2956	0 038	469	25	2	5	10	0 02	0 03	0 05	0 06	23017
18	2550	2550	0 039	457	49	4	10	20	0 02	0 04	0 06	0 07	22762
20	2893	2893	0 038	474	28	2	5	11	0 02	0 04	0 07	0 09	22951
24	2810	2810	0 038	427	34	2	6	13	0 03	0 06	0 09	0 12	23101
Tubesheet													
6	3161	3161	0 041	846	15	2	4	8	0 01	0 02	0 03	0 04	29499

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1502550

TABLE 3-14
150# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 25000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	7591	7591	0 042	4091	4	1	2	3	0 00	0 01	0 01	0 01	64705
2	9575	9575	0 042	4284	2	0	1	1	0 00	0 01	0 01	0 02	67574
2-1/2	8207	8207	0 042	3271	4	1	1	2	0 01	0 01	0 02	0 02	70107
3	5329	5329	0 042	2071	37	5	12	21	0 01	0 01	0 02	0 03	67392
4	7125	7125	0 042	2518	8	1	2	5	0 01	0 02	0 03	0 03	63630
6	5925	5925	0 041	1684	23	2	6	12	0 01	0 02	0 04	0 05	58235
8	4118	4118	0 041	946	166	15	38	77	0 02	0 03	0 05	0 06	52708
10	7000	7000	0 041	1255	12	1	2	5	0 02	0 04	0 06	0 08	50076
12	5033	5033	0 041	824	66	5	12	26	0 02	0 04	0 07	0 09	45909
14	6062	6062	0 041	917	27	2	5	10	0 03	0 05	0 08	0 11	46218
16	5912	5912	0 041	938	29	2	5	11	0 03	0 06	0 09	0 12	46033
18	5100	5100	0 041	913	57	4	11	23	0 04	0 07	0 11	0 15	45524
20	5786	5786	0 041	949	32	2	6	12	0 04	0 09	0 13	0 18	45902
24	5620	5620	0 041	854	40	3	7	15	0 06	0 12	0 18	0 24	46203
Tubesheet													
6	6322	6322	0 044	1693	4	2	4	9	0 02	0 04	0 06	0 08	58998

Bolt Yield Stress = 101,500 psi (SA193-B7)

DGmax = 0 050 in

Calculation 1502500Y

TABLE 3-15
300# ANSI FLANGE AND GASKET DATA

Flange Diameter (in)	*-----GASKET-----*				Pressure Area (sq in)	*-----BOLTS-----*			
	OD (in)	ID (in)	Width (in)	Area (sq in)		Number	Diameter (in)	Area (sq in)	Length (in)
Flanges									
1-1/2	2.750	2.125	0.3125	2.393	3.547	4	3/4	0.3340	1.875
2	3.375	2.750	0.3125	3.007	5.940	8	5/8	0.2256	2.000
2-1/2	3.875	3.250	0.3125	3.497	8.296	8	3/4	0.3340	2.250
3	4.750	4.000	0.3750	5.154	12.566	8	3/4	0.3340	2.500
4	5.875	5.000	0.4375	7.474	19.635	8	3/4	0.3340	2.750
6	8.250	7.188	0.5313	12.882	40.574	12	3/4	0.3340	3.125
8	10.375	9.188	0.5938	18.245	66.296	12	7/8	0.4612	3.500
10	12.500	11.313	0.5938	22.209	100.509	16	1	0.6051	4.000
12	14.750	13.375	0.6875	30.373	140.500	16	1-1/8	0.7896	4.250
14	16.000	14.625	0.6875	33.073	167.989	20	1-1/8	0.7896	4.500
16	18.250	16.625	0.8125	44.510	217.077	20	1-1/4	0.9985	4.750
18	20.750	18.688	1.0313	63.884	274.279	24	1-1/4	0.9985	5.000
20	22.750	20.688	1.0313	70.364	336.129	24	1-1/4	0.9985	5.250
24	27.000	24.750	1.1250	91.450	481.105	24	1-1/2	1.4899	5.750

Bolt areas correspond to tensile stress area and not thread root area which is the basis for the bolt torque values

TABLE 3-16
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 20000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	14325	9000	0 050	6797	2	0	1	1	0 00	0 00	0 01	0 01	45574
2	15320	11400	0 050	7004	1	0	0	0	0 00	0 01	0 01	0 02	55246
2-1/2	19441	9800	0 050	6455	2	0	1	1	0 00	0 01	0 01	0 02	46721
3	12622	6350	0 050	4132	16	3	6	9	0 01	0 01	0 02	0 02	45418
4	8439	8439	0 050	3617	3	0	1	1	0 01	0 02	0 03	0 04	61728
6	7110	7110	0 050	2386	9	1	2	3	0 02	0 03	0 05	0 06	55029
8	6824	4965	0 050	1661	68	7	14	21	0 02	0 04	0 06	0 07	44669
10	9796	8500	0 050	1950	5	0	1	1	0 03	0 05	0 08	0 10	44465
12	9192	6100	0 050	1614	27	2	5	7	0 03	0 06	0 09	0 12	39406
14	10548	7350	0 050	1596	11	1	2	2	0 03	0 06	0 10	0 13	36974
16	9756	7150	0 050	1548	12	1	2	3	0 04	0 08	0 11	0 15	36827
18	8012	6200	0 050	1435	22	2	3	5	0 04	0 09	0 13	0 18	36419
20	7272	7000	0 050	1192	12	1	1	2	0 05	0 10	0 15	0 21	36721
24	8300	6900	0 050	1261	14	1	2	3	0 07	0 14	0 21	0 27	36962

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3002000

TABLE 3-17
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 25000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect. (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at .25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1.25GLP (sq in)	Leak Area at 1.5GLP (sq in)	Leak Area at 1.75GLP (sq in)	Leak Area at 2.0GLP (sq in)	Bolt Stress at 2.0GLP (psi)
Flanges													
1-1/2	17906	9000	0.050	7466	2	0	1	1	0.00	0.00	0.01	0.01	50063
2	19150	11400	0.050	7683	1	0	0	0	0.00	0.01	0.01	0.02	60596
2-1/2	24301	9800	0.050	7270	2	0	1	1	0.00	0.01	0.01	0.02	52624
3	15778	6350	0.050	4689	17	3	7	10	0.01	0.01	0.02	0.03	51533
4	10549	8500	0.050	4020	3	0	1	1	0.01	0.02	0.04	0.05	68616
6	8887	7150	0.050	2721	9	1	2	3	0.02	0.04	0.05	0.07	62736
8	8530	4965	0.050	1967	70	8	17	25	0.02	0.04	0.07	0.09	52890
10	12245	8500	0.050	2332	5	0	1	1	0.03	0.06	0.09	0.12	53191
12	11490	6100	0.050	1981	28	3	6	9	0.04	0.07	0.11	0.14	48368
14	13185	7350	0.050	1995	11	1	2	3	0.04	0.08	0.12	0.16	46218
16	12195	7150	0.050	1935	12	1	2	3	0.05	0.09	0.14	0.19	46033
18	10015	6200	0.050	1793	23	2	4	6	0.05	0.11	0.16	0.22	45524
20	9090	7000	0.050	1490	12	1	2	3	0.06	0.13	0.19	0.26	45902
24	10375	6900	0.050	1576	15	1	2	3	0.09	0.17	0.26	0.34	46203

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0.050 in

Calculation 3002500

TABLE 3-18
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 30000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	21487	9000	0 050	8135	2	0	1	1	0 00	0 01	0 01	0 01	54552
2	22979	11400	0 050	8361	1	0	0	0	0 00	0 01	0 01	0 02	65947
2-1/2	29161	9800	0 050	8086	2	0	1	1	0 01	0 01	0 02	0 02	58527
3	18933	6350	0 050	5245	17	4	7	11	0 01	0 01	0 02	0 03	57649
4	12659	8500	0 050	4409	3	1	1	2	0 01	0 03	0 04	0 05	75252
6	10665	7150	0 050	3049	9	1	2	4	0 02	0 04	0 06	0 08	70303
8	10236	4965	0 050	2273	72	10	19	29	0 03	0 05	0 08	0 10	61112
10	14694	8500	0 050	2715	5	0	1	1	0 04	0 07	0 11	0 15	61916
12	13788	6100	0 050	2348	29	3	7	10	0 04	0 08	0 13	0 17	57331
14	15822	7350	0 050	2394	11	1	2	4	0 05	0 10	0 14	0 19	55462
16	14634	7150	0 050	2322	13	1	3	4	0 06	0 11	0 17	0 23	55240
18	12018	6200	0 050	2152	24	3	5	8	0 07	0 13	0 20	0 26	54628
20	10908	7000	0 050	1788	13	1	2	3	0 08	0 15	0 23	0 31	55082
24	12450	6900	0 050	1891	15	1	3	4	0 10	0 21	0 31	0 41	55443

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3003000

TABLE 3-19

300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	25068	9000	0 050	8805	2	0	1	1	0 00	0 01	0 01	0 01	59041
2	26809	11400	0 050	9039	1	0	0	0	0 00	0 01	0 01	0 02	71297
2-1/2	34021	9800	0 050	8902	2	0	1	1	0 01	0 01	0 02	0 02	64431
3	22089	6350	0 050	5802	17	4	8	12	0 01	0 02	0 02	0 03	63765
4	14769	8500	0 050	4798	4	1	1	2	0 01	0 03	0 04	0 06	81888
6	12442	7150	0 050	3377	10	1	3	4	0 02	0 04	0 07	0 09	77869
8	11942	4965	0 050	2579	74	11	22	33	0 03	0 06	0 09	0 11	69333
10	17143	8500	0 050	3098	5	1	1	2	0 04	0 08	0 12	0 17	70642
12	16086	6100	0 050	2715	30	4	8	12	0 05	0 10	0 15	0 20	66294
14	18459	7350	0 050	2792	12	1	3	4	0 06	0 11	0 17	0 22	64705
16	17073	7150	0 050	2709	13	2	3	5	0 07	0 13	0 20	0 26	64447
18	14021	6200	0 050	2510	24	3	6	9	0 08	0 15	0 23	0 31	63733
20	12726	7000	0 050	2086	13	1	3	4	0 09	0 18	0 27	0 36	64262
24	14525	6900	0 050	2206	16	2	3	5	0 12	0 24	0 36	0 48	64684

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3003500

TABLE 3-20
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 40000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	28649	9000	0 050	9474	2	1	1	2	0 00	0 01	0 01	0 01	63529
2	30639	11400	0 050	9718	1	0	0	0	0 01	0 01	0 02	0 02	76648
2-1/2	38881	9800	0 050	9717	2	0	1	1	0 01	0 01	0 02	0 03	70334
3	25245	6350	0 050	6358	18	4	9	13	0 01	0 02	0 03	0 04	69881
4	16879	8500	0 050	5187	4	1	1	2	0 02	0 03	0 05	0 06	88523
6	14219	7150	0 050	3705	10	1	3	4	0 02	0 05	0 07	0 10	85436
8	13648	4965	0 050	2884	76	12	25	37	0 03	0 06	0 10	0 13	77554
10	19592	8500	0 050	3480	5	1	1	2	0 05	0 09	0 14	0 19	79367
12	18384	6100	0 050	3082	31	4	9	13	0 06	0 11	0 17	0 22	75257
14	21096	7350	0 050	3191	12	2	3	5	0 06	0 13	0 19	0 25	73949
16	19512	7150	0 050	3096	13	2	3	5	0 08	0 15	0 23	0 30	73653
18	16023	6200	0 050	2869	25	3	7	10	0 09	0 18	0 26	0 35	72838
20	14544	7000	0 050	2384	14	1	3	4	0 10	0 21	0 31	0 41	73443
24	16600	6900	0 050	2521	16	2	4	5	0 14	0 27	0 41	0 55	73924

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3004000

TABLE 3-21
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 50000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	35812	9000	0 050	10813	2	1	1	2	0 00	0 01	0 01	0 01	72507
2	38299	11400	0 050	11074	1	0	0	1	0 01	0 01	0 02	0 02	87349
2-1/2	48601	9800	0 050	11348	2	1	1	2	0 01	0 02	0 02	0 03	82141
3	31556	6350	0 050	7471	18	5	10	16	0 01	0 02	0 03	0 04	82112
4	21098	8500	0 050	5965	4	1	1	2	0 02	0 04	0 05	0 07	101795
6	17774	7150	0 050	4361	10	2	3	5	0 03	0 06	0 09	0 12	100569
8	17060	4965	0 050	3496	79	15	30	45	0 04	0 08	0 12	0 16	93996
10	24489	8500	0 050	4246	5	1	2	2	0 06	0 11	0 17	0 23	96818
12	22979	6100	0 050	3816	32	6	11	17	0 07	0 14	0 21	0 27	93182
14	26370	7350	0 050	3989	13	2	4	6	0 08	0 16	0 24	0 32	92436
16	24390	7150	0 050	3870	14	2	4	6	0 09	0 19	0 28	0 38	92067
18	20029	6200	0 050	3586	26	4	8	13	0 11	0 22	0 33	0 44	91047
20	18180	7000	0 050	2980	14	2	4	5	0 13	0 26	0 39	0 52	91803
24	20751	6900	0 050	3152	17	2	5	7	0 17	0 34	0 51	0 68	92406

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3005000

TABLE 3-22
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 60000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1 1/2	42974	9000	0 050	12152	2	1	1	2	0 00	0 01	0 01	0 02	81484
2	45959	11400	0 050	12431	1	0	0	1	0 01	0 01	0 02	0 03	98050
2 1/2	58322	9800	0 050	12979	2	1	1	2	0 01	0 02	0 03	0 03	93948
3	37867	6350	0 050	8584	19	6	12	18	0 01	0 02	0 04	0 05	94343
4	25318	8500	0 050	6742	4	1	2	2	0 02	0 04	0 06	0 81	115066
6	21329	7150	0 050	5018	10	2	4	6	0 03	0 07	0 10	1 53	115702
8	20472	4965	0 050	4107	82	18	35	53	0 05	0 09	0 14	0 18	110439
10	29387	8500	0 050	5011	6	1	2	3	0 07	0 13	0 20	2 04	114269
12	27575	6100	0 050	4550	33	7	13	20	0 08	0 16	0 25	0 33	111107
14	31644	7350	0 050	4787	13	2	5	7	0 10	0 19	0 29	0 38	110923
16	29268	7150	0 050	4644	15	3	5	8	0 11	0 23	0 34	0 45	110480
18	24035	6200	0 050	4304	27	5	10	15	0 13	0 26	0 39	0 53	109257
20	21816	7000	0 050	3576	15	2	4	7	0 15	0 31	0 46	0 62	110164
24	24901	6900	0 050	3782	18	3	5	8	0 21	0 41	0 62	0 82	110887

Bolt Yield Stress = 111,700 psi (SA193 B7)

DGmax = 0 050 in

Calculation 3006000

TABLE 3-23
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 15%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	25068	7650	0 050	8187	2	1	1	2	0 00	0 01	0 01	0 01	56236
2	26809	9690	0 050	8396	1	0	0	0	0 00	0 01	0 04	0 02	67739
2-1/2	34021	8330	0 050	8423	2	0	1	1	0 01	0 01	0 03	0 02	61963
3	22089	5398	0 050	5516	17	4	9	13	0 01	0 02	0 03	0 03	61572
4	14769	7225	0 050	4487	4	1	1	2	0 01	0 03	0 09	0 05	78294
6	12442	6078	0 050	3215	10	1	3	4	0 02	0 04	0 14	0 09	75491
8	11942	4220	0 050	2513	74	12	24	37	0 03	0 06	0 14	0 11	68285
10	17143	7225	0 050	3035	5	1	1	2	0 04	0 08	0 28	0 16	69832
12	16086	5185	0 050	2693	30	4	9	13	0 05	0 10	0 26	0 19	66019
14	18459	6248	0 050	2792	12	2	3	5	0 06	0 11	0 33	0 22	64705
16	17073	6078	0 050	2709	13	2	3	5	0 07	0 13	0 38	0 26	64447
18	14021	5270	0 050	2510	24	3	7	10	0 08	0 15	0 40	0 31	63733
20	12726	5950	0 050	2086	13	1	3	4	0 09	0 18	0 52	0 36	64262
24	14525	5865	0 050	2206	16	2	4	5	0 12	0 24	0 64	0 48	64684

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0.050 in

Calculation 3003515

TABLE 3-24
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 25%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	25068	6750	0 050	7775	2	1	1	2	0 00	0 01	0 01	0 01	54366
2	26809	8550	0 050	7967	1	0	0	0	0 00	0 01	0 01	0 02	65367
2-1/2	34021	7350	0 050	8103	2	0	1	1	0 01	0 01	0 02	0 02	60318
3	22089	4763	0 050	5325	17	5	9	11	0 01	0 01	0 02	0 03	60111
4	14769	6375	0 050	4279	3	1	1	2	0 01	0 03	0 04	0 05	75899
6	12442	5363	0 050	3107	9	2	3	5	0 02	0 04	0 06	0 08	73906
8	11942	3724	0 050	2469	73	13	19	36	0 03	0 05	0 08	0 11	67587
10	17143	6375	0 050	2993	5	1	1	2	0 04	0 08	0 12	0 16	69292
12	16086	4575	0 050	2679	30	5	10	12	0 05	0 10	0 14	0 19	65836
14	18459	5513	0 050	2792	12	2	3	4	0 06	0 11	0 17	0 22	64705
16	17073	5363	0 050	2709	13	2	4	5	0 07	0 13	0 20	0 26	64447
18	14021	4650	0 050	2510	24	4	7	10	0 08	0 15	0 23	0 31	63733
20	12726	5250	0 050	2086	13	2	2	5	0 09	0 18	0 27	0 36	64262
24	14525	5175	0 050	2206	16	2	3	6	0 12	0 24	0 36	0 48	64684

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3003525

TABLE 3-25
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 33%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	25068	6030	0 050	7446	2	1	1	2	0 00	0 00	0 01	0 01	52870
2	26809	7638	0 050	7623	1	0	0	1	0 00	0 01	0 01	0 02	63469
2-1/2	34021	6566	0 050	7848	2	1	1	2	0 01	0 01	0 01	0 02	59001
3	22089	4255	0 050	5173	17	5	10	15	0 01	0 01	0 02	0 03	58942
4	14769	5695	0 050	4113	3	1	1	2	0 01	0 02	0 03	0 05	73982
6	12442	4791	0 050	3020	9	2	3	5	0 02	0 04	0 05	0 08	72638
8	11942	3327	0 050	2434	73	14	29	43	0 03	0 05	0 07	0 11	67028
10	17143	5695	0 050	2959	5	1	1	2	0 04	0 08	0 11	0 16	68860
12	16086	4087	0 050	2667	30	5	11	16	0 05	0 10	0 13	0 19	65690
14	18459	4925	0 050	2792	12	2	4	6	0 06	0 11	0 15	0 22	64705
16	17073	4791	0 050	2709	13	2	4	6	0 07	0 13	0 18	0 26	64447
18	14021	4154	0 050	2510	24	4	8	12	0 08	0 15	0 21	0 31	63733
20	12726	4690	0 050	2086	13	2	4	5	0 09	0 18	0 24	0 36	64262
24	14525	4623	0 050	2206	16	2	4	7	0 12	0 24	0 32	0 48	64684

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3003533

TABLE 3-26
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 50%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	25068	6030	0 050	7446	2	1	1	2	0 00	0 00	0 01	0 01	52870
2	26809	7638	0 050	7623	1	0	0	1	0 00	0 01	0 01	0 02	63469
2 1/2	34021	6566	0 050	7848	2	1	1	2	0 01	0 01	0 01	0 02	59001
3	22089	4255	0 050	5173	17	5	10	15	0 01	0 01	0 02	0 03	58942
4	14769	5695	0 050	4113	3	1	1	2	0 01	0 02	0 03	0 05	73982
6	12442	4791	0 050	3020	9	2	3	5	0 02	0 04	0 05	0 08	72638
8	11942	3327	0 050	2434	73	14	29	43	0 03	0 05	0 07	0 11	67028
10	17143	5695	0 050	2959	5	1	1	2	0 04	0 08	0 11	0 16	68860
12	16086	4087	0 050	2667	30	5	11	16	0 05	0 10	0 13	0 19	65690
14	18459	4925	0 050	2792	12	2	4	6	0 06	0 11	0 15	0 22	64705
16	17073	4791	0 050	2709	13	2	4	6	0 07	0 13	0 18	0 26	64447
18	14021	4154	0 050	2510	24	4	8	12	0 08	0 15	0 21	0 31	63733
20	12726	4690	0 050	2086	13	2	4	5	0 09	0 18	0 24	0 36	64262
24	14525	4623	0 050	2206	16	2	4	7	0 12	0 24	0 32	0 48	64684

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3003550

TABLE 3-27
300# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	25068	9000	0 050	8805	2	0	1	1	0 00	0 01	0 01	0 01	59041
2	26809	11400	0 050	9039	1	0	0	0	0 00	0 01	0 01	0 02	71297
2-1/2	34021	9800	0 050	8902	2	0	1	1	0 01	0 01	0 02	0 02	64431
3	22089	6350	0 050	5802	17	4	8	12	0 01	0 02	0 02	0 03	63765
4	14769	8500	0 050	4798	4	1	1	2	0 01	0 03	0 04	0 06	81888
6	12442	7150	0 050	3377	10	1	3	4	0 02	0 04	0 07	0 09	77869
8	11942	4965	0 050	2579	74	11	22	33	0 03	0 06	0 09	0 11	69333
10	17143	8500	0 050	3098	5	1	1	2	0 04	0 08	0 12	0 17	70642
12	16086	6100	0 050	2715	30	4	8	12	0 05	0 10	0 15	0 20	66294
14	18459	7350	0 050	2792	12	1	3	4	0 06	0 11	0 17	0 22	64705
16	17073	7150	0 050	2709	13	2	3	5	0 07	0 13	0 20	0 26	64447
18	14021	6200	0 050	2510	24	3	6	9	0 08	0 15	0 23	0 31	63733
20	12726	7000	0 050	2086	13	1	3	4	0 09	0 18	0 27	0 36	64262
24	14525	6900	0 050	2206	16	2	3	5	0 12	0 24	0 36	0 48	64684

Bolt Yield Stress = 101,500 psi (SA193-B7)

DGmax = 0 050 in

Calculation 3003500Y

TABLE 3-28
600# ANSI FLANGE AND GASKET DATA

Flange Diameter (in)	*-----GASKET-----*					*-----BOLTS-----*			
	OD (in)	ID (in)	Width (in)	Area (sq in)	Pressure Area (sq in)	Number	Diameter (in)	Area (sq in)	Length (in)
Flanges									
1-1/2	2.750	2.125	0.3125	2.393	3.547	4	3/4	0.3340	2.375
2	3.375	2.750	0.3125	3.007	5.940	8	5/8	0.2256	2.625
2-1/2	3.875	3.250	0.3125	3.497	8.296	8	3/4	0.3340	2.875
3	4.750	4.000	0.3750	5.154	12.566	8	3/4	0.3340	3.125
4	5.875	4.750	0.5625	9.388	17.721	8	7/8	0.4612	3.625
6	8.250	6.875	0.6875	16.334	37.122	12	1	0.6051	4.375
8	10.375	8.875	0.7500	22.678	61.862	12	1-1/8	0.7896	5.000
10	12.500	10.813	0.8438	30.897	91.821	16	1-1/4	0.9985	5.625
12	14.750	12.875	0.9375	40.681	130.192	20	1-1/4	0.9985	5.875
14	16.000	14.250	0.8750	41.577	159.485	20	1-3/8	1.2319	6.125
16	18.250	16.250	1.0000	54.192	207.394	20	1-1/2	1.4899	6.625
18	20.750	18.500	1.1250	69.360	268.803	20	1-5/8	1.7723	7.125
20	22.750	20.500	1.1250	76.429	330.064	24	1-5/8	1.7723	7.625
24	27.000	24.750	1.1250	91.450	481.105	24	1-7/8	2.4107	8.625
Tubesheet									
44	47.625	45.625	1.0000	146.48	1634.92	--	--	--	--

Bolt areas correspond to tensile stress area and not thread root area which is the basis for the bolt torque values

TABLE 3-29
600# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 35000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	25068	9000	0 050	8872	2	0	1	1	0 00	0 01	0 01	0 01	59490
2	26809	11400	0 050	9750	1	0	0	0	0 01	0 01	0 02	0 03	76901
2-1/2	34021	9800	0 050	9132	2	0	1	1	0 01	0 02	0 02	0 03	66101
3	22089	6350	0 050	5925	18	4	8	12	0 01	0 02	0 03	0 04	65120
4	15659	8500	0 050	6516	3	1	1	2	0 02	0 03	0 05	0 07	77152
6	17258	7150	0 050	5573	8	2	3	5	0 03	0 05	0 08	0 10	68290
8	15510	4965	0 050	4099	62	13	27	40	0 04	0 07	0 11	0 14	64659
10	19058	8500	0 050	5014	4	1	1	2	0 05	0 11	0 16	0 21	68638
12	17935	6100	0 050	4217	23	4	9	13	0 06	0 12	0 19	0 25	64921
14	21007	7350	0 050	4197	10	2	3	5	0 07	0 15	0 22	0 29	65008
16	19416	7150	0 050	3843	11	2	3	5	0 09	0 18	0 26	0 35	63699
18	18999	6200	0 050	3713	24	4	8	12	0 11	0 21	0 32	0 43	63159
20	20684	7000	0 050	3703	14	2	4	6	0 13	0 26	0 38	0 51	63734
24	23502	6900	0 050	3570	17	3	5	8	0 18	0 36	0 54	0 72	64684
Tubesheet													
44	7108	7108	0 019	584	20	1	2	5	0 41	0 83	1 24	1 66	54339

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 6003500

TABLE 3-30
600# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 20000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect. (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	14325	9000	0 050	7554	2	0	1	1	0 00	0 01	0 01	0 01	50652
2	15320	11400	0 050	8327	1	0	0	0	0 01	0 01	0 02	0 02	65680
2-1/2	19441	9800	0 050	7314	2	0	1	1	0 01	0 01	0 02	0 03	52938
3	12622	6350	0 050	4631	17	3	7	11	0 01	0 02	0 02	0 03	50898
4	8948	8500	0 050	5317	3	0	1	2	0 01	0 03	0 04	0 05	62955
6	9862	7150	0 050	4100	7	1	3	5	0 02	0 04	0 06	0 08	50240
8	8863	4965	0 050	2768	57	9	18	33	0 02	0 05	0 07	0 10	43668
10	10890	8500	0 050	3327	3	0	1	2	0 04	0 07	0 11	0 14	45537
12	10248	6100	0 050	2640	21	3	6	11	0 04	0 08	0 12	0 16	40637
14	12004	7350	0 050	2511	9	1	2	4	0 04	0 09	0 13	0 17	38888
16	11095	7150	0 050	2228	10	1	2	4	0 05	0 10	0 15	0 20	36919
18	10857	6200	0 050	2122	21	2	5	9	0 06	0 12	0 18	0 24	36091
20	11820	7000	0 050	2116	12	1	2	5	0 07	0 15	0 22	0 29	36420
24	13430	6900	0 050	2040	16	1	3	6	0 10	0 21	0 31	0 41	36962
Tubesheet													
44	4061	4061	0 011	334	365	16	41	95	0 24	0 47	0 71	0 95	31051

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 6002000

TABLE 3-31
600# FLANGE GASKET STRESS, GROSS LEAK PRESSURE, AND LEAK RATE

INITIAL BOLT STRESS = 60000 psi

JOINT RELAXATION = 0%

Flange Diameter (in)	Eff Gasket Stress (psi)	Act Gasket Stress (psi)	Gasket Deflect (in)	Gross Leak Pressure (psi)	Leak Rate at GLP (mg/sec)	Leak Rate at 25GLP (mg/sec)	Leak Rate at 50GLP (mg/sec)	Leak Rate at 75GLP (mg/sec)	Leak Area at 1 25GLP (sq in)	Leak Area at 1 5GLP (sq in)	Leak Area at 1 75GLP (sq in)	Leak Area at 2 0GLP (sq in)	Bolt Stress at 2 0GLP (psi)
Flanges													
1-1/2	42974	9000	0 050	11069	2	1	1	2	0 00	0 01	0 01	0 02	74220
2	45959	11400	0 050	12121	1	0	0	1	0 01	0 02	0 03	0 03	95603
2-1/2	58322	9800	0 050	12163	2	1	1	2	0 01	0 02	0 03	0 04	88038
3	37867	6350	0 050	8082	19	6	11	17	0 01	0 03	0 04	0 06	88822
4	26844	8500	0 050	8514	3	1	1	2	0 02	0 04	0 06	0 09	100815
6	29585	7150	0 050	8029	8	2	5	7	0 04	0 07	0 11	0 15	98373
8	26588	4965	0 050	6317	68	21	41	62	0 06	0 11	0 17	0 22	99645
10	32671	8500	0 050	7827	4	1	2	3	0 08	0 16	0 25	0 33	107141
12	30745	6100	0 050	6846	25	7	14	21	0 10	0 20	0 30	0 40	105393
14	36012	7350	0 050	7008	11	3	5	8	0 12	0 24	0 36	0 49	108541
16	33285	7150	0 050	6537	12	3	6	9	0 15	0 30	0 45	0 60	108333
18	32570	6200	0 050	6366	27	7	14	20	0 18	0 37	0 55	0 73	108273
20	35459	7000	0 050	6348	15	4	7	11	0 22	0 44	0 66	0 88	109259
24	40290	6900	0 050	6120	20	4	9	13	0 31	0 62	0 92	1 23	110887
Tubesheet													
44	12184	12184	0 033	1002	1	0	0	0	0 71	1 42	2 13	2 84	93153

Bolt Yield Stress = 111,700 psi (SA193-B7)

DGmax = 0 050 in

Calculation 6006000

FIGURE 3-1
INCREMENTAL GASKET LOAD TO TOTAL PRESSURE LOAD RATIO
FOR 150 TO 600 LB. RATED FLANGES

FIGURE 3-2
TYPICAL LOAD-DEFLECTION DIAGRAM SHOWING A STANDARD TEST SEQUENCE

FIGURE 3-3

A TYPICAL MASS-LEAK-RATE VS. GASKET STRESS PLOT

FIGURE 3-4

GASKET STRESS VS. TIGHTNESS PARAMETER FOR TWO SPIRAL-WOUND ASBESTOS GASKETS,
CYCLIC TESTS WITH WATER

FIGURE 3-5

MASS LEAK RATE VERSUS PRESSURE FOR 8"-150# FLANGE

4. VALVES

Three failure modes are postulated for the various valves present in the RWP safety systems under consideration. These include failure of the valve body, failure of the stem packing, and failure of the bolted bonnet. Since the valve body thickness is typically greater than that of the adjacent piping, it was judged that failure of the adjacent piping will occur prior to failure of the valve body. Also the types of valve stem packing currently used in most nuclear plants tends to compress under high pressure conditions providing a greater resistance to leakage. Although it is certainly possible that the stem packing for some valves could deteriorate in response to service conditions, it was judged that any resulting leak rate or leak area would be quite small and have a negligible effect on both the valve and system operation. Thus, it was felt that the only credible failure mode for the system valves pertains to failure of the bolted bonnet seal.

A total of eight bolted bonnet valves were evaluated for the RWP. The results of the evaluation are presented in Table 4-1. All but two of the valves examined have design capacities of 2485 psig at 650 degrees Fahrenheit and hence are expected to have leak capacities well above 2485 psig. The remaining two valves have pressure capacities and estimated leak areas provided in Table 4-1.

TABLE 4-1

VALVES

Valves 1 NI-184B (Safety Injection System)
 1 NI-185A (Safety Injection System)

Median Leak Pressure 2360 psig; $\beta = 0.1$

For Pressures $2360 < p < 4340$ psig,

Median Leak Area = $4.10^{-5} \Delta p$ in²; $\beta = 0.7$

where Δp = Pressure - 2360 psig

(i.e., at Pressure = 3360 psig; $\Delta p = 1000$ psig

Median Leak Area = 0.04 in²)

The following valves have leak capacities well in excess of 2485 psig.

1 NI 173A (Safety Injection System)
 1 NI 178B (Safety Injection System)
 1 NI 183B (Safety Injection System)
 1 NI 152B (Safety Injection System)
 1 ND 1B/2A/36B/37A (Residual Heat Removal System)
 1 NI 162A (Safety Injection System)

5. PUMPS

There are two pumps of interest in this analysis; the multistage Safety Injection Pump and the single-stage Residual Heat Removal Pump. These pumps were designed for pressures of 1750 and 600 psi, respectively. The casing and mechanical seal of the Residual Heat Removal Pump were hydrostatically tested up to pressures of 935 and 1200 psig, respectively. For the Safety Injection Pump, the ultimate pressure capacities of the following items were estimated and found to be greater than 2500 psig; the casing, bolted flange connection of the casing, and the mechanical seal. Similarly, for the Residual Heat Removal Pump the following items were evaluated and found to have pressure capacities in excess of 2500 psig; the casing, the bolted flange connection of the casing, the mechanical seal, and the seal water cooler tubes. Thus, the pumps of interest in this investigation are not expected to leak when subjected to pressures as high as the Reactor Coolant System operating pressure.

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