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Risk Assessment for the Intentional Depressurization Strategy in PWRs

Prepared by
S. E. Dingman

Sandia National Laboratories
Operated by
Sandia Corporation

Prepared for
U.S. Nuclear Regulatory Commission

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Prepared by
S. E. Dingman

Sandia National Laboratories
Albuquerque, NM 87185

Prepared for
Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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Abstract

An accident management strategy has been proposed in which the reactor coolant system is intentionally depressurized during an accident. The aim is to reduce the containment pressurization that would result from high pressure ejection of molten debris at vessel breach. Probabilistic risk assessment (PRA) methods were used to evaluate this strategy for the Surry nuclear power plant. Sensitivity studies were conducted using event trees that were developed for the NUREG-1150 study. It was found that depressurization (intentional or unintentional) had minimal impact on the containment failure probability at vessel breach for Surry because the containment loads assessed for NUREG-1150 were not a great threat to the containment survivability. An updated evaluation of the impact of intentional depressurization on the probability of having a high pressure melt ejection was then made that reflected analyses that have been performed since NUREG-1150 was completed. The updated evaluation confirmed the sensitivity study conclusions that intentional depressurization has minimal impact on the probability of a high pressure melt ejection. The updated evaluation did show a slight benefit from depressurization because depressurization delayed core melting, which led to a higher probability of recovering emergency core coolant injection, thereby arresting the core damage.

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Acronyms

AFW	(Steam Generator) Auxiliary Feedwater
APET	Accident Progression Event Tree
ATWS	Anticipated Transient Without Scram
CDF	Core Damage Frequency
CPI	Containment Performance Improvement
DCH	Direct Containment Heating
ECC	Emergency Core Coolant
FCI	Fuel Coolant Interaction
HPME	High Pressure Melt Ejection
INEL	Idaho National Engineering Laboratory
LOCA	Loss of Coolant Accident
NRC	Nuclear Regulatory Commission
PORV	Power-operated Relief Valve
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RY	Reactor Year
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratories
SRV	Safety Relief Valve

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Dr. James Han, the NRC technical monitor for this work, provided valuable guidance that greatly improved the analyses. We appreciate his assistance and support. We would also like to thank Tom Brown and Allen Camp of SNL for the numerous insights and suggestions they provided regarding PRA modeling, Darrell Knudson of INEL for furnishing and interpreting the results of the SCDAP/RELAP5 calculations, Ann Shiver of SNL for developing ac recovery probabilities, and Tom Brown and Ed Boucheron of SNL for providing review comments which greatly improved the report.

Executive Summary

The Nuclear Regulatory Commission (NRC) has been developing accident management strategies with the potential for terminating or mitigating severe accidents at nuclear power plants. Mitigative actions have been identified to address a particular concern that arises if vessel breach occurs while the reactor coolant system (RCS) is at an elevated pressure. For such cases, the pressurized blowdown of the RCS could sweep the molten material and gases exiting the RCS out of the reactor cavity and into the containment. This could lead to very rapid and efficient heat transfer to the atmosphere, possibly accompanied by oxidation reactions and hydrogen burning that further enhance the energy transfer. The pressurization accompanying this process, which is labeled direct containment heating (DCH), could potentially fail containment.

For pressurized water reactors (PWRs), a strategy has been developed to mitigate the DCH threat. In this strategy, termed the intentional depressurization strategy, the RCS would be depressurized after the core uncovers and the core exit thermocouples reach 922 K (1200°F) by intentionally opening the power-operated relief valves (PORVs). If this action were to succeed in reducing the RCS pressure, the driving force for DCH and the resultant containment threat would be eliminated. SCDAP/RELAP5 code evaluations indicate that this strategy can reduce the pressure sufficiently to mitigate the DCH loads for a short-term station blackout sequence (immediate loss of all coolant injection when ac power is lost) at some plants.

The Probabilistic risk assessment (PRA) methods developed by Sandia National Laboratories (SNL) for NUREG-1150 were used to evaluate the impact of the intentional depressurization strategy for the Surry plant for station blackout sequences. To evaluate the intentional depressurization strategy, an accident progression event tree (APET) is used. The APET is a logical framework used to determine possible accident progressions for sequences that have proceeded to core damage, and the likelihood of each accident progression. For this study, the APET delineates various pathways that can occur during the core melt progression and estimates which of these pathways would lead to vessel breach while at high system pressure. The probability of having vessel breach while at a high system pressure can then be compared for cases with and without intentional depressurization, giving an indication of the effectiveness of the strategy.

The PRA evaluation of intentional depressurization was conducted in two phases: (1) examination of the sensitivity of the core damage progression to various accident progression uncertainties, and (2) updated evaluation using uncertainty distributions that reflect the current knowledge of the phenomena involved. The first phase of the evaluation was performed to determine which questions in the APET have the largest impact on the probability of high pressure melt ejection and containment failure, so that more emphasis could be directed at determining their uncertainty distributions during the second phase of the intentional depressurization evaluation.

The sensitivity studies indicated that with the NUREG-1150 treatment of DCH loads, intentional depressurization would not give a significant reduction in overall risk at Surry because:

- a large fraction of the station blackout sequences have inadvertent failures in the reactor system boundary that lead to depressurization,

- a large fraction of the core melt sequences are terminated before vessel breach because of ac power recovery, and

- those sequences that continue to vessel breach at high pressure do not usually result in sufficient loads at vessel breach to fail containment.

In fact, when in-vessel fuel-coolant interactions (FCIs) are included, intentional depressurization might actually increase risk slightly because FCIs are more likely at lower pressures. However, the updated evaluation indicated that the probability of arresting core damage following ac recovery was higher with intentional depressurization. The net effect of these two factors (higher probability of FCIs at low pressure, higher probability of arresting core damage before the FCI occurred) was that the probability of FCIs was actually reduced when the RCS was intentionally depressurized.

The sensitivity studies have shown that the key factor for Surry in determining risks related to containment failure from loads generated by direct containment heating are the loads themselves and the containment structural integrity, rather than the likelihood of being at a reduced pressure at vessel breach. This is a Surry-specific result, but the methodology described in this report could be used to evaluate other plants.

The updated evaluation indicates that the probability of a high pressure vessel breach is even lower than estimated for the NUREG-1150 study. Intentional depressurization decreased the probability of being at high pressure at vessel breach, but by only a small amount since the probability of a high pressure vessel breach is not very high even without intentional depressurization. This study has indicated that intentional depressurization at Surry would give minimal benefit. It has also demonstrated a methodology that could be used to evaluate the strategy for other plants that might see more benefit from depressurization, possibly Babcock & Wilcox or Combustion Engineering plants.

1.0 Introduction and Objectives

The Nuclear Regulatory Commission (NRC) has been developing accident management strategies with the potential for terminating or mitigating severe accidents at nuclear power plants. Mitigative actions have been identified to address a particular concern that arises if vessel breach occurs while the reactor coolant system (RCS) is at an elevated pressure. For such cases, the pressurized blowdown of the RCS could sweep the molten material and gases exiting the RCS out of the reactor cavity and into the containment as shown schematically in Figure 1.1. This could lead to very rapid and efficient heat transfer to the atmosphere, possibly accompanied by oxidation reactions and hydrogen burning that further enhance the energy transfer. The pressurization accompanying this process, which is labeled direct containment heating (DCH), could potentially fail containment.

For pressurized water reactors (PWRs), a strategy has been developed to mitigate the DCH threat. In this strategy, termed the intentional depressurization strategy, the RCS would be depressurized after the core uncovers and the core exit thermocouples reach 922 K (1200°F) by intentionally opening the power-operated relief valves (PORVs).[1] If this action were to succeed in reducing the RCS pressure, the driving force for DCH and the resultant containment threat would be eliminated. SCDAP/RELAP5 code evaluations[2,3] indicate that this strategy can reduce the pressure sufficiently to mitigate the DCH loads for a short-term station blackout sequence (immediate loss of all coolant injection when ac power is lost) at some plants, and is thus worthy of further evaluation.

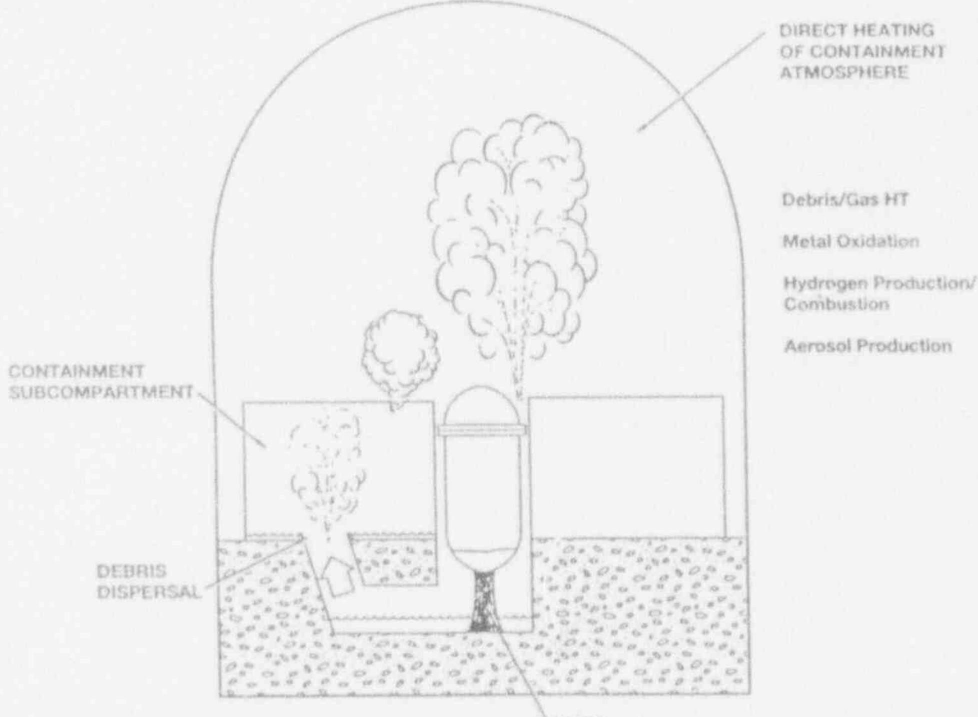

Figure 1.1 Schematic of direct containment heating process

The effectiveness of the intentional depressurization strategy is influenced by other factors that could make depressurization either unnecessary or undesirable. For example, while some station blackout sequences would be expected to proceed to vessel breach at high RCS pressure, there is also a possibility of induced failures (stuck-open PORV, pump seal failure, hot leg failure, surge line failure) that could depressurize the RCS before vessel breach. If there is a sufficiently high probability that these failures occur, the intentional depressurization strategy would have minimal impact. Figure 1.2 is a schematic of these possible failure locations. An additional possibility involves recovering ac power and restoring emergency core coolant (ECC) during the in-vessel core melt progression phase, which could arrest core damage and prevent vessel breach. This factor would also reduce the impact of the intentional depressurization strategy. Further, intentionally depressurizing the RCS would impact the accident timing, and thus the time window available for recovering ac power.

When evaluating the intentional depressurization strategy, it is also important to understand how this action may affect other portions of the accident progression. Severe reactor accidents involve extremely complex system and phenomenological responses that are often nonintuitive. For example, intentional depressurization might reduce the threat from DCH, but the lower RCS pressure increases the likelihood of in-vessel fuel-coolant interactions (FCIs), which could fail the reactor vessel and pressurize containment. Thus, integral evaluations that incorporate all such possibilities must be performed to determine the impact of the strategy.

The phenomena in severe accidents are also highly uncertain. The current uncertainty in the phenomenology yields a wide range of potential outcomes that must be considered when evaluating intentional depressurization.

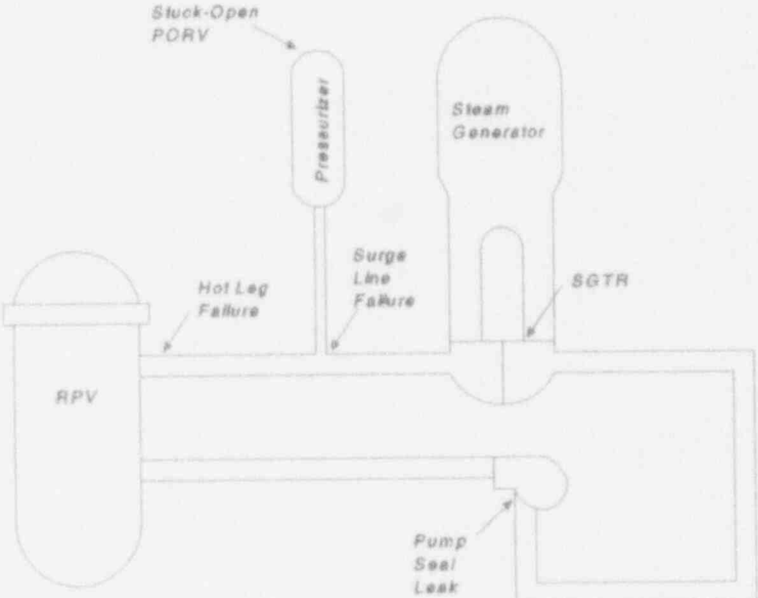

Figure 1.2 Potential RCS failure locations

The Probabilistic Risk Assessment (PRA) methods developed by Sandia National Laboratories (SNL) for NUREG-1150[4] provide an integrated analysis framework that can be used to evaluate the potential ramifications of a certain action over a wide range of possible outcomes. The framework provides the capability to compare various strategies based on selected risk measures, such as health and economic risk or the probability of vessel breach occurring at high pressure. A key area where the NUREG-1150 methods can contribute to accident management is in the treatment of uncertainties in accident progressions. PRA results can supplement detailed deterministic calculations by identifying alternative outcomes for the important accident sequences.

These PRA techniques were thus used to evaluate the impact of the intentional depressurization strategy for the Surry plant for station blackout sequences. To evaluate the intentional depressurization strategy, an accident progression event tree (APET) is used. The APET is a logical framework used to determine possible accident progressions for sequences that have proceeded to core damage, and the likelihood of each accident progression. For this study, the APET delineates various pathways that can occur during the core melt progression and estimates which of these pathways would lead to vessel breach while at high system pressure. The probability of having vessel breach while at a high system pressure can then be compared for cases with and without intentional depressurization, giving an indication of the effectiveness of the strategy.

The PRA evaluation of intentional depressurization was conducted in two phases: (1) examination of the sensitivity of the core damage progression to various accident progression uncertainties, and (2) updated evaluation using uncertainty distributions that reflect the current knowledge of the phenomena involved. The first phase of the evaluation was performed using the NUREG-1150 quantification for key parameters with the objective of determining which questions in the APET have the largest impact on the risk results. The second phase of the intentional depressurization evaluation used updated SCDAP/RELAP5 analyses to improve the probabilistic estimates of key parameters. The study also demonstrated the usefulness of PRA in evaluating accident management strategies.

The results of the two phases of the PRA evaluation are presented in this report. Chapter 2 includes a brief summary of PRA methodology and summarizes the base Surry risk results and sensitivity cases from NUREG-1150.[4,5] Chapter 3 describes the approach used to perform the evaluation of the impact of intentional depressurization for Surry, and Chapter 4 includes the results of the evaluations. Chapter 5 provides the conclusions of this study.

2.0 Background

This chapter provides an overview of PRA methodology and summarizes previous risk evaluations for Surry. The intent is to summarize the perspectives gained previously which have guided the current effort.

2.1 Brief Review of PRA Methodology

A review of the PRA methodology used in NUREG-1150 is provided here for reader convenience. The discussion is a summary of the methodology discussion in Reference 4, which can be consulted for further details.

The assessment of severe accident risks can be divided into five general parts as shown schematically in Figure 2.1. The accident frequency, accident progression, source term and consequence analyses are summarized in the following paragraphs. An uncertainty analysis is included in each of the analyses to reflect uncertainties in phenomena or equipment failure rates. The risk integration combines the information from the first four parts into estimates of risk.

The accident frequency analysis estimates the frequencies of accident sequences leading to core damage. In this portion of the analysis, combinations of potential accident initiating events (e.g., a pipe break in the RCS) and system failures that could result in core damage are defined and frequencies of occurrence are calculated. A combination of event trees and fault trees is used to perform the evaluation. Individual accident sequences with similar characteristics are grouped into "plant damage states" for use in the subsequent stages of the risk evaluation. These states are defined by the operability of plant systems (e.g., the availability of containment spray systems) and by certain key physical conditions in an accident (e.g., RCS pressure).

The accident progression analysis considers the progression of the accident after the core has begun to degrade. For each general type of accident, defined by the plant damage states, the analysis considers the important characteristics of the core melting process, the challenges to the containment building, and the response of the building to those challenges. Event trees termed accident progression event trees (APETs) are used to organize and quantify the large amounts of information used in this analysis. The event trees combine information from many sources, e.g., detailed computer accident simulations and panels of experts providing interpretations of available data. The APETs used

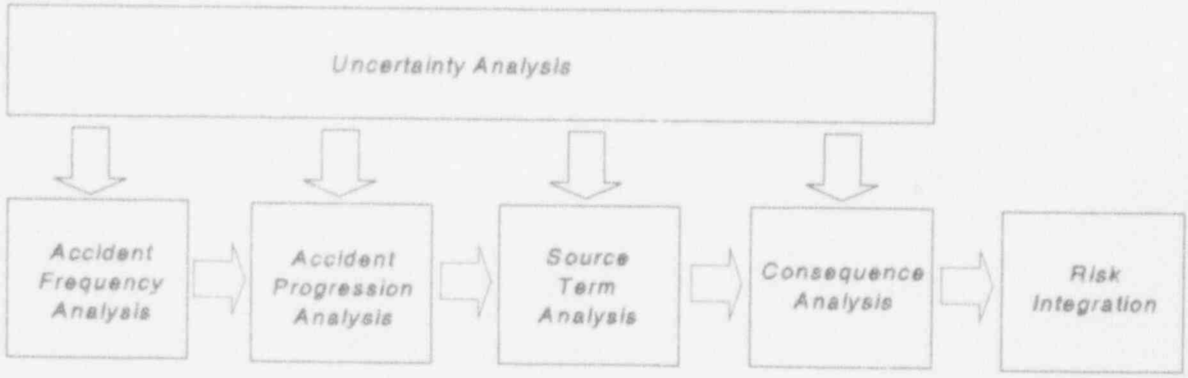

Figure 2.1 PRA analysis sequence

for NUREG-1150 produce a large set of alternative outcomes of a severe accident. These outcomes are then grouped into "accident progression bins" for use in the subsequent stages of the risk evaluation. The bins are defined by parameters such as timing of containment failure, spray system operability, and type of containment failure.

The source term analysis tracks the transport of the radioactive materials from the fuel to the reactor coolant system, then to the containment, and finally into the environment. The fractions of the core inventory released to the atmosphere, and the timing and other release information needed to calculate the offsite consequences, together are termed the "source term." The removal and retention of radioactive material by natural processes, such as deposition on surfaces, and by engineered systems, such as sprays, are accounted for in each location. Because of the complexity and cost of radioactive material transport calculations performed with detailed codes, the number of accidents that can be investigated with these codes is usually rather limited. Therefore, source terms for NUREG-1150 were calculated using simplified algorithms. Radioactive releases were then grouped according to their potential to cause early and latent cancer fatalities and warning time. Through this "partitioning" process, the large number of radioactive releases were collected into a small set of source term groups. These groups were then used in the offsite consequence calculations.

The severe accident radioactive releases are of concern because of their potential for impacting the surrounding environment and population. The impacts of such releases to the atmosphere can manifest themselves in a variety of early and delayed health effects, loss of habitability of areas close to the reactor, and economic losses. The consequence analysis stage of the PRA process estimates these offsite consequences.

2.2 Review of NUREG-1150 and Containment Performance Improvement (CPI) Risk Evaluations

The results of the NUREG-1150 study identify the sequences that are the largest contributors to risk at Surry. The dominant contributors include station blackout sequences, bypass scenarios, loss of coolant accidents (LOCAs), and other transients. Station blackout sequences involve total loss of onsite and offsite ac power, which fails ECC and leads to core damage. Station blackout sequences are further divided into (1) long-term station blackout sequences in which auxiliary feedwater is operating early in the sequence, but subsequently fails when the batteries are depleted and (2) short-term station blackout sequences in which auxiliary feedwater is failed at the onset of the transient. The operating procedures prohibit opening the PORVs (which require dc power for operation) to depressurize when ac power is not available, so station blackout sequences have the potential for progressing to vessel breach with a high RCS pressure. Station blackout sequences might thus be mitigated by intentionally depressurizing the RCS. Similar mitigation is not expected for bypass sequences or LOCAs because these sequences involve RCS breaks that would depressurize the system. The other transients do not lose ac power, and depressurization of the RCS before vessel breach is directed in the operating procedures for Surry. Therefore, only station blackout sequences were considered in this analysis when determining the impact of intentional depressurization.

The fractional contributions of the various sequences to the core damage frequency (CDF) and to overall risk for internal events are shown in Figures 2.2 and 2.3, respectively. Long- and short-term station blackout sequences are the dominant contributors (56% and 13%, respectively) to CDF, but are much less important (15% combined) than the containment bypass scenarios to overall risk.

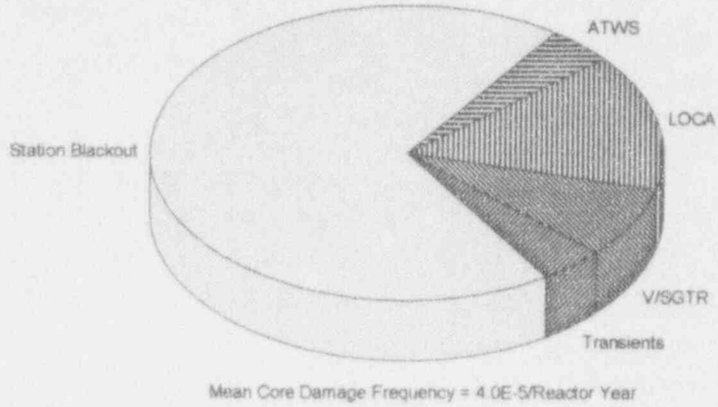

Figure 2.2 Surry core damage frequencies

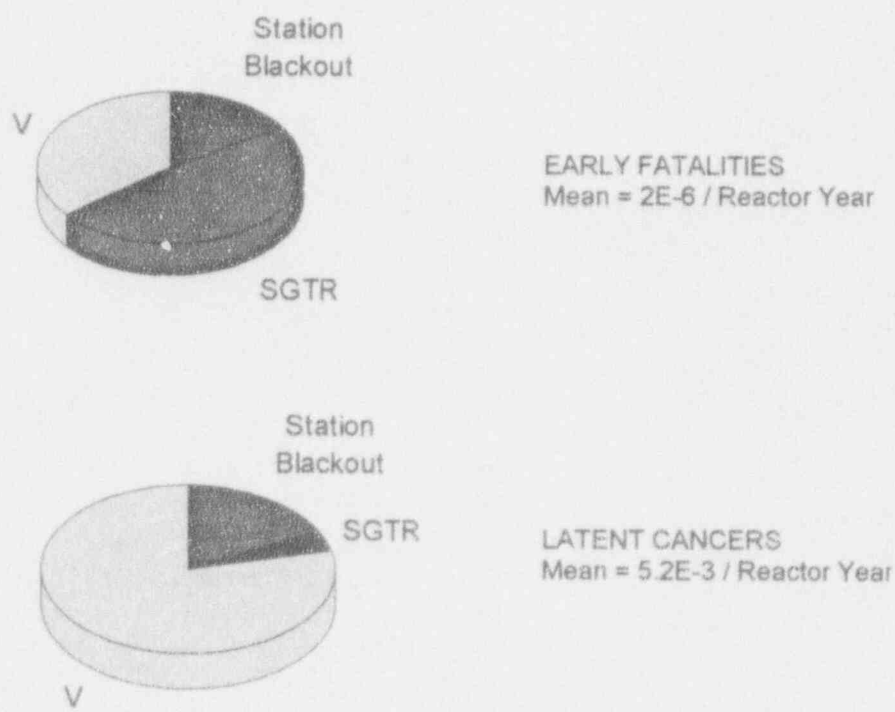

Figure 2.3 Surry risk contributors

Because of the dominance of bypass sequences to the risk at Surry, evaluation of the intentional depressurization strategy would not be expected to show a large effect in terms of overall risk reduction. It is possible that station blackout sequences will be more important to risk at other plants because of differences among plants that will affect core damage frequencies, accident progression pathways, source terms, and consequences. To provide a more useful evaluation of intentional depressurization that can provide some guidance when evaluating the strategy for other plants, the results of the PRA evaluations of intentional depressurization for the station blackout sequences are expressed in terms of conditional probabilities, that is, the fraction of a particular core damage sequence that will follow a particular accident progression **after** core damage has occurred.

A sensitivity evaluation was performed in the NUREG-1150 study and reported in Reference 5 in which temperature-induced steam generator tube ruptures and hot leg breaks were precluded. The results for short- and long-term station blackouts are compared to the NUREG-1150 base results in Table 2.1. The effect on the reactor coolant system (RCS) pressure at vessel breach is shown by a comparison of the mean probabilities of having vessel breach at four pressure levels: (1) system setpoint pressure (approximately 2500 psia), (2) high pressure (600 - 2000 psia), (3) intermediate pressure (200 - 600 psia), and (4) low pressure (less than 200 psia). There is a significant difference between the NUREG-1150 and sensitivity evaluations, but there is still a large fraction of cases that are depressurized at vessel breach when the temperature-induced failures (SGTR and hot leg break) are not allowed. This occurs because depressurization by either pump seal failure or a stuck-open PORV is still considered in the evaluation. Also shown in Table 2.1 is the impact on containment failure at vessel breach and the final containment condition. The evaluation without temperature-induced steam generator tube ruptures or hot leg failures has slightly more containment failures at vessel breach, but 99% of the station blackout sequences did not have containment failure before or at vessel breach. This occurs because **in the NUREG-1150 study**, (1) ac power is recovered, and terminates the accident in a significant fraction of the sequences, and (2) the Surry containment is not threatened by DCH loads, except for those at the high end of the probability distribution. The NUREG-1150 DCH loads are currently being updated in another NRC program; those revisions could affect this result.

An evaluation of intentional depressurization was conducted for the NRC Containment Performance Improvement (CPI) program[6] in which intentional opening of the PORVs was assumed to always occur. A full risk evaluation was not performed for the study. Instead, early fatality and latent cancer "risk potentials" were defined in the study to approximate early fatality and latent cancer risk. Using these approximations, the reductions in risk from intentional depressurization were calculated to be 5% and 3% for early fatalities and latent cancers, respectively. This small benefit was largely due to the large contribution of bypass sequences to the Surry risk.

Table 2.1 NUREG-1150 sensitivity results

Long-Term Station Blackout:

Pressure Range (psia)	At Core Uncovering	At Vessel Breach	
		Base Case	No Temperature-Induced Breaks
~ 2500	0.541	0.055	0.245
600 - 2500	0.126	0.101	0.103
200 - 600	0.333	0.190	0.191
< 200	0.000	0.654	0.461
Fraction with no Vessel Breach		.618	.618
Fraction with Containment Failure at Vessel Breach		.0083	.0124
Fraction With Final Condition of No Containment Failure		.9130	.9209

Short-Term Station Blackout:

Pressure Range (psia)	At Core Uncovering	At Vessel Breach	
		Base Case	No Temperature-Induced Breaks
~ 2500	1.000	0.029	0.150
600 - 2500	0.000	0.116	0.116
200 - 600	0.000	0.198	0.197
< 200	0.000	0.657	0.537
Fraction with no Vessel Breach		.508	.508
Fraction with Containment Failure at Vessel Breach		.0074	.0130
Fraction with Final Condition of No Containment Failure		.8862	.8948

3.0 Analysis Approach

The NUREG-1150 and CPI results discussed in Section 2.2 indicate that expressing the results in terms of either overall risk or containment failure probability would probably show little impact from the depressurization strategy for Surry. Because the strategy was not expected to have a large impact for Surry, a more generic approach was taken to evaluate the impact of intentional depressurization. The analysis approach is described in this chapter.

3.1 General Approach

The process that was used to perform the evaluation of the intentional depressurization strategy is shown in Figure 3.1. As noted in Chapter 1, the analysis was conducted in two 2 phases: (1) sensitivity evaluations to determine the key variables influencing the risk impact of intentional depressurization, and (2) an updated evaluation using the results of analyses that have been performed since the NUREG-1150 study was completed. Both phases concentrated on the accident progression step of the PRA process as shown in Figure 3.2.

In Phase 1, the Surry APET that had been used for NUREG-1150 was modified to investigate sensitivities. First, the questions in the APET that are affected by depressurization and the questions that have the largest effect on the pressure at vessel breach were identified. Both short-term (immediate loss of heat removal) and long-term (heat removal

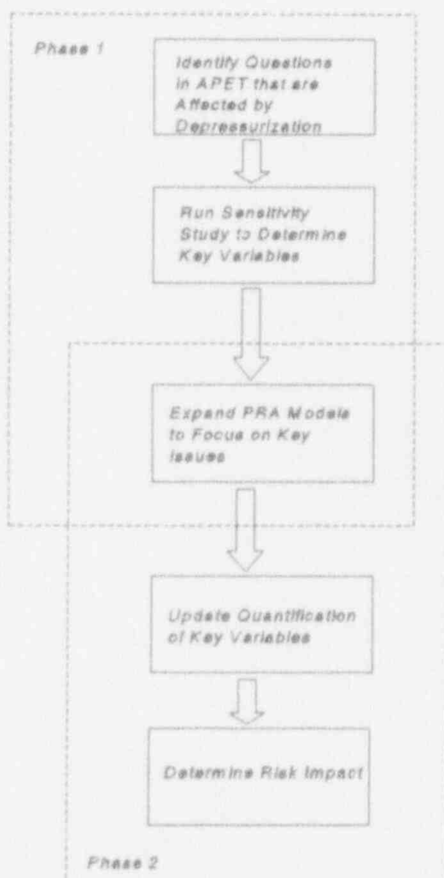

Figure 3.1 Analysis approach

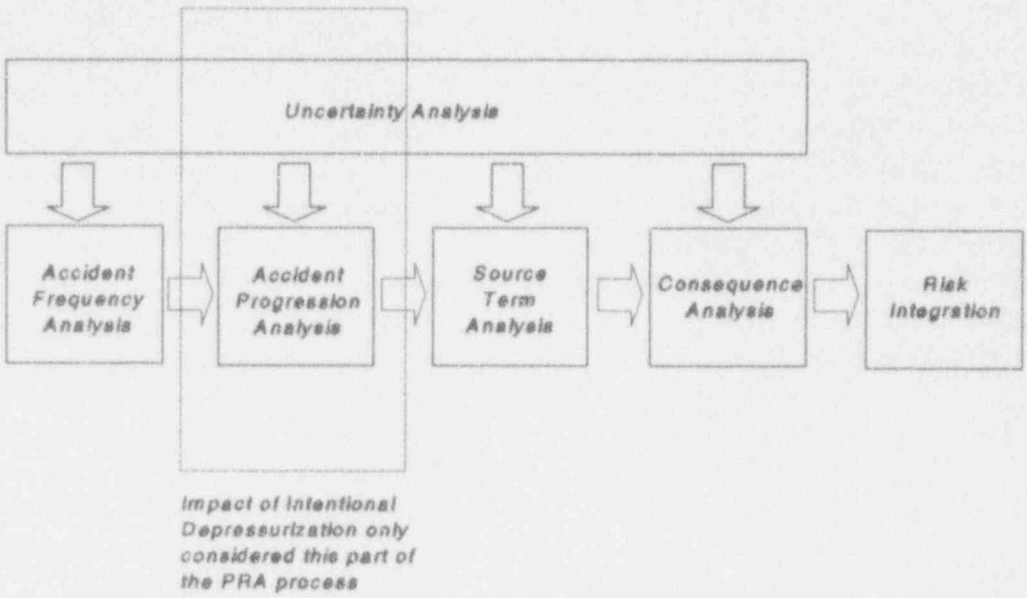

Figure 3.2 PRA step used for intentional depressurization evaluation

initially available, but failing about 4 hours after ac power loss) station blackout sequences were evaluated. Appropriate sensitivity studies to examine these parameters were then defined. The APETs for short-term and long-term station blackout sequences were modified to evaluate these sensitivities, computer calculations were performed to provide the quantification, and the results were evaluated.

In Phase 2, the treatment of key variables was modified to reflect more recent analyses for Surry. This included revised probability distributions for hot leg/surge line failure and pressure at vessel breach that were developed at Idaho National Engineering Laboratory (INEL).[7] The approach for both phases is described more fully in the following sections.

3.2 Identifying Key Phenomena

Depressurization affects the accident timing, which affects the time available for recovery actions. If depressurization were initiated before core uncovering, the core damage frequency analysis and the accident progression analysis would both be affected by this, causing a change in the CDF and relative frequency of accident progression outcomes. For this evaluation, intentional depressurization after core uncovering is being considered, so only the accident progression analysis is affected.

The Surry APET from NUREG-1150 consists of 71 questions, which can be grouped into four time regimes of the accident sequences:

- initial questions defining the plant damage state, that is, plant conditions at core uncovering and equipment status,
- early time regime - from core uncovering until just before vessel breach,
- intermediate/late intermediate time regimes - events occurring near vessel breach, and
- late/very late time regimes - after vessel breach.

The questions in the APET that are important to the depressurization evaluation were identified, and each of these is discussed in this section. A list of the affected questions is provided in Table 3.1.

Question 1 identifies whether there is a cycling PORV or a break in the RCS when the core uncovers. Most of the long-term station blackout sequences have a cycling PORV, but there are also cases with either a stuck-open PORV or pump seal failure. For NUREG-1150, the stuck-open valves were considered to be locked fully open; leaking valves were not considered. Stuck-open PORVs are treated as the class of breaks labeled S2 (0.5 - 2 in. diameter) and pump seal failures are treated as the smaller S3 breaks (less than 0.5 in. diameter). For the short-term station blackouts, there are only cycling PORVs at the time of core uncovering, because the accident proceeds rapidly enough that there is not sufficient time to introduce a significant probability of a stuck-open PORV or pump seal failure. This question is important to the depressurization evaluation because initially-occurring breaks will reduce the pressure, at least partially, even without operator actions to depressurize the system.

The RCS pressure at the start of core degradation (question 15) is used in conjunction with the RCS pressure just before vessel breach (question 23) to determine if accumulators dump before, during, or after the time when the bulk of the core damage occurs. The timing of accumulator injection affects the amount of zirconium oxidation, which affects the containment hydrogen concentration and the core composition at vessel breach. These parameters affect DCH loads and late containment burns. Although early containment failure is more severe in terms of early fatalities from an accident, late containment failures are also important because the number of latent cancer fatalities from an accident depends more on whether or not containment failure occurs than on the timing of the failure.

Questions 16, 17, 19 and 20 consider induced failures of the RCS boundary after core uncovering. When breaches are predicted, there is a higher likelihood of the system being at a lower pressure at vessel breach. Question 16 addresses the likelihood of the PORV sticking open. Question 17 addresses temperature-induced pump seal failures. As in question 1, stuck-open PORVs and pump seal failures that occur after core uncovering are treated as S2 and S3 breaks, respectively. Temperature-induced steam generator tube ruptures (SGTRs) are considered in question 19. Although this possibility is included in the APET, the NUREG-1150 expert panel concluded that this failure mechanism is much less likely than hot leg or surge line failure, which is considered in question 20. Thus, the induced steam generator tube rupture cases did not significantly impact the Surry risk results. The hot leg and surge line failure probability distributions derived for NUREG-1150 were sufficiently high to impact the results, however. These breaks are treated as large breaks (greater than 2 in. diameter), termed A breaks.

Recovery of ac power between core uncovering and vessel breach is considered in question 21. If ac power is recovered in a station blackout sequence, ECC will become available, and injecting the coolant would be expected to arrest the accident progression, if it has not proceeded to an unrecoverable state. The likelihood of being able to arrest core damage if ac power is recovered is considered in question 24. The combination of questions 21 and 24 gives the likelihood that a sequence which has proceeded to core damage will be terminated before vessel breach occurs.

The reactor pressure vessel (RPV) pressure at vessel breach, which is quantified in question 23, determines whether or not there is sufficient driving head for DCH. This, combined with the amount of water in the reactor cavity at vessel breach (question 32), the fraction of core released at vessel breach (question 33), the type of vessel breach (question

Table 3.1 Surry NUREG-1150 APET questions important to depressurization evaluation

1.	Size and location of break when core uncovers?
15.	RCS pressure at start of degradation?
16.	Do PORVs or SRVs stick open?
17.	Is there a temperature-induced pump seal failure?
19.	Is there a temperature-induced steam generator tube rupture?
20.	Is there a temperature-induced hot leg or surge line failure?
21.	Is ac power available early?
23.	RPV pressure just before vessel breach?
24.	Is core damage arrested?
25.	Baseline containment pressure just before vessel breach?
30.	Fraction zirconium oxidized in-vessel?
32.	Amount of water in the reactor cavity at vessel breach?
33.	Fraction of core released from vessel at breach?
35.	Is there an in-vessel steam explosion that fails containment?
36.	Type of vessel breach?
38.	Size of hole for debris ejection?
39.	Pressure rise at vessel breach? (large RPV hole cases)
40.	Pressure rise at vessel breach? (small RPV hole cases)
42.	Containment failure pressure?
43.	Containment failure and type of failure?

36), and the size of the hole in the RPV bottom head (question 38) determine the pressure rise during a DCH event, for the NUREG-1150 treatment. The pressure rise at vessel breach is quantified in questions 39 and 40.

The fraction of zirconium oxidized in-vessel is determined in question 30. This quantity is dependent on pressure, and it determines the amount of hydrogen in containment for a late burn.

Question 35 concerns the likelihood of an in-vessel steam explosion that fails containment (alpha mode failure). The failure likelihood depends on the RPV pressure.

The containment failure pressure is determined in question 42, and is compared to the loads at vessel breach and the baseline containment pressure (question 28) to determine the likelihood of containment failure at vessel breach in question 43.

For this evaluation, only the questions directly affecting the likelihood of breaching the RPV while at high pressure were considered. DCH loads are being examined in other NRC programs.

The Surry APET used for the NUREG-1150 study considered the likelihood of in-vessel FCIs that fail the vessel and then fail the containment by impact of a missile that is generated during vessel failure (alpha mode failure). However, the Surry APET did not address in-vessel FCIs that would fail the vessel without failing the containment by missile impact. Such FCIs could fail the containment by a different mechanism, rapid containment pressurization, at the time of vessel breach. FCIs are more likely to occur at low pressure than at high pressure, and could thus be important to the evaluation of the intentional depressurization strategy. To address this, the Surry APET was modified to account for these FCIs in several of the sensitivity evaluations as discussed in Section 3.3.

3.3 Approach for Sensitivity Evaluations

Evaluations were performed for ten variations of the base NUREG-1150 evaluation for the long-term and short-term station blackout groups to determine the sensitivity of the risk results to the key questions identified in the previous section. Seven variations were examined in which the Surry NUREG-1150 treatment of FCIs was used. Next, the base NUREG-1150 results and two of the variations were re-examined with a modified treatment of in-vessel FCIs. The specific modifications made to expand the treatment of FCIs and the modifications that were made for the other variations are discussed in the following paragraphs. A summary description of the ten variations is provided in Table 3.2. The evaluations were performed to identify areas that would be most important to quantify during the second phase of the intentional depressurization evaluation. The sensitivity results also give an indication of the possible risk impact of operator depressurization at other plants, which can and probably will have different uncertainty distributions for the affected parameters.

In Variation 1, all depressurization mechanisms in the APET were assigned probability zero, giving an indication of the impact of unintentional depressurization on the results. The potential for recovering ac power and arresting the core melt progression before vessel breach was retained in Variation 1. Variations 2 through 4 examined the impact of each of the unintentional depressurization mechanisms separately, i.e., the probability of two of the unintentional depressurization mechanisms were set to zero and the NUREG-1150 probability distribution was used for the third mechanism. In these three variations, then, the single depressurization mechanisms of pump seal failure, stuck-open PORV, and hot leg/surge line failure were examined. In Variation 5, the PORVs were always assumed to be opened by the operator after core uncovering. This was different from Variation 3 because a constant probability of 1.0 was used for the PORV opening, whereas in Variation 3, a probability distribution was used. Variation 6 was the same as Variation 1, except ac power recovery before vessel breach was not included. Stuck-open PORVs that occur after core uncovering were treated as leaks (S3 breaks) in Variation 7, rather than being fully open (S2 breaks) as in NUREG-1150.

Variations 8 through 10 examined the effect of in-vessel FCIs on the results of the base NUREG-1150 results, and on Variations 1 (no depressurization) and 5 (PORVs always opened). As noted in Section 3.2, the Surry APET used for NUREG-1150 considered alpha mode failures but not in-vessel FCIs that fail the vessel without generating a

Table 3.2 APET Variations

Case	Description
NUREG-1150	Results from NUREG-1150 evaluation
Variations:	
1	Probability of all depressurization mechanisms (stuck-open PORV, pump seal failure, SGTR, hot leg/surge line failure) set to zero
2	Probability of all depressurization mechanisms set to zero, except pump seal failures (at 250 gpm), which use NUREG-1150 values
3	Probability of all depressurization mechanisms set to zero, except stuck-open PORV, which use NUREG-1150 values
4	Probability of all depressurization mechanisms set to zero, except hot leg/ surge line failures, which use NUREG-1150 values
5	Probability of stuck-open PORV after core uncovering set to one (approximates intentional depressurization strategy)
6	Probability of all depressurization mechanisms set to zero, and probability of ac recovery between core uncovering and vessel breach set to zero
7	Stuck-open PORV treated as S3 break instead of S2 break (leaking PORV instead of fully open)
8	Same as NUREG-1150 except in-vessel FCIs are considered
9	Same as Variation 1 except in-vessel FCIs are considered
10	Same as Variation 5 except in-vessel FCIs are considered

containment-failing missile. Such FCIs could also lead to containment failure, but by overpressurization instead of missile penetration. This possibility was considered in the Grand Gulf APET[8] used for NUREG-1150, and the logic used in that APET is depicted in Figure 3.3. The Grand Gulf APET included the probabilities of an in-vessel FCI occurring, and of the FCI leading to various modes of vessel failure. The containment loads for cases with in-vessel FCIs were the same as for cases without FCIs. That is, there was no extra pressurization considered from the FCI.

Variations 8 through 10 of the Surry sensitivity evaluation used the mean values that had been used in the Grand Gulf APET for the probability of in-vessel FCIs occurring and the likelihood they would

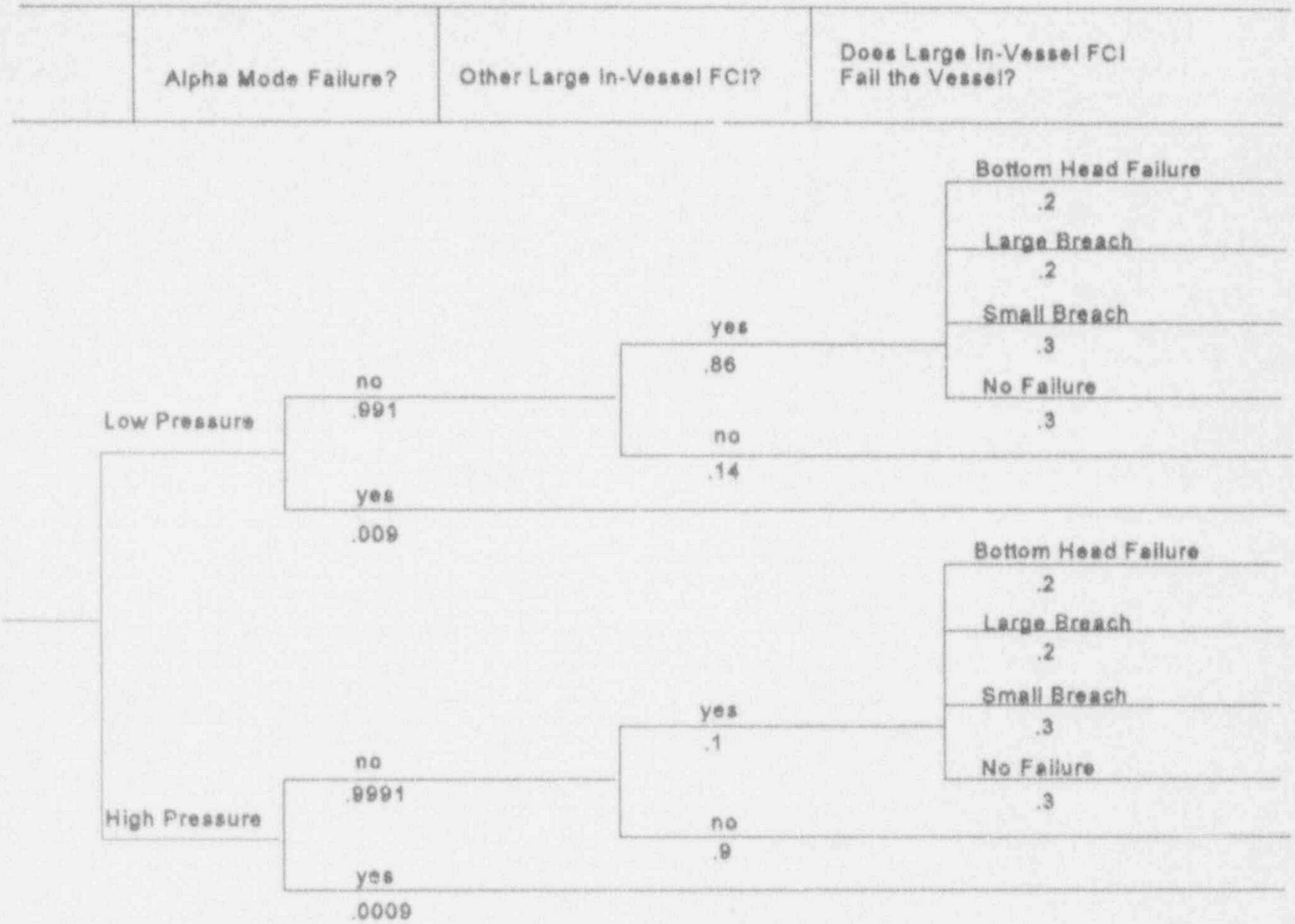

Figure 3.3 NUREG-1150, Grand Gulf logic for in-vessel FCIs

lead to vessel breach. However, the containment loading was treated differently. For the Surry sensitivity evaluations, in-vessel FCIs that failed the vessel were treated equivalently to cases with high pressure melt ejection (HPME) and with a large fraction of the core ejected (40% or more). Attempts to quantify the amount of material involved in the FCI were beyond the scope of this project. Because this study is examining sensitivities, the high fraction of material involvement was chosen.

The APET for each variation was evaluated using Sandia's EVNTRE[9] code. EVNTRE is designed to evaluate large event trees with complex interdependencies.

3.4 Approach for Updated Evaluation

The impact of intentional depressurization at Surry was examined using an updated evaluation that reflected new experiments and analyses that have been performed since NUREG-1150 was completed. This analysis was performed to determine the impact of intentional depressurization on the RCS pressure at vessel breach for Surry.

This program considered the impact of intentional depressurization for internal events; external events were not addressed. The dominant core damage sequences for NUREG-1150 were discussed in Section 2.2. LOCA sequences were not considered in this evaluation because the RCS pressure at vessel breach would be reduced by the break, such that intentional depressurization would not be necessary. In anticipated transient without scram (ATWS) and other transient sequences (other than station blackout), the operators are already directed to depressurize the RCS. Thus, only short- and long-term station blackout sequences were evaluated.

The evaluations were performed using an improved event tree analysis package, FOREST*, that has recently been developed at SNL. FOREST is a personal computer based system that uses preprocessors and postprocessors to simplify the input needed to run the EVNTRE code (which was used without the pre- and postprocessors for the first phase sensitivity studies). The input to the pre/post processors is in a more readable format[10] than the actual EVNTRE input, and is much easier to modify. FOREST runs as an add-on to the Microsoft® Excel spreadsheet program in conjunction with the @RISK add-on. In this environment, uncertainty distributions are sampled through @RISK, eliminating much of the awkwardness of developing and modifying uncertainty distributions that was present in the NUREG-1150 suite of codes.

Because of the ease of modifying event trees with the new input format, a revised APET was constructed, including only the questions from the NUREG-1150 APET that are needed to address the impact of intentional depressurization on the RCS pressure at vessel breach. The logic was modified to incorporate the probability distributions developed by INEL for hot leg/surge line failure and for the RCS pressure at vessel breach. In addition, the treatment of pump seal leakage, ac recovery, and in-vessel FCIs was updated. The specific modifications that were made to the APET are described in the following paragraphs.

Intentional Depressurization

A two-branch question was added to the event tree to address intentional depressurization. One branch is for the case without intentional depressurization of the RCS and the other branch is for the case with intentional depressurization after the core exit thermocouples reach 922 K (1200°F). The impact of intentional depressurization could then easily be determined by alternately setting the probability of the intentional depressurization branch to zero and one.

Pump Seal Leakage

The treatment of pump seal leakage was refined from that used in NUREG-1150. The accident frequency analysis in NUREG-1150[11] considered several different modes of failure, based on the results from an expert panel. Each failure mode had a different flow rate, and the flow rates were time-dependent as shown in Table 3.3. The outcomes were grouped into the cases listed in Table 3.4, with some cases having constant leak rates and other cases having leak rates that increased at a particular time in the transient. The accident progression analysis did not differentiate among the various leak rates. Instead, only cases with pump seal failure or no pump seal failure were distinguished. However, the SCDAP/RELAP5 calculations,[7] which were performed by INEL after NUREG-1150 was completed, indicated that a different response is

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* M. Fuentes, "FOREST: An Event Tree Analysis Program," Sandia National Laboratories, In preparation.

Table 3.3 Pump seal leakage as function of transient time[1]

Leak Rate (gpm/pump)[2]	1.5 hrs	2.5 hrs	3.5 hrs	4.5 hrs	5.5 hrs
21	.306	.290	.274	.274(.258)[3]	.274(.241)[3]
34	---	---	---	---	---
61, 74	.148	.0370	.0502	.0478(.0640)[3]	.0466(.0790)[3]
78	---	---	---	---	---
124	8.5E-3	5.0E-3	4.5E-3	3.7E-3	3.3E-3
142	---	---	---	---	---
172,175,182	3.5E-4	3.4E-4	3.2E-4	3.2E-4	3.2E-4
201,205	.001	0	0	0	0
250	.530	.660	.660	.660	.660
480	4.3E-3	4.3E-3	4.3E-3	4.3E-3	4.3E-3

[1] These values are the probabilities of being at a particular leak rate at a particular time
[2] Leak rate per pump is listed; all three pumps have leaks of this size
[3] Parentheses denote probabilities for cases without operator actions to depressurize or stuck-open PORV before core damage.

Table 3.4 NUREG-1150 pump seal leakage cases

Leakage Description (gpm per pump)
21 throughout the transient
61 throughout the transient
initially 61, then increasing to 250 at 2.5 hours
initially 124, then increasing to 250 at 2.5 hours
250 throughout the transient
480 throughout the transient

expected for the 480 and 250 gpm leak cases. Calculations at the other leak rates were not performed, but it was expected that similar responses would occur for the following groups: 21 gpm per pump or less, between 21 and 250 gpm per pump, or 480 gpm per pump. Because of this, the NUREG-1150 APET was modified such that these three cases were distinguished.

The NUREG-1150 expert panel on pump seal leakage addressed conditions that would be expected in the RCS before core damage. The panel did not address any additional leakage mechanisms that might result for RCS conditions after core damage. It was not possible to pursue this concern within the scope of this program, so any additional failures resulting from elevated temperatures after core damage were also neglected for this study.

Additional changes were made to the logic for pump seal leaks to capture the time-dependence shown in Table 3.3. First, note that the accident frequency analysis considers the failure of the pump seals before core damage, and the accident progression analysis considers the additional potential for failure of pump seals after core damage. Thus, care must be exercised to ensure consistency between the two PRA stages, and to ensure that failure probabilities are not double counted. In the NUREG-1150 accident progression analysis, the possibility of pump seal leakage was considered for cases without pump seal cooling, according to the logic shown in Figure 3.4. This logic was applied for all sequences, irrespective of the length of the transient before core damage. Note, however, in Table 3.3 that the probability of pump seal failure does not increase after the first few hours; that is, if the pump seals do not fail early in the transient, they are not likely to fail later. Thus, the probability of pump seal failure after core damage is dependent on the time between accident initiation and core damage. The APET was modified as described in the following paragraph to address this concern.

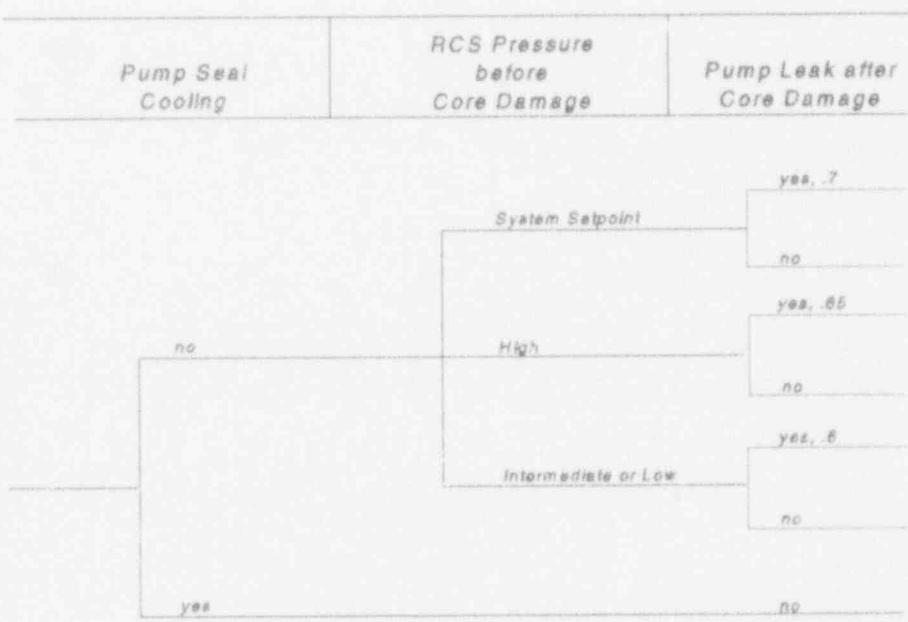

Figure 3.4 NUREG-1150 APET treatment of pump seal leakage

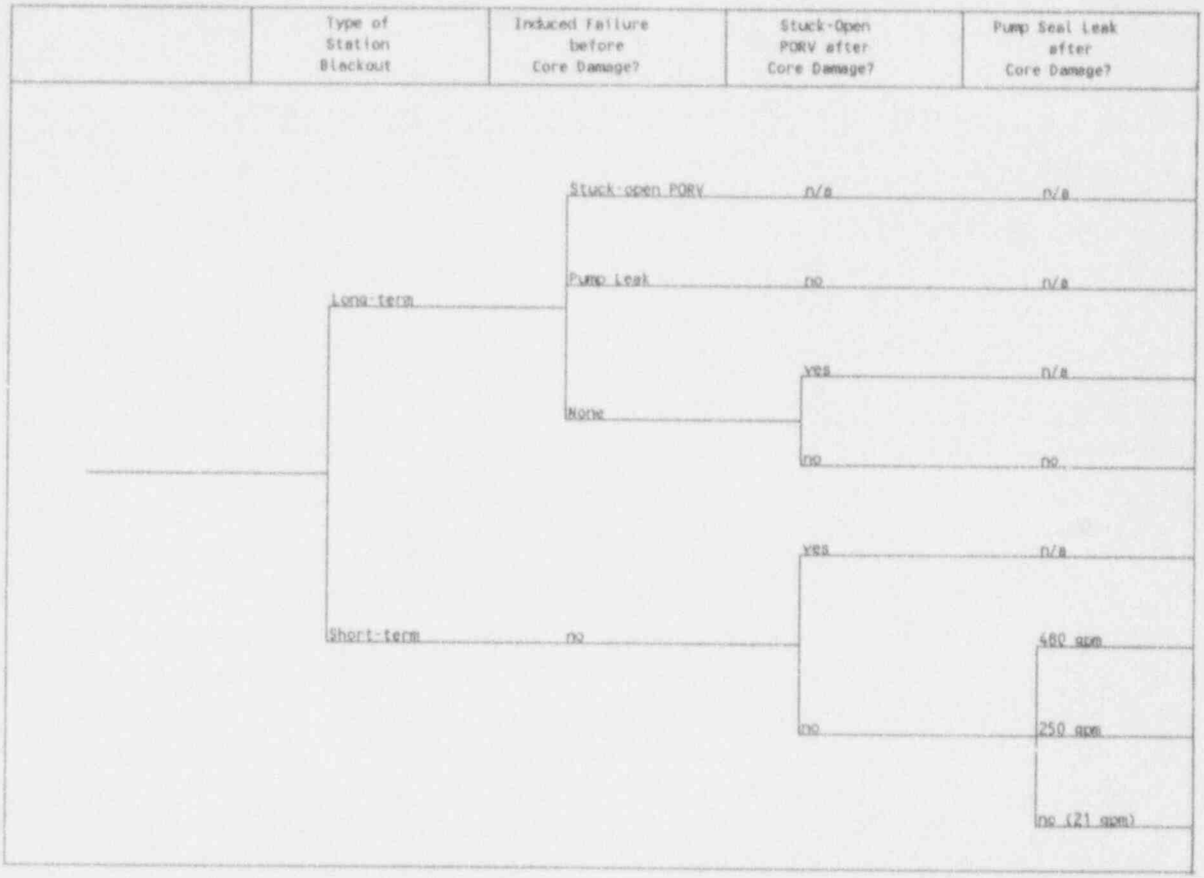

Figure 3.5 Logic for pump seal leakage for updated evaluation

The NUREG-1150 estimate for the time between transient initiation and core damage in long-term station blackouts is seven hours for cases with no pump seal leak and no stuck-open PORV. Thus, for these long-term station blackout scenarios, the probability of pump seal failure after core damage was set to zero for the updated evaluation. Pump seal failure after core damage is irrelevant for long-term station blackout sequences with a stuck-open PORV because the stuck-open PORV would dominate the transient response. For short-term station blackout sequences, the time between transient initiation and core damage is short enough that the probability of pump seal failure before core damage was found to be negligible in NUREG-1150. For these short-term station blackout cases, then, the values at 5.5 hours from Table 3.3 were used for pump seal leakage probabilities for the updated evaluation. The revised pump seal leakage treatment for the updated evaluation is illustrated in Figure 3.5.

Hot Leg/Surge Line Failure

The probability of hot leg or surge line failure before vessel breach was estimated in Reference 7 for four variations of a short-term station blackout sequence: stuck-open PORV after core uncovery, 250 gpm pump seal leak, 480 gpm pump seal leak, and no induced RCS failure. The probabilities reflect insights gained from SCDAP/RELAP5 calculations that have been performed since NUREG-1150 was completed. The development of the probabilities is described in Reference 7. The hot leg/surge line failure probabilities for the four cases are listed in Table 3.5.

Table 3.5 Probabilities of hot leg/surge line failure[1]

Scenario	Failure Probability
Without other induced RCS failures	.98
With 250 gpm pump seal leak (per pump)	.98
With 480 gpm pump seal leak (per pump)	0.0
With stuck-open PORV or intentional depressurization	1.0

[1] from Reference 7

It was not possible to perform SCDAP/RELAP5 calculati ns for all cases in the APET, so it was necessary to extend the SCDAP/RELAP5 results. The logic is illustrated by the event tree in Figure 3.6 (The final question in the figure will be discussed later in this section). For long-term station blackout sequences without induced failures (stuck-open PORV or pump seal leak) before core damage, the analogous short-term station blackout probabilities were used. This is an approximation because the lower decay heat in the long-term station blackout sequence could somewhat impact the timing of hot leg/surge line failure relative to lower head failure. For long-term station blackout sequences with either a stuck-open PORV or pump seal leak combined with steam generator depressurization before core damage, the RCS pressure would be less than 600 psia before core uncovery, reducing the potential for hot leg or surge line failure. The SCDAP/RELAP5 calculations indicated that the reduced RCS pressure accompanying the 480 gpm pump leak made hot leg or surge line failure unlikely. Thus, the probability of hot leg or surge line failure was conservatively set to zero for these cases. The long-term station blackout case with pump seal leakage but no steam generator depressurization would be above accumulator setpoint at core uncovery; if the RCS were intentionally depressurized, the RCS response would be expected to resemble the short-term station blackout case with intentional depressurization. For this case, then, the probability of hot leg/surge line failure was set to the same value as in the short-term station blackout (i.e., one).

RCS Pressure at Vessel Breach

INEL also estimated the probability of vessel breach occurring with the RCS at various pressures for cases with pump seal leaks, stuck-open PORVs, and hot leg/surge failure. The pressure ranges considered are: system setpoint pressure (2500 psia), high pressure (1000 - 2500 psia), intermediate pressure (200 - 1000 psia), and low pressure (< 200 psia). The probabilities are listed in Table 3.6, and the rationale for the probabilities is documented in Reference 7.

Core Damage Arrest

The handling of core damage arrest was modified from that in NUREG-1150 to incorporate more recent calculations that have been performed since NUREG-1150 was completed, particularly the INEL SCDAP/RELAP5 calculations. Also, the cases considered in the NUREG-1150 APET were expanded to include the impact of intentional depressurization and induced RCS failures after core uncovery. The modified approach is described in the following paragraphs.

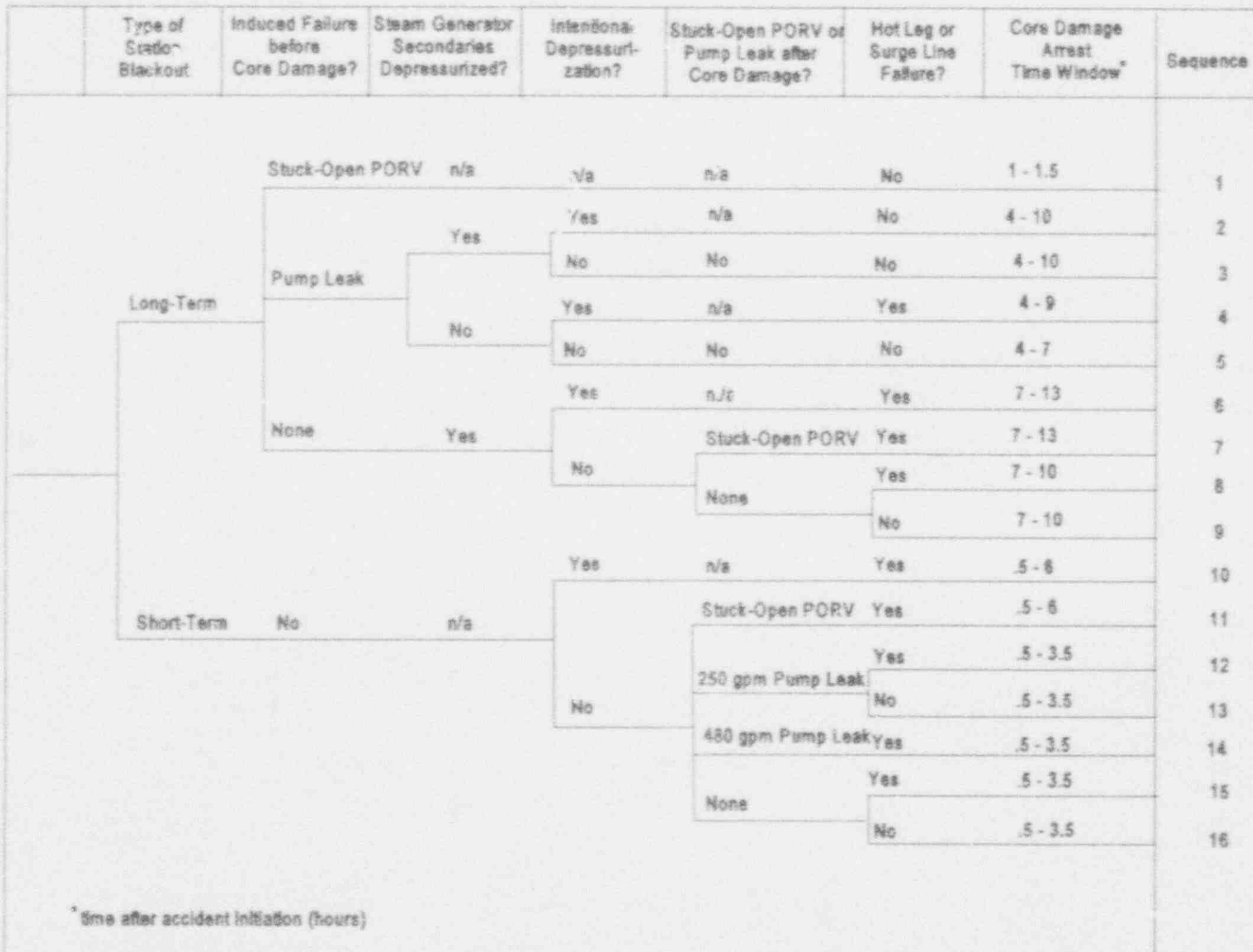

Figure 3.6 Time windows for ac recovery for various sequences

Table 3.6 Probabilities of RCS pressures at vessel breach

	Conditional Probability of Vessel Breach in Pressure Range		
Scenario	High (> 1000 psia)	Intermediate (200-1000 psia)	Low (< 200 psia)
With hot leg or surge line failure	0.	0.	1.
Without hot leg or surge line failure:			
250 gpm pump seal leak	0.21	0.75	0.04
480 gpm pump seal leak	0.13	0.40	0.47
Stuck-open PORV	0.	0.	1.

The methodology used to determine the core damage arrest probabilities in NUREG-1150 is described in Reference 5. It is based on analyses that indicate that ECC restoration would not prevent vessel breach if more than 60 mt of debris are on the lower head, that ECC restoration would likely prevent vessel breach if less than 30 mt of debris are on the lower head, and for debris masses between the two ranges, the recovery probability is highly uncertain. The time to reach each of these melt masses was then estimated. These time windows are illustrated in Figure 3.7. For NUREG-1150, if ac power was restored between t_1 and t_4, a high probability of arresting core damage was assigned; if ac power was restored between t_4 and t_5, the probability was set to about 0.5; and if ac power was restored between t_5 and t_7, the probability was set to zero.

Note that t_1 in Figure 3.7 is set in the accident frequency analysis, and is the time at which ECC must be restored to prevent core damage. At Surry, there is a 0.5 hour delay between ac recovery and ECC restoration, so ac power must be restored 0.5 hours before the critical point in the accident progression. For the Surry accident frequency analysis in NUREG-1150, core damage was conservatively defined to occur when the core uncovered and conservative (short) times to core uncovery were used in some cases.

A revised approach was used for estimating the probability of core damage arrest for the updated evaluation. Estimates of transient core melt masses are highly uncertain, but the available analyses indicate that there is a relatively short time between the onset of core melting and the accumulation of a relatively large molten pool. That is, there is a short time period between times t_3, t_5 and t_6 in Figure 3.7. Further, the effect of ECC restoration on accident progression is not well understood when there is substantial core damage. Thus, to estimate the probability of core damage arrest in this analysis, the probability of arresting core damage during the time period t_3 to t_5 was neglected. The time window between the beginning of the accident progression analysis (t_1 in Figure 3.7) and the formation of a molten pool (t_3 in Figure 3.7) was first estimated for each of the various cases using the t_1 value from the accident frequency analysis and using available code calculations (SCDAP/RELAP5, MELCOR, Source Term Code Package), which are summarized in

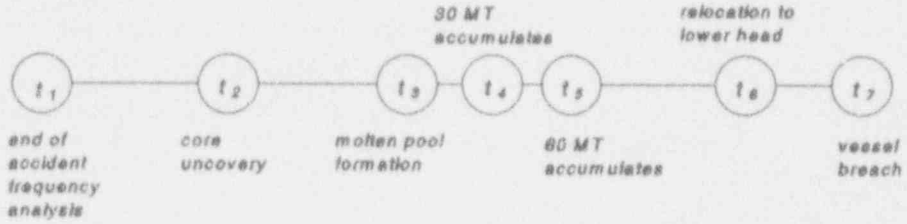

Figure 3.7 Time line for melt progression

Table 3.7, for t_3. The probability of restoring ac power during these time windows was then calculated using the NUREG-1150 methodology. If ac power was restored before molten pool formation for a particular case, the core damage was arrested. Otherwise, the case proceeded to vessel breach. The time windows considered for the various cases are shown in Figure 3.6; the justification is provided in the following paragraphs.

The time windows for the short-term station blackout cases were based on the SCDAP/RELAP5 calculations. Calculations 1 through 3 of Table 3.7 are short-term station blackout sequences with either no induced failures or with pump seal leaks. These calculations all had roughly the same time of molten pool formation. Also, hot leg or surge line failure was not predicted until after molten pool formation. Thus, the sequences in Figure 3.6 corresponding to these cases (12 through 16) all used the same value for t_3 (4 hours for molten pool formation, minus 0.5 hour delay for delivering ECC). The short-term station blackout cases with a stuck-open PORV were not differentiated from those with intentional depressurization. These cases (Sequences 10 and 11 from Figure 3.6) correspond to Calculation 4 from Table 3.7. However, a slightly shorter time window was actually used for these cases (0.5 to 6 hours, instead of 0.5 to 7 hours) because the recovery estimates were available from previous calculations, and the estimates would not be significantly different if recalculated for the interval of 0.5 to 7 hours.

SCDAP/RELAP5 calculations were not available for the long-term station blackout cases, so the time windows were conservatively estimated using the short-term station blackout time intervals and information from the MELCOR calculations. Sequences 5, 9 and 10 from Figure 3.6 are cases with either no induced failures, with only pump seal leakage, or with only hot leg/surge line failures. The short-term station blackout time window of 3 hours was used for these cases. Sequence 1 is a long-term station blackout with a stuck-open PORV even before core damage. Calculation 6 from Table 3.7 is a MELCOR calculation corresponding to this case, and it was used for the estimate of t_3. Sequences 2 and 3 from Figure 3.6 are long-term station blackouts with an early pump leak and with the steam generator secondaries depressurized. Calculation 5 from Table 3.7, a MELCOR calculation, was used for the estimate of t_3 for this case. Sequence 4 is a long-term station blackout with a pump leak before core damage and with intentional depressurization. Sequences 7 and 8 are similar, but without a pump leak. The expert panel did not predict significant pump seal leakage until after about 1.5 hours into the transients. Also, the time at which intentional depressurization would be initiated was not predicted to vary much in the SCDAP/RELAP5 calculations with and without pump leaks. Thus, approximately the same time interval was used for these cases as was used for the equivalent short-term station blackout cases (Sequences 10 and 11).

In-vessel FCIs

The treatment of in-vessel FCIs was modified for the updated evaluation, using the same logic described in Section 3.3 for the sensitivity evaluations. However, since the updated evaluation of the impact of intentional depressurization focuses on the probability of HPME, only the probability of FCIs was considered, not the resultant containment load.

Table 3.7 Summary of calculations giving sequence timing

Case	Sequence Description	Code	Core Uncovery Time (t_1 in Fig. 3.7) (hr)	Molten Pool Formation (t_3 in Fig 3.7) (hr)	Reference
1	Short-term SBO[1] with no induced failures	SCDAP/ RELAP5	2.4	4.6	Case 1 of Ref. 7
2	Short-term SBO with 250 gpm pump leak	SCDAP/ RELAP5	2.45	4.0	Case 3 of Ref. 7
3	Short-term SBO with 480 gpm pump leak	SCDAP/ RELAP5	2.36	3.9	Case 4 of Ref. 7
4	Short-term SBO with PORV opened when core exit temperature reached 922 K (1200 °F)	SCDAP/ RELAP5	2.5	7.6	Ref. 3
5	S_3 break with AFW[2] & SG[3] depressurized but no active ECC	MELCOR	9.8	10.4	Note 4
6	S_2 break with AFW & SG depressurized but no active ECC	MELCOR	1.7	2.2	Note 4
7	Long-term SBO with AFW lost at 5 hr	STCP	11.1	--	Ref. 5
8	Short-term SBO	STCP	1.6	--	Ref. 5
9	Long-term SBO with small pump leak at 1 hr & AFW lost at 5 hr	STCP	8.7	--	Ref. 5
10	Short-term SBO with small pump leak at transient initiation	STCP	1.5	--	Ref. 5

[1] SBO = Station Blackout Sequence
[2] AFW = Steam Generator Auxiliary Feedwater
[3] SG = Steam Generator
[4] Personal communication with L. N. Kmetyk, Sandia National Laboratories

4.0 Results

The results of the two phases of the program are presented in this Chapter. The results of the sensitivity evaluations (which were performed to gain insights into the key variables that would determine the impact of intentional depressurization) are discussed first, in Section 4.1. The results of the subsequent evaluation using updated probability distributions for the key variables are discussed in Section 4.2.

Graphical representations are used to show summary measures of the probability distributions for the various accident progression outcomes determined during the APET evaluations. A rectangular display is used in which the bottom and top of the rectangle represent the 5th and 95th percentiles, respectively. Also noted are the median and mean of the distribution.

4.1 Results of Sensitivity Evaluations

The results of the APET sensitivity evaluations that were described in Section 3.3 are reported in this section. Table 3.2, which defines the cases is repeated here as Table 4.1 for reader convenience.

Table 4.1 APET Variations

Case	Description
NUREG-1150	Results from NUREG-1150 evaluation
Variations:	
1	Probability of all depressurization mechanisms (stuck-open PORV, pump seal failure, SGTR, hot leg/surge line failure) set to zero
2	Probability of all depressurization mechanisms set to zero, except pump seal failures (at 250 gpm), which use NUREG-1150 values
3	Probability of all depressurization mechanisms set to zero, except stuck-open PORV, which use NUREG-1150 values
4	Probability of all depressurization mechanisms set to zero, except hot leg/surge line failures, which use NUREG-1150 values
5	Probability of stuck-open PORV after core uncovering set to one (approximates intentional depressurization strategy)
6	Probability of all depressurization mechanisms set to zero, and probability of ac recovery between core uncovering and vessel breach set to zero
7	Stuck-open PORV treated as S3 break instead of S2 break (leaking PORV instead of fully open)
8	Same as NUREG-1150 except in-vessel FCIs are considered
9	Same as Variation 1 except in-vessel FCIs are considered
10	Same as Variation 5 except in-vessel FCIs are considered

4.1.1 Results of Sensitivity Evaluations Without In-Vessel FCIs.

Figures 4.1 and 4.2 show the conditional probability of having vessel breach at low pressure for the NUREG-1150 evaluation and for Variations 1 through 7 (excluding Variations 1 and 6 which never breach the vessel at low pressure). The results for the long-term and short-term station blackout plant damage states are shown in Figures 4.1 and 4.2, respectively.

The NUREG-1150 evaluation had a relatively large conditional probability of having vessel breach at low pressure. The mean value is about 25% and 35% for the long- and short-term station blackouts, respectively. The first box in each of Figures 4.1 and 4.2 shows the distribution of the probability about these mean values.

The next three boxes in Figures 4.1 and 4.2 show the effect of pump seal failures, stuck-open PORVs, and hot leg/surge line failures. For both long- and short-term station blackouts, the cases with a hot leg/surge line failure have the largest probability of being at low pressure, followed by the cases with a stuck-open PORV, and finally, the cases with a pump seal failure. This result is not solely due to the relative sizes of the RCS ruptures; the conditional probability of failure is largest for the hot leg/surge line failure, then stuck-open PORVs, then pump seal failures.

The fifth box in each of Figures 4.1 and 4.2 shows the probability of having a low pressure vessel breach if the PORVs are always opened after core uncovering. The probability of having a low pressure vessel breach is quite high for this variation, but there is also a small probability of being at intermediate pressure to account for the possibility that the pressure rise following core slump does not decay completely before vessel breach.

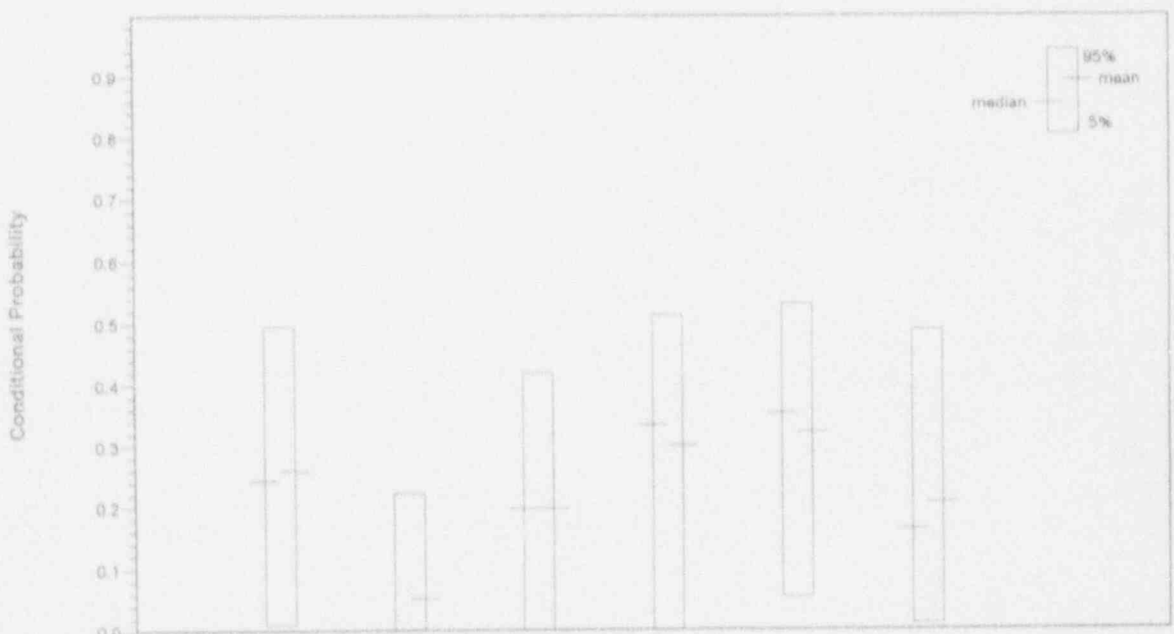

Figure 4.1 Conditional probability of vessel breach at low pressure for long-term blackouts

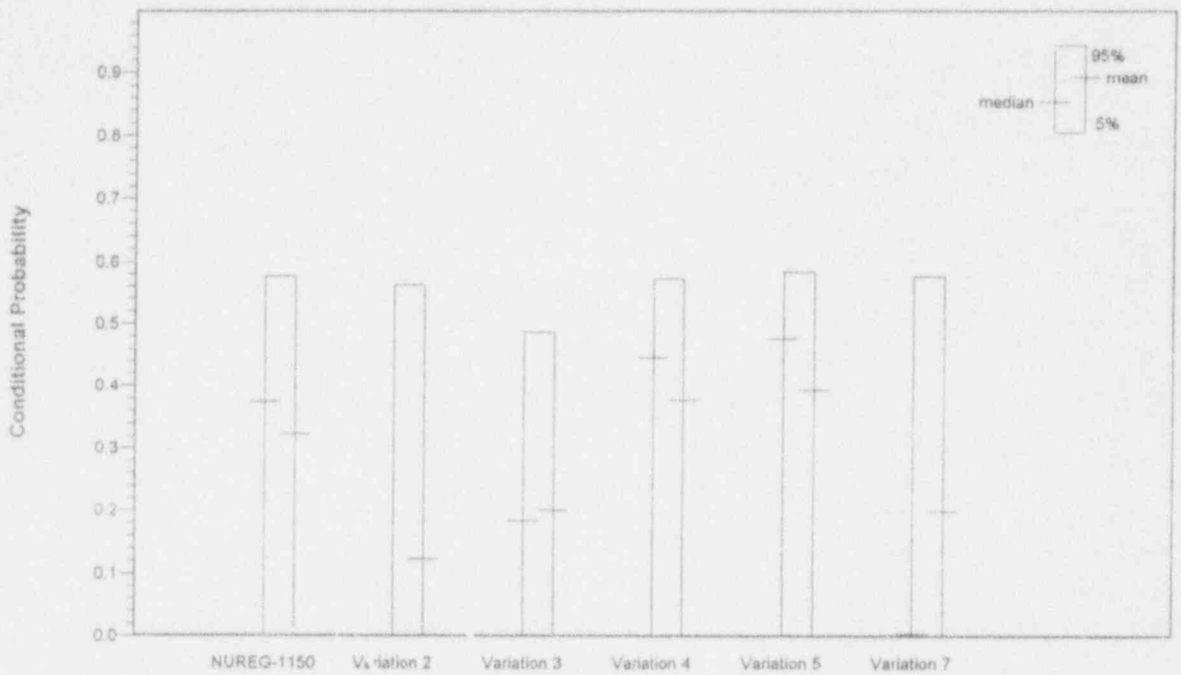

Figure 4.2 Conditional probability of vessel breach at low pressure for short-term blackouts

The final box reflects the impact of treating a stuck-open PORV as a leak instead of being treated as fully open. The probability distributions for having a low pressure vessel breach are shifted downward considerably from the NUREG-1150 results.

Figures 4.3 and 4.4 show the conditional probability that the loads generated near the time of vessel breach will fail the containment for the long- and short-term station blackout plant damage states, respectively. The figures show results for the base NUREG-1150 evaluation and Variations 1 through 7.

Note that even in the variations that are not depressurized from the system setpoint pressure, the conditional probability of containment failure at vessel breach is low. For a large fraction of the cases at high system pressure, core damage is arrested because ac power is recovered and ECC injection is restored, terminating the melt progression. The Surry APET includes a distribution for the likelihood of arresting core damage if ECC is recovered; it is not assumed that ECC recovery guarantees recovery from the accident. The likelihood of recovering the accident depends on the condition of the RCS boundary at core uncovery (stuck-open PORV, pump seal leak, cycling PORV), and on whether it is a short- or long-term station blackout. For all the cases in which ac power recovery was allowed, there was a high probability of arresting core damage, as shown in Figures 4.5 and 4.6.

Figures 4.7 through 4.10 show the combined effect of ac recovery and depressurization mechanisms on the relative likelihood of vessel breach occurring while at high/system setpoint and intermediate RPV

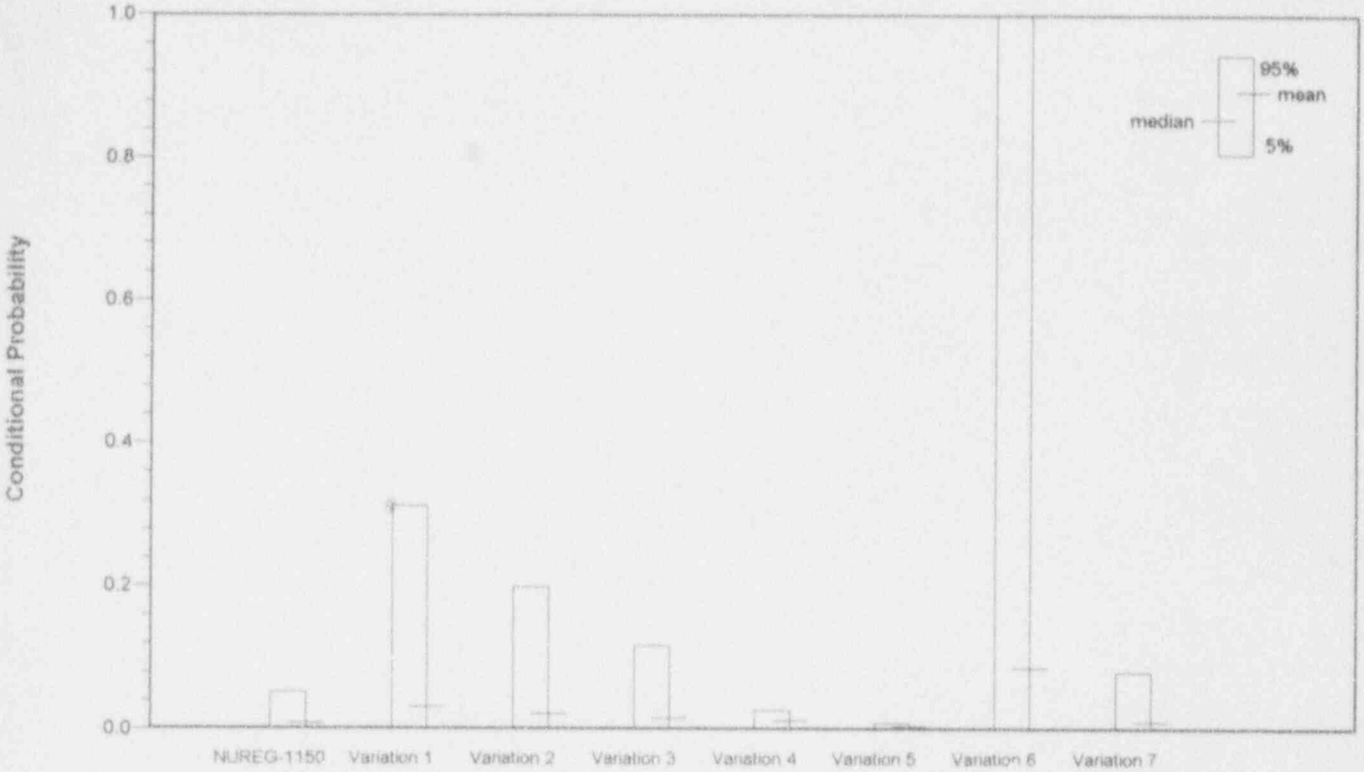

Figure 4.3 Conditional probability of containment failure at vessel breach for long-term blackouts

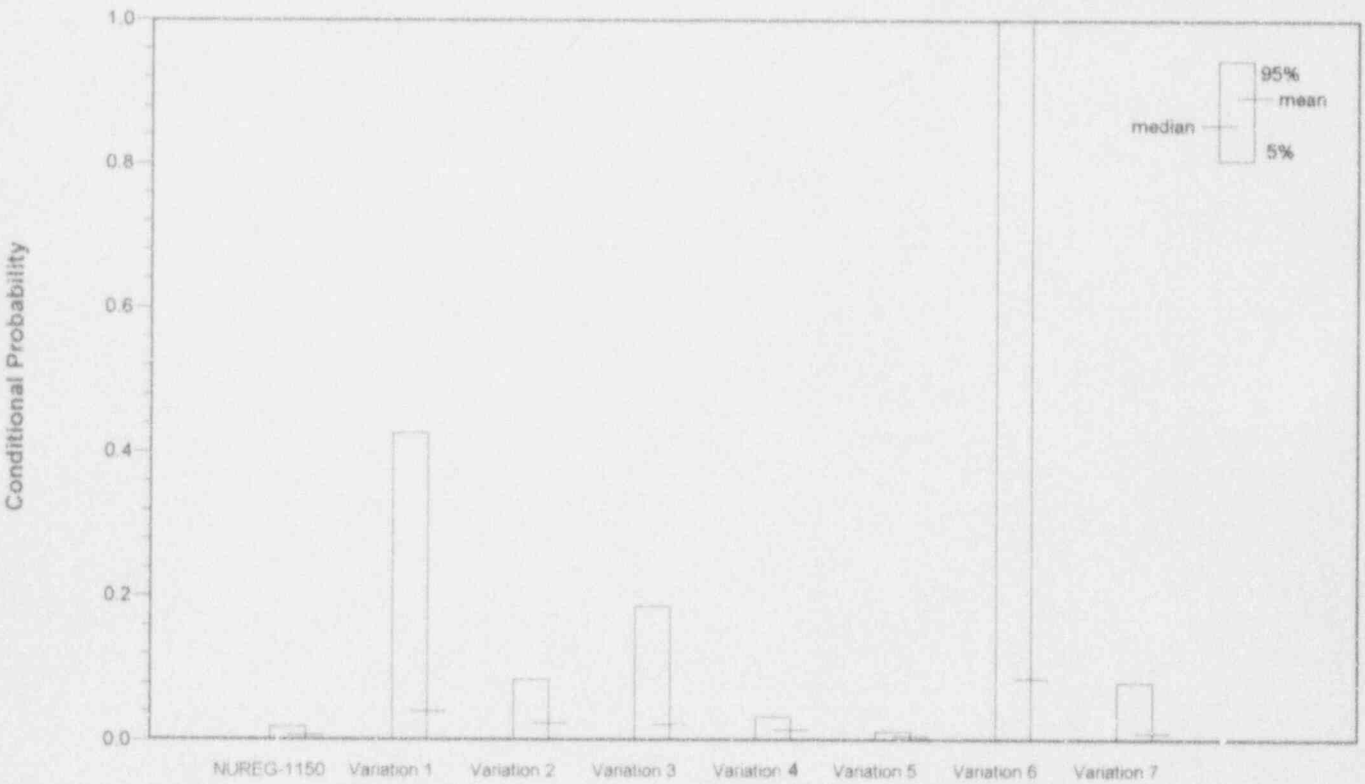

Figure 4.4 Conditional probability of containment failure at vessel breach for short-term blackouts

Figure 4.5 Conditional probability that core damage is arrested for long-term blackouts

Figure 4.6 Conditional probability that core damage is arrested for short-term blackouts

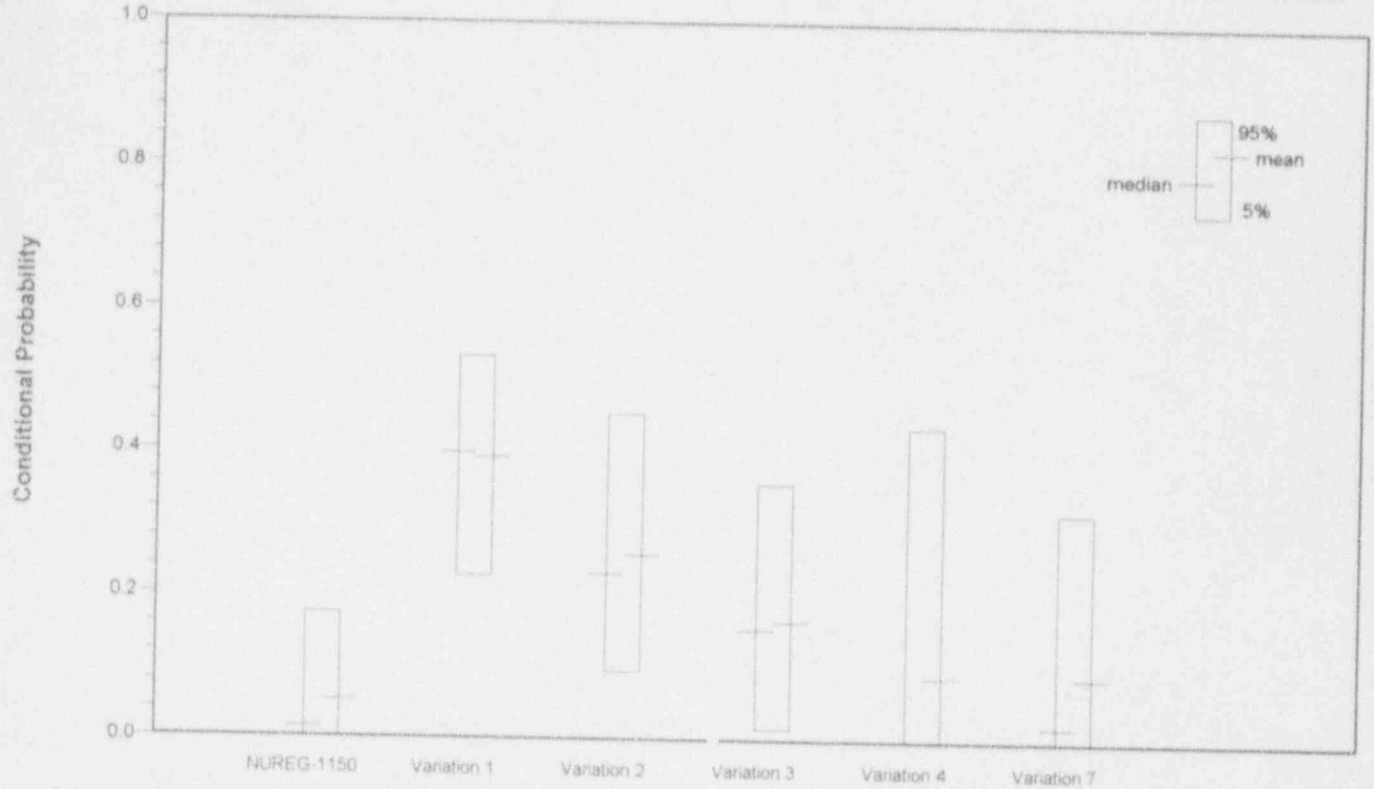

Figure 4.7 Conditional probability of vessel failure above 600 psia for long-term blackouts

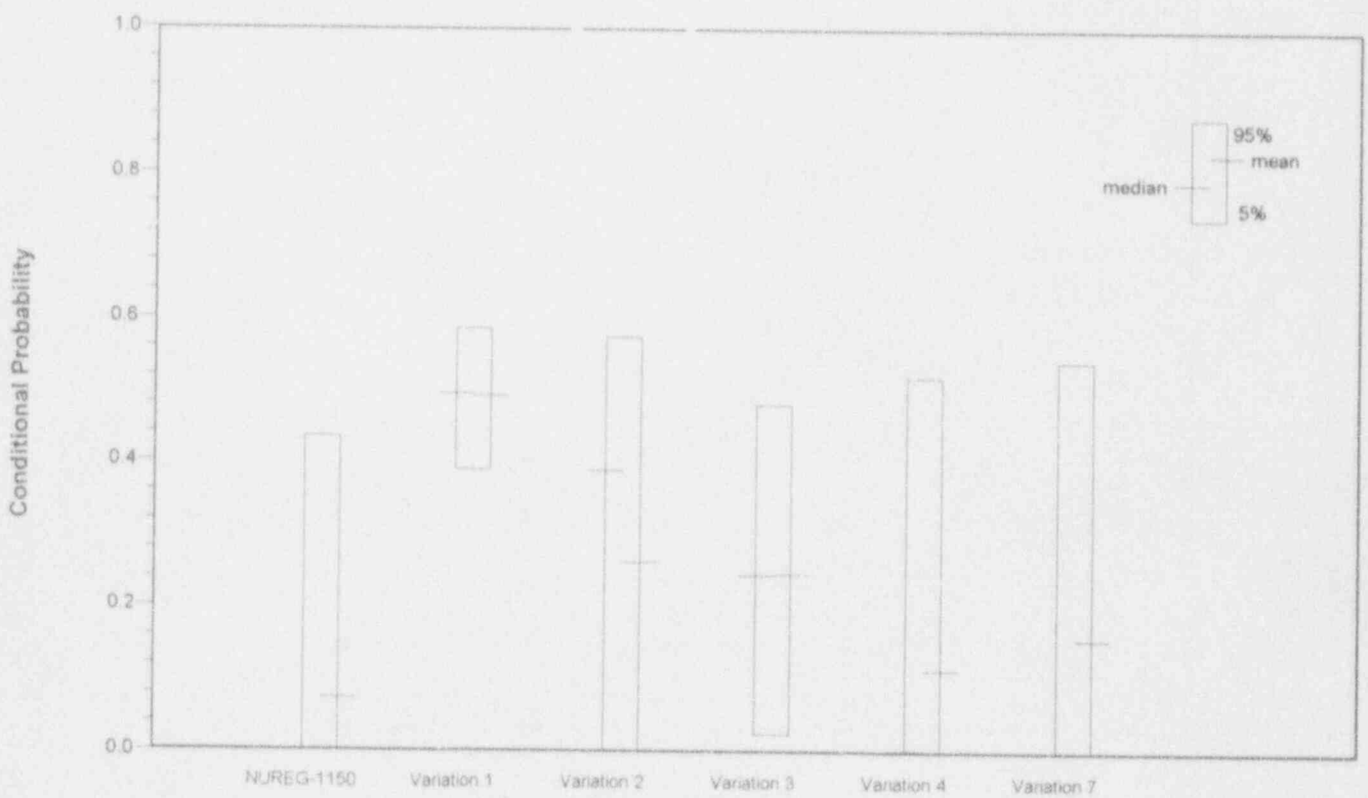

Figure 4.8 Conditional probability of vessel failure above 600 psia for short-term blackouts

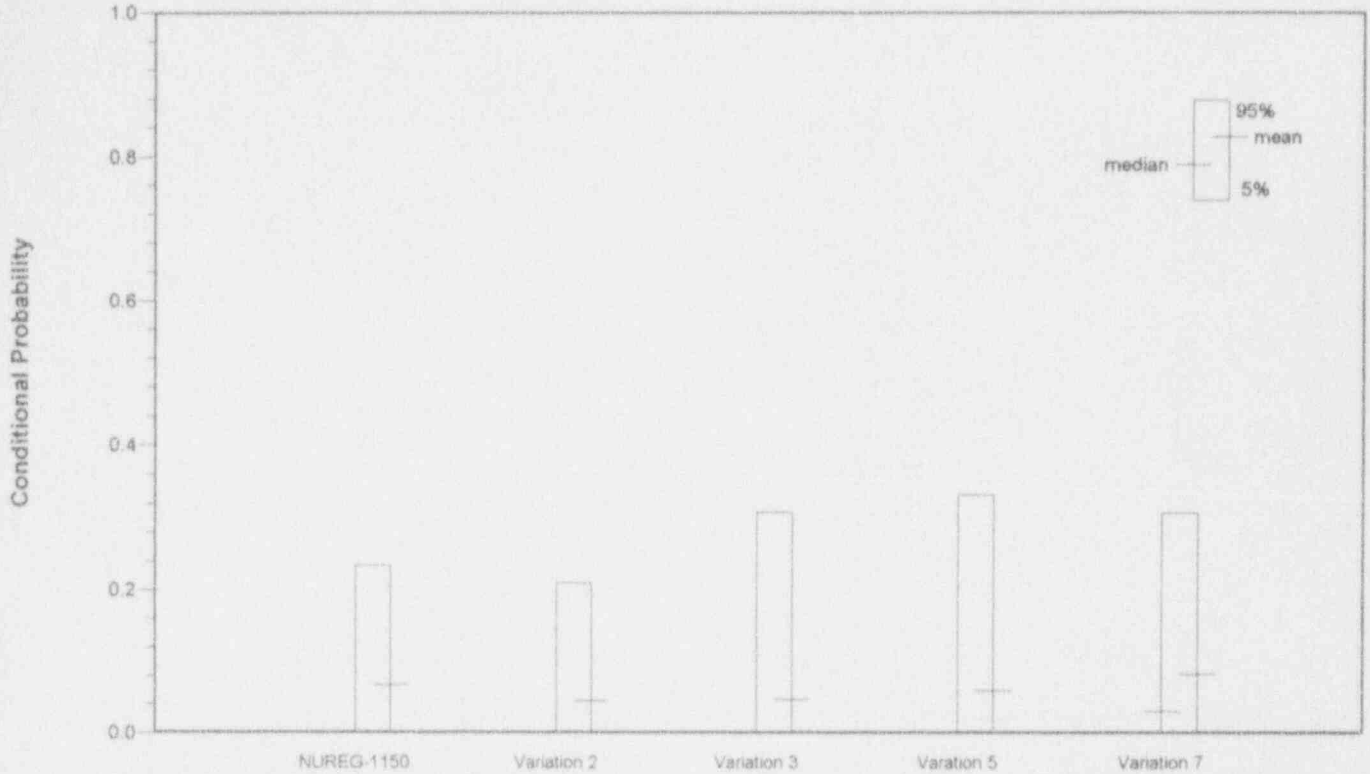

Figure 4.9 Conditional probability of RPV failure at intermediate pressure for long-term blackouts

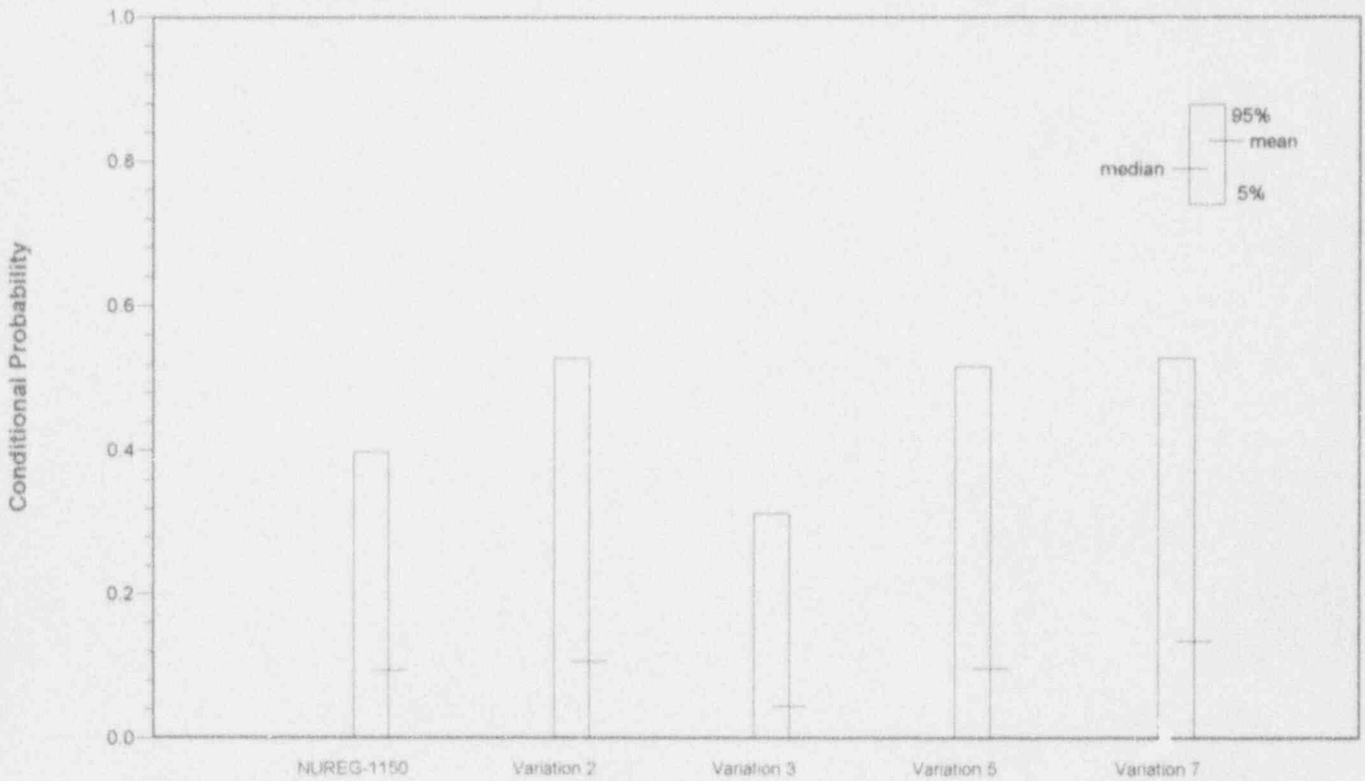

Figure 4.10 Conditional probability of RPV failure at intermediate pressure for short-term blackouts

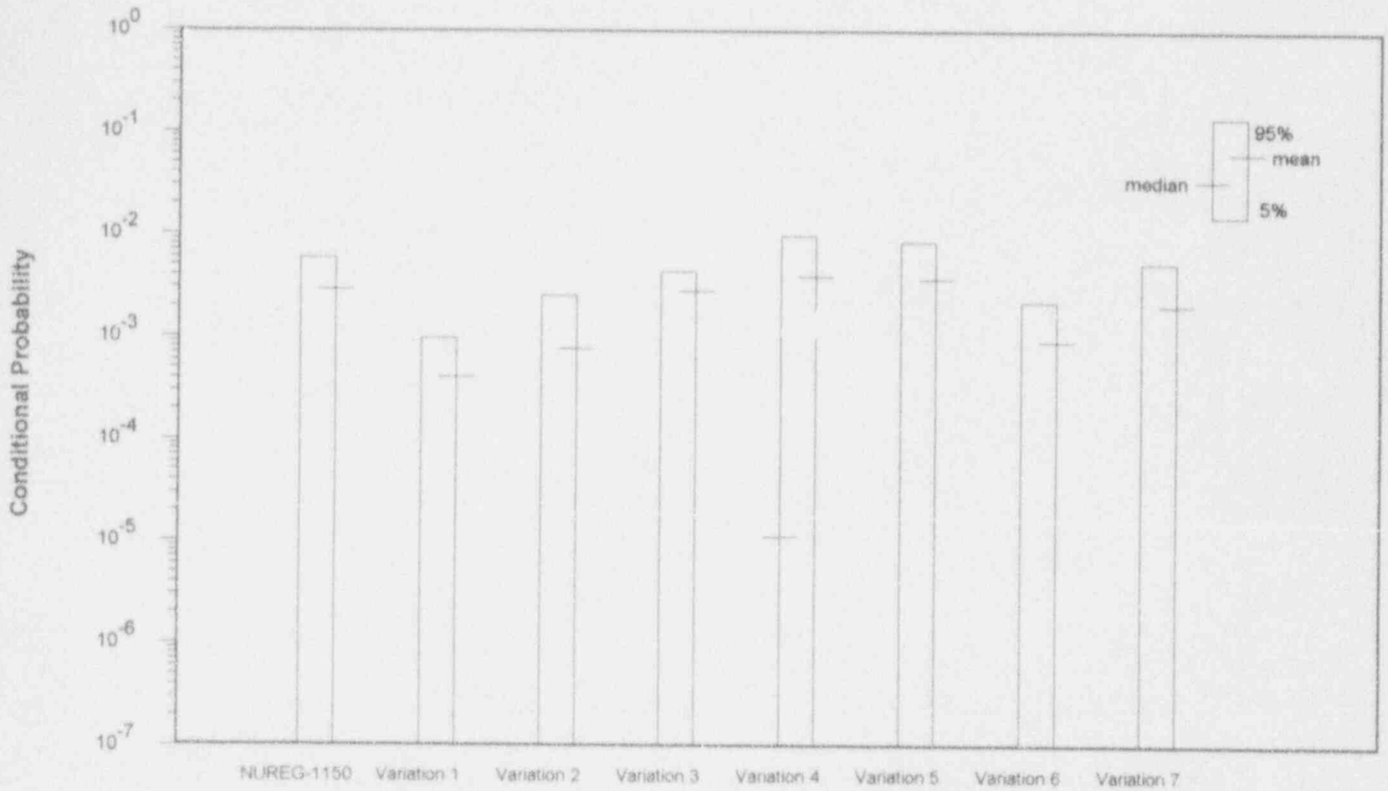

Figure 4.11 Conditional probability of alpha mode containment failure for long-term blackouts

pressures. These figures indicate that both ac recovery and failures of the RCS boundary before vessel breach had a large impact on the pressure at vessel breach.

It is also interesting to note that the variations with high probability of being at low pressure before vessel breach did not see a large enough increase in containment failures from alpha mode failures to significantly affect the results. The conditional probabilities of alpha mode containment failure for the long- and short-term station blackout groups are shown in Figures 4.11 and 4.12, respectively. The conditional probability of this failure mode increases for the variations with higher likelihood of being at low pressure, but the conditional probability is still relatively low. More of an impact was observed when non-alpha in-vessel FCIs were included, as will be discussed below.

As a final point of comparison, Figures 4.13 and 4.14 show the conditional probability of being at a state with no containment failure at the end of the evaluation (24 hours). The probability of maintaining an intact containment is quite high for the base cases and the 7 variations. The assumptions regarding depressurization did not have a large impact on the results, indicating that depressurization options do not affect post-vessel breach phenomena excessively, as considered in the Surry APET used for NUREG-1150. It is interesting that eliminating depressurization increased the mean probability of containment survival. This is because the debris dispersal during DCH lowered the extent of core-concrete interactions, which lessened the containment challenge from basemat meltthrough. Additional plots, showing the full probability distributions for accident progression outcomes are included in Appendix A.

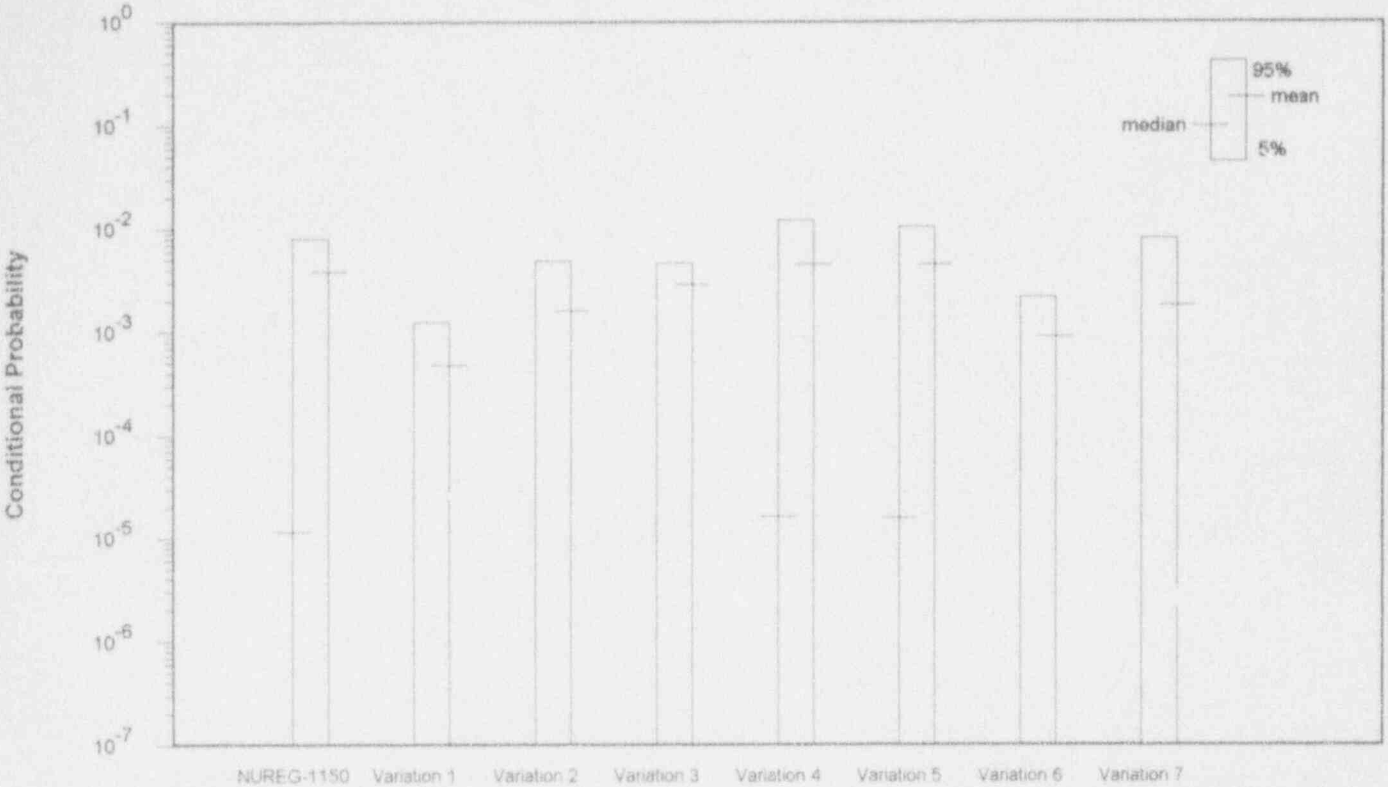

Figure 4.12 Conditional probability of alpha mode containment failure for short-term blackouts

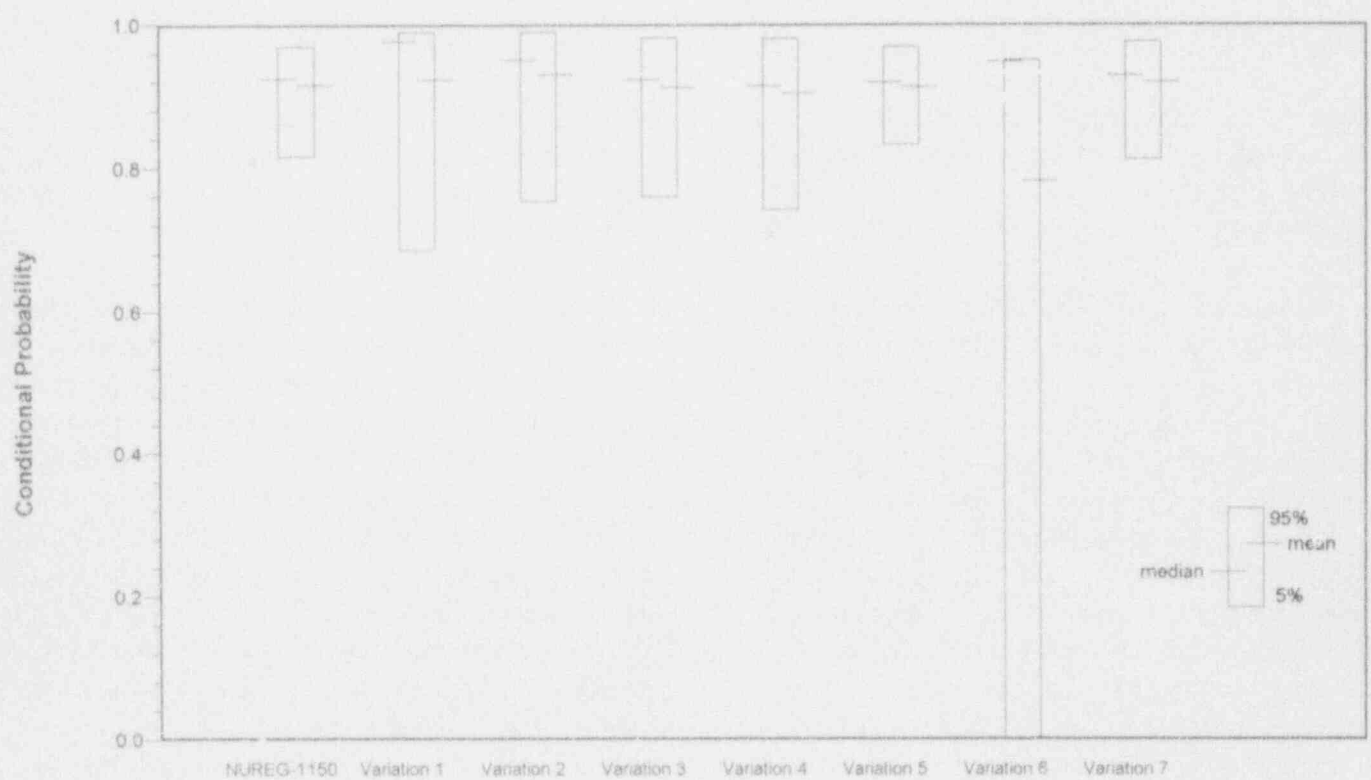

Figure 4.13 Conditional probability of having intact containment at 24 hours for long-term blackouts

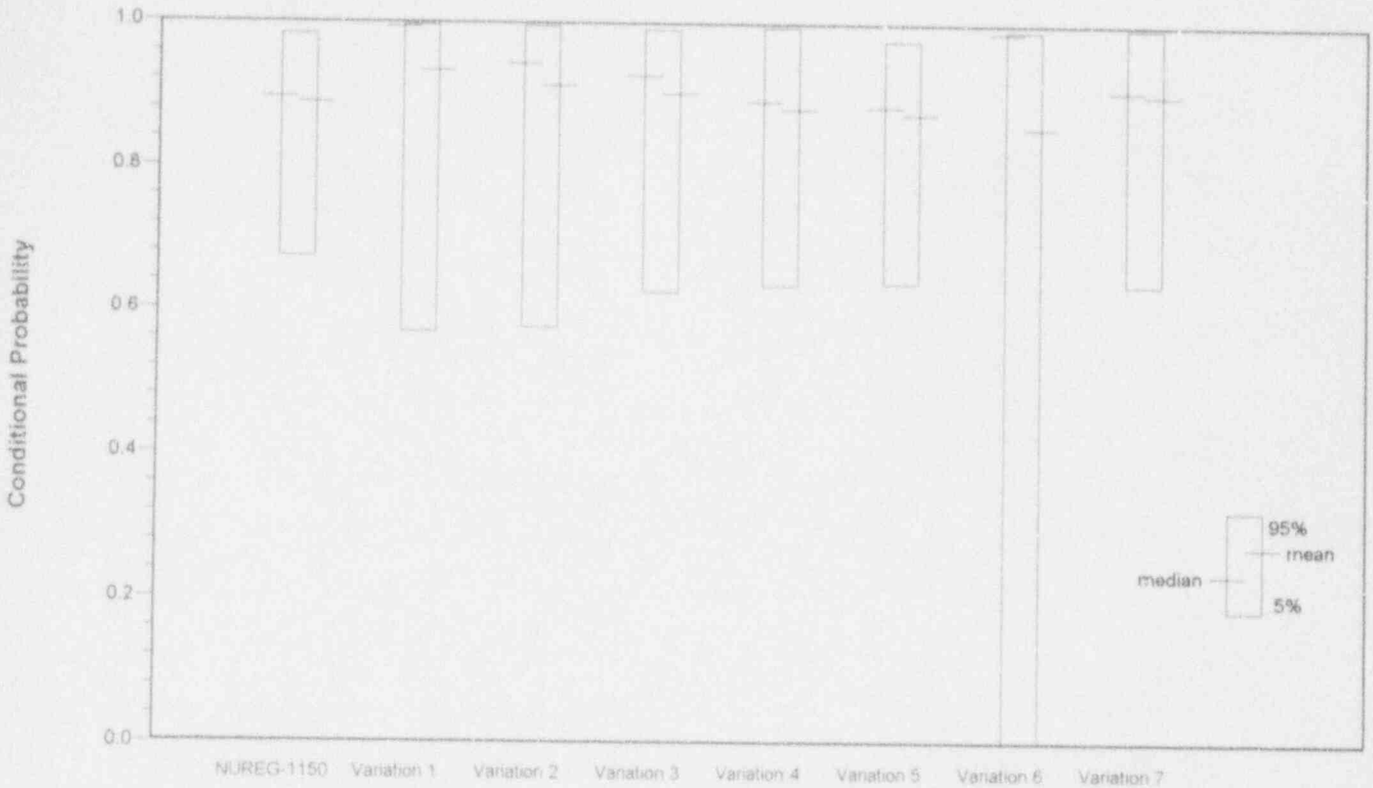

Figure 4.14 Conditional probability of having intact containment at 24 hours for short-term blackouts

4.1.2 Results of Sensitivity Evaluations Including In-Vessel FCIs

Figures 4.15 and 4.16 show the probability that an in-vessel FCIs occurs that breaches the RPV, for the long- and short-term station blackout plant damage states, respectively. Alpha mode failures are not included in these distributions. Results for the variations that add in-vessel FCIs to the base NUREG-1150 results, the variation without depressurization, and the variation with PORVs always opened are shown. The probabilities of vessel failure from an in-vessel FCI are much higher for cases with low RPV pressure than for cases with high RPV pressure.

Figures 4.17 and 4.18 show the effect of including in-vessel FCIs on the conditional probability of early containment failure, for the long- and short-term station blackout plant damage states, respectively. Results for the base NUREG-1150 results, the variation without depressurization, and the variation with PORVs always opened are shown with and without in-vessel FCIs considered. As shown in these figures, including in-vessel FCIs had a large impact on the results. The distributions for early containment failure probability were higher for all cases when FCIs were included. The effect of FCIs was large enough to negate the beneficial reduction in DCH loads from depressurization as can be seen by comparing the distributions for Variations 8, 9, and 10. That is, when FCIs were included in the evaluation, the probability of early containment failure was slightly higher for the cases with PORVs always opened than for the cases with no depressurization.

Figure 4.15 Conditional probability of vessel failure from FCI for long-term blackouts

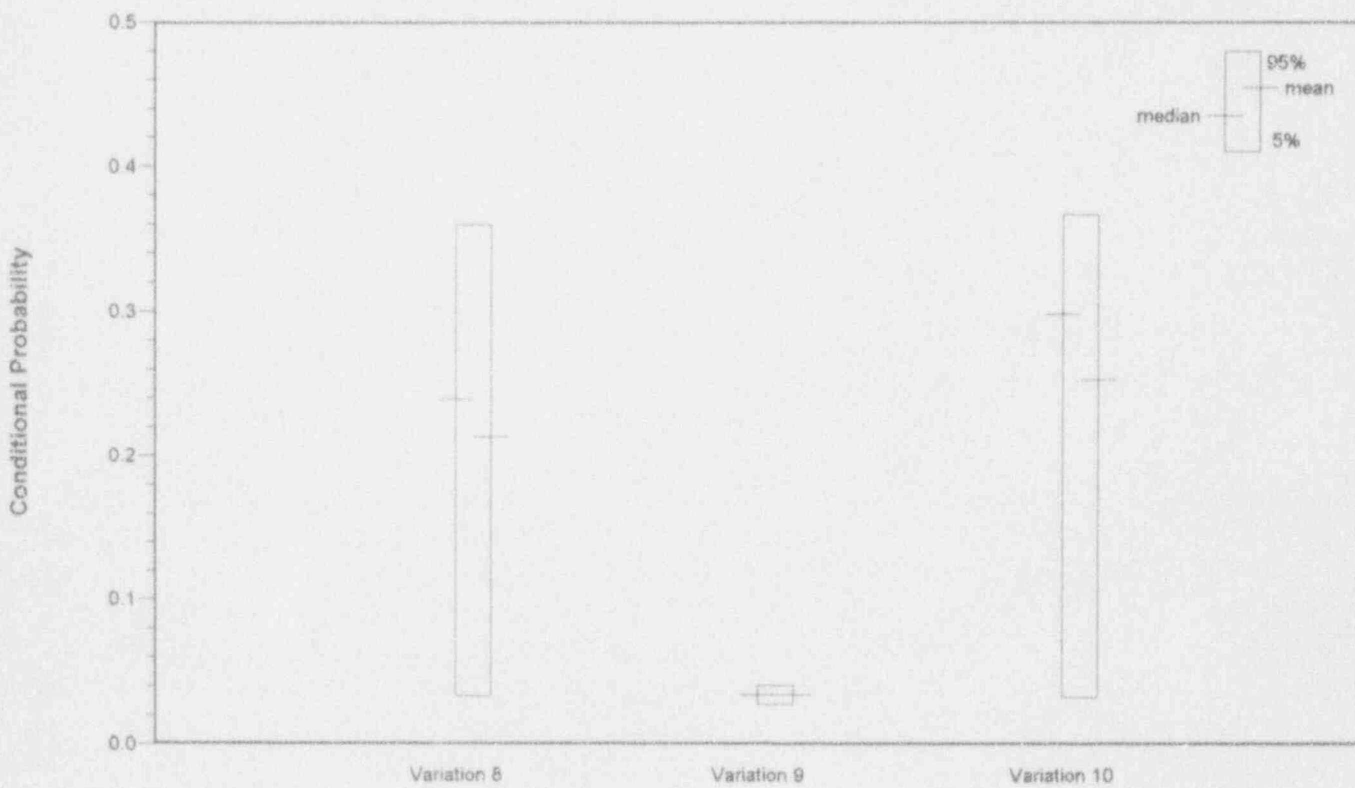

Figure 4.16 Conditional probability of vessel failure from FCI for short-term blackouts

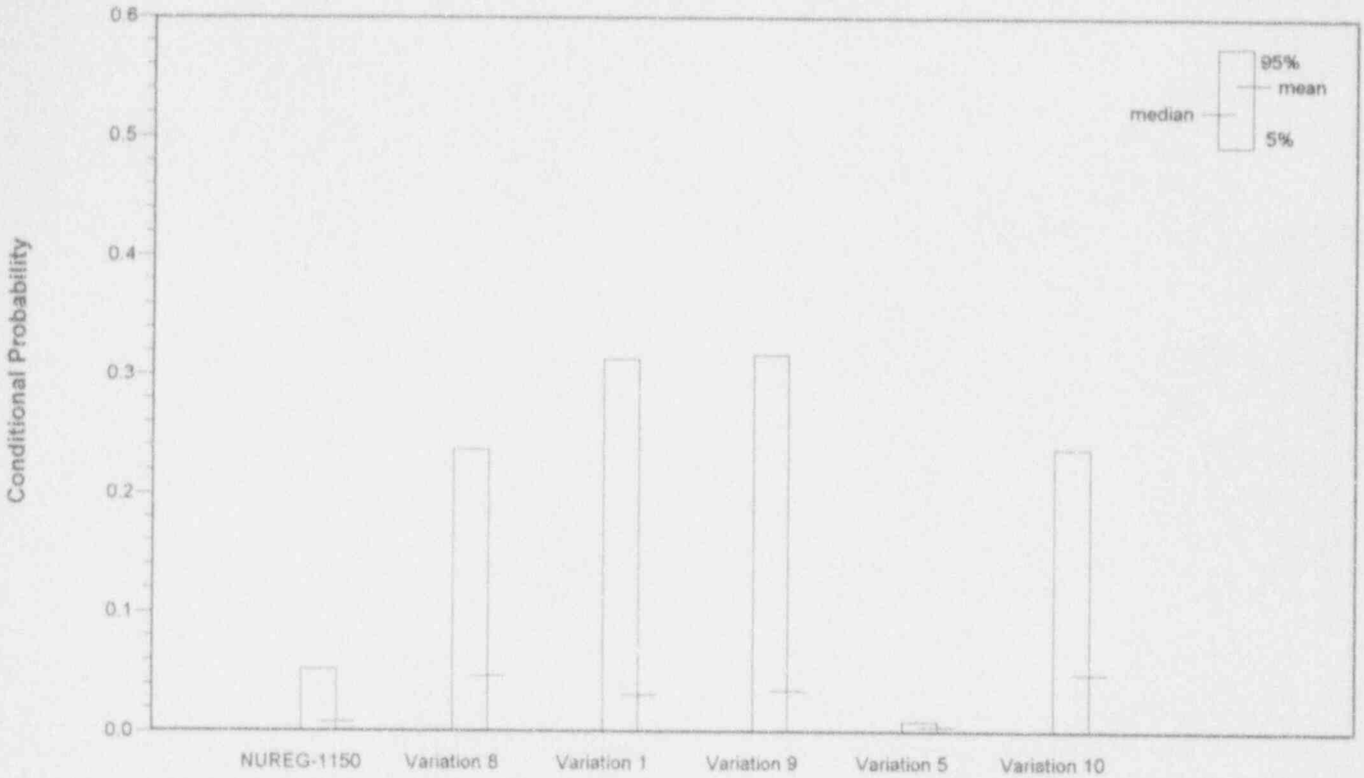

Figure 4.17 Conditional probability of early containment failure for long-term blackouts with in-vessel FCIs included

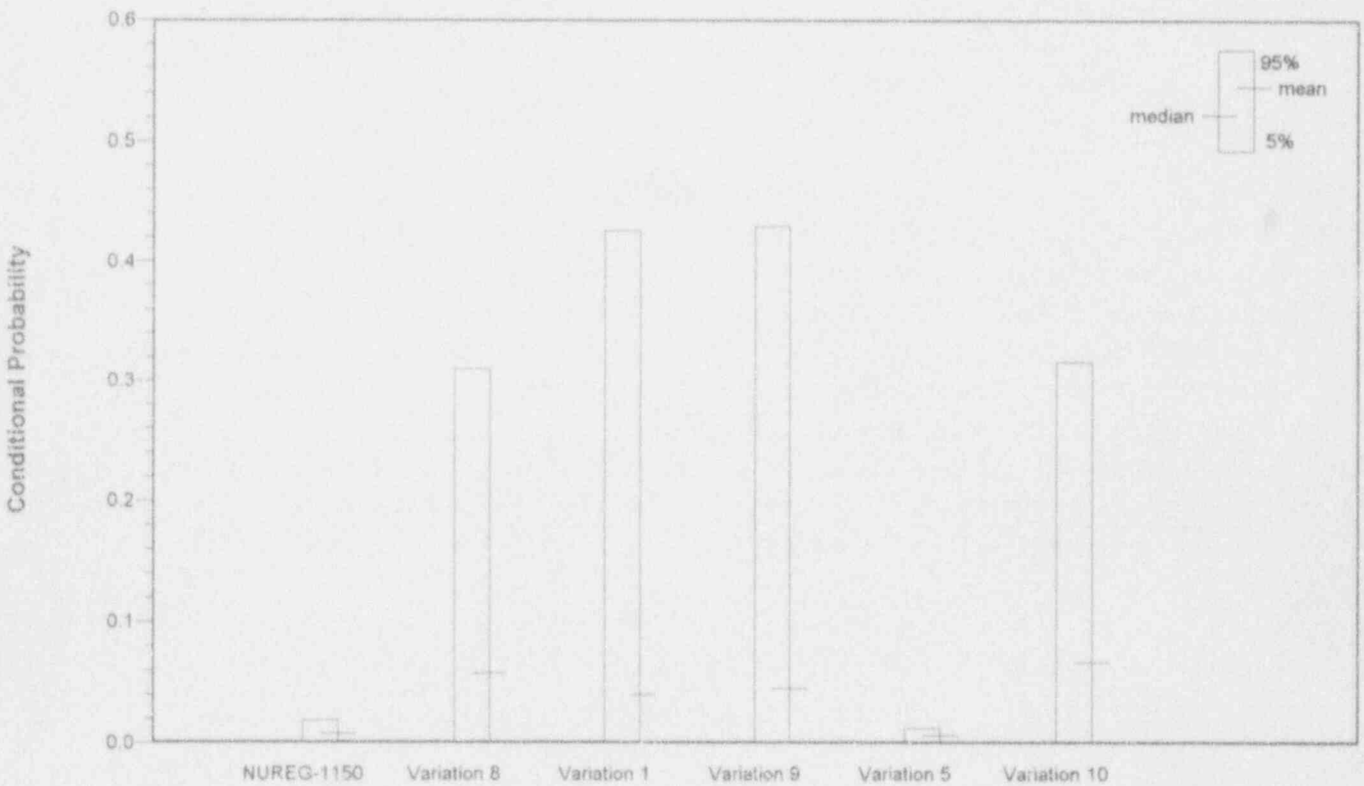

Figure 4.18 Conditional probability of early containment failure for long-term blackouts with in-vessel FCIs included

Figure 4.19 Conditional probability of avoiding containment failure for 24 hours in long-term blackouts with in-vessel FCIs included

The probability of the accident proceeding for 24 hours without containment failure for each of the FCI variations are shown in Figures 4.19 and 4.20, for the long- and short-term station blackout plant damage states, respectively. Results for the base NUREG-1150 results, the variation without depressurization, and the variation with PORVs always opened are shown with and without in-vessel FCIs considered. Including in-vessel FCIs decreases the probability of maintaining an intact containment for both the long- and short-term station blackout groups because of the increased probability of containment failure at vessel breach. As in the cases without FCIs, the probability of maintaining an intact containment for 24 hours was highest when depressurization mechanisms were eliminated.

Additional plots, showing the full probability distributions for accident progression outcomes are included in Appendix A.

Figure 4.20 Conditional probability of avoiding containment failure for 24 hours in short-term blackouts with in-vessel FCIs included

4.2 Updated Evaluation

The results of the updated evaluations that were described in Section 3.4 are reported in this section.

4.2.1 Long-Term Station Blackout.

The impact of the intentional depressurization strategy on the probability of avoiding vessel breach or of vessel breach occurring with the RCS at various pressures is shown in Figure 4.21 for the long-term station blackout scenario. The pressure ranges are defined as follows:

system setpoint -	2500 psia
high -	1000 - 2000 psia
intermediate -	200 - 1000 psia
low -	< 200 psia.

Note that the division between the intermediate and high pressure ranges (1000 psia) is slightly different from NUREG-1150 (600 psia). The ranges used here are the same as those used by INEL in NUREG/CR-5949, and are fully adequate for our purposes. NUREG-1150 divided the high and intermediate ranges at the accumulator setpoint. For our purposes the more uniform division is more meaningful.

The intentional depressurization strategy slightly increased the probability of avoiding vessel breach, and eliminated the potential for being at intermediate or higher pressures at vessel breach. The probability of having vessel breach with the RCS at the system pressure is low even without intentional depressurization because of the induced

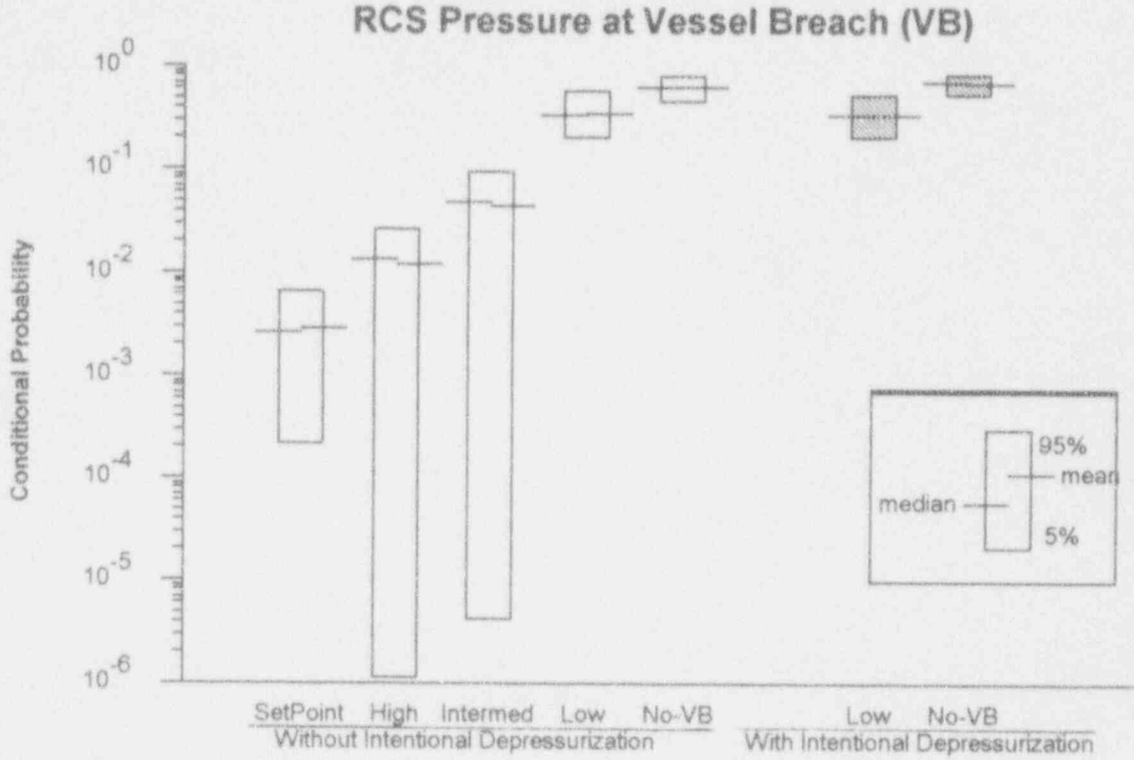

Figure 4.21 Impact of intentional depressurization on HPME for long-term station blackout

RCS failures (stuck-open PORV, pump seal leak, hot leg/surge line failure), but there is a small probability of being in the high or intermediate pressure ranges. The intentional depressurization strategy results brings the pressure down to the low pressure range, with no probability of being in the higher pressure ranges.

Figure 4.22 shows the probability of in-vessel FCIs with various consequences for scenarios with and without intentional depressurization. The probability of FCIs was slightly lower for the intentional depressurization strategy because there was a higher probability of arresting core damage. Intentional depressurization delays the time to core melt because depressurization allows the accumulators to inject, cooling the core. This delay gives more time to restore ac power and initiate ECC while the core is still in a recoverable geometry. The increase was small because the probability of having an elevated pressure, which impacts the probability of FCIs, is low both with and without intentional depressurization.

The full probability distributions for having vessel breach in the various pressure ranges are included in Appendix B for scenarios with and without intentional depressurization.

4.2.2 Short-Term Station Blackout.

The impact of the intentional depressurization strategy on the probability of avoiding vessel breach or of vessel breach occurring with the RCS at various pressures is shown in Figure 4.23 for the short-term station blackout scenario. The pressure ranges are the same as those defined in Section 4.2.1. Similarly to the long-term station blackout, intentional depressurization increased the probability of avoiding vessel breach because of the higher probability of arresting core damage.

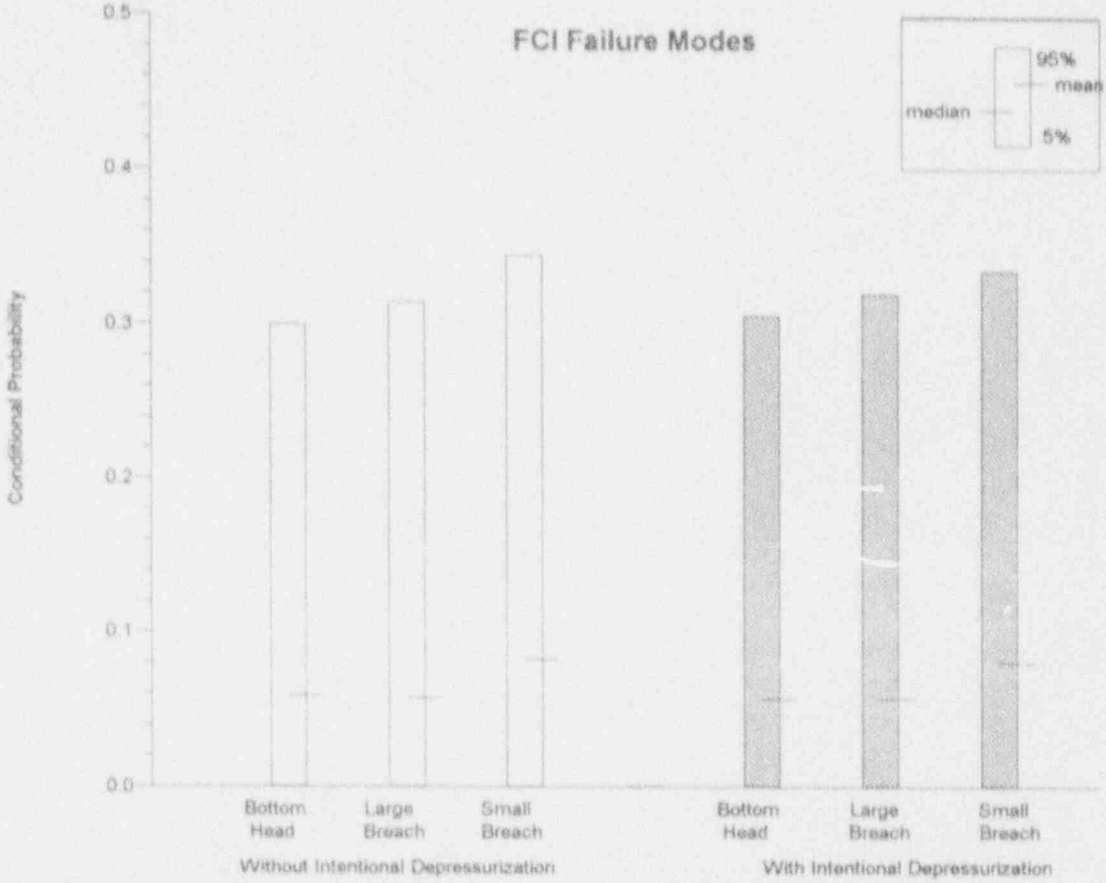

Figure 4.22 Impact of intentional depressurization on in-vessel FCIs for long-term station blackout

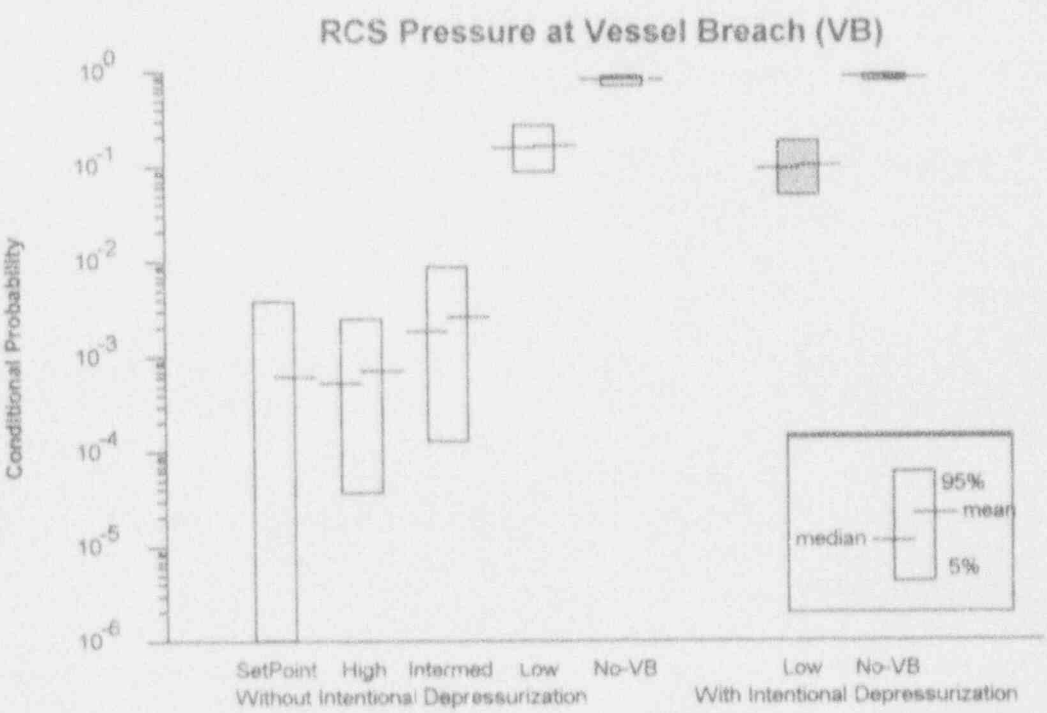

Figure 4.23 Impact of intentional depressurization on HPME for short-term station blackout

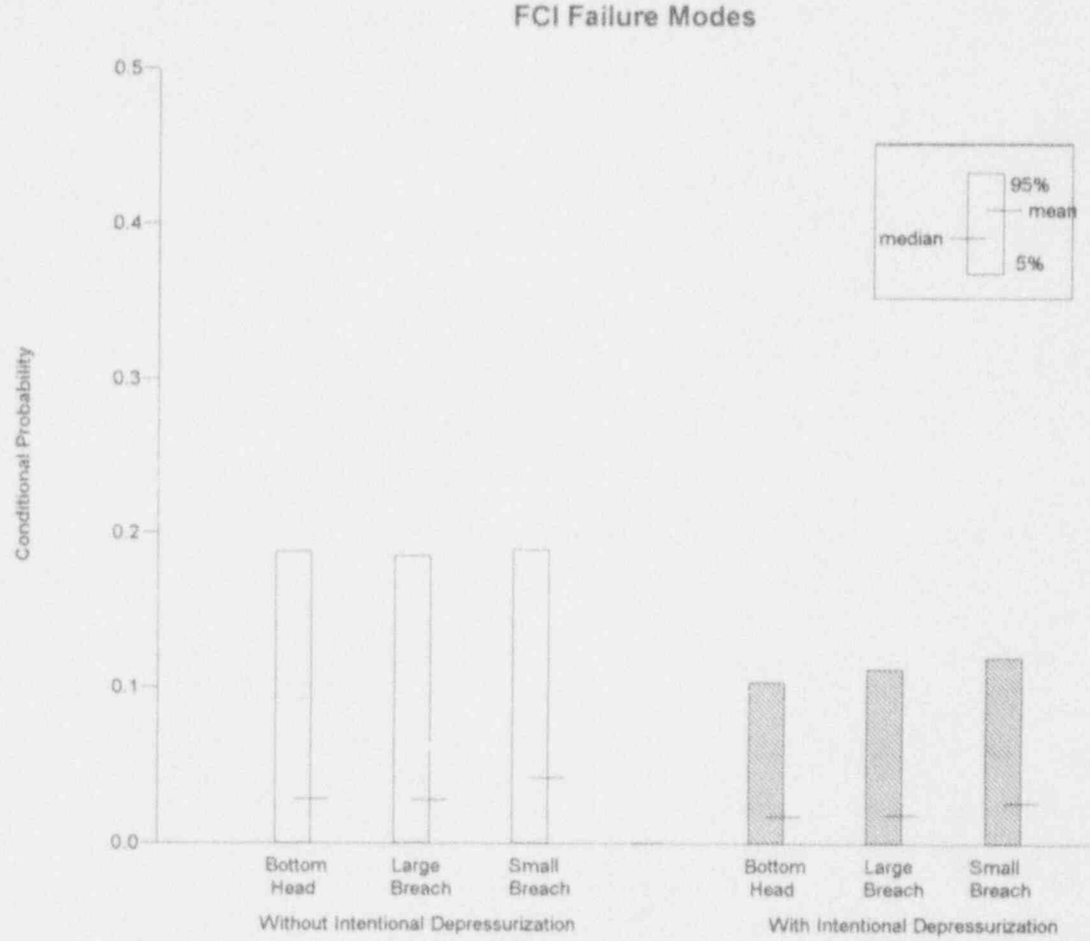

Figure 4.24 Impact of intentional depressurization on in-vessel FCIs for short-term station blackout

As seen for the long-term station blackout, intentional depressurization eliminated the possibility of being at intermediate or higher pressures at vessel breach. The probability of having vessel breach with the RCS pressure above the low pressure range is low even without intentional depressurization because of the induced RCS failures (stuck-open PORV, pump seal leak, hot leg/surge line failure).

Figure 4.24 shows the probability of in-vessel FCIs with various consequences for scenarios with and without intentional depressurization. The probability of FCIs was reduced slightly with intentional depressurization because of the higher probability of core damage arrest.

The full probability distributions for having vessel breach in the various pressure ranges are included in Appendix B for scenarios with and without intentional depressurization.

5.0 Conclusions

The sensitivity studies indicated that with the NUREG-1150 treatment of DCH loads, intentional depressurization would not significantly reduce the containment failure probability at Surry because a large fraction of the station blackout sequences have inadvertent failures in the reactor system boundary that lead to depressurization, a large fraction of the core melt sequences are terminated before vessel breach because of ac power recovery, and those sequences that continue to vessel breach at high pressure do not usually result in sufficient loads at vessel breach to fail containment. In fact, when in-vessel FCIs (non-alpha) are included, intentional depressurization might actually increase the containment failure probability slightly because FCIs are more likely at lower pressures. However, the updated evaluation indicated that the probability of arresting core damage following ac recovery was higher with intentional depressurization. The net effect of these two factors (higher probability of FCIs at low pressure, higher probability of arresting core damage with intentional depressurization) was that the probability of FCIs was actually reduced when the RCS was intentionally depressurized.

The sensitivity studies have shown that the key factor for Surry in determining risks related to containment failure from loads generated by direct containment heating are the loads themselves and the containment structural integrity, rather than the likelihood of being at a reduced pressure at vessel breach. This is a Surry-specific result, but the methodology described in this report could be used to evaluate other plants.

The updated evaluation indicates that the probability of a high pressure vessel breach is even lower than estimated for the NUREG-1150 study. Intentional depressurization decreased the probability of being at high pressure at vessel breach, but by only a small amount since the probability of a high pressure vessel breach is not very high even without intentional depressurization. This study has indicated that intentional depressurization at Surry would give minimal benefit. It has also demonstrated a methodology that could be used to evaluate the strategy for other plants that might see more benefit from depressurization, possibly Babcock & Wilcox or Combustion Engineering plants.

6.0 References

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Appendix A: Additional Plots for Sensitivity Evaluations

This appendix contains plots showing the full distributions for the results of the sensitivity studies. For each of the variations examined, figures are included that show the relative likelihood of vessel breach occurring while at high, intermediate and low RPV pressures. Also shown are the likelihood of a steam generator tube rupture (SGTR) and the likelihood of recovering from the accident and avoiding vessel breach.

Histograms are used to show the probability distributions, in which the length of the histogram rectangles in the x direction represents the fraction of the samples that fall within the range indicated on the y axis. Also noted on the histogram plots are the mean, the median, and the 5th and 95th percentiles of the distributions.

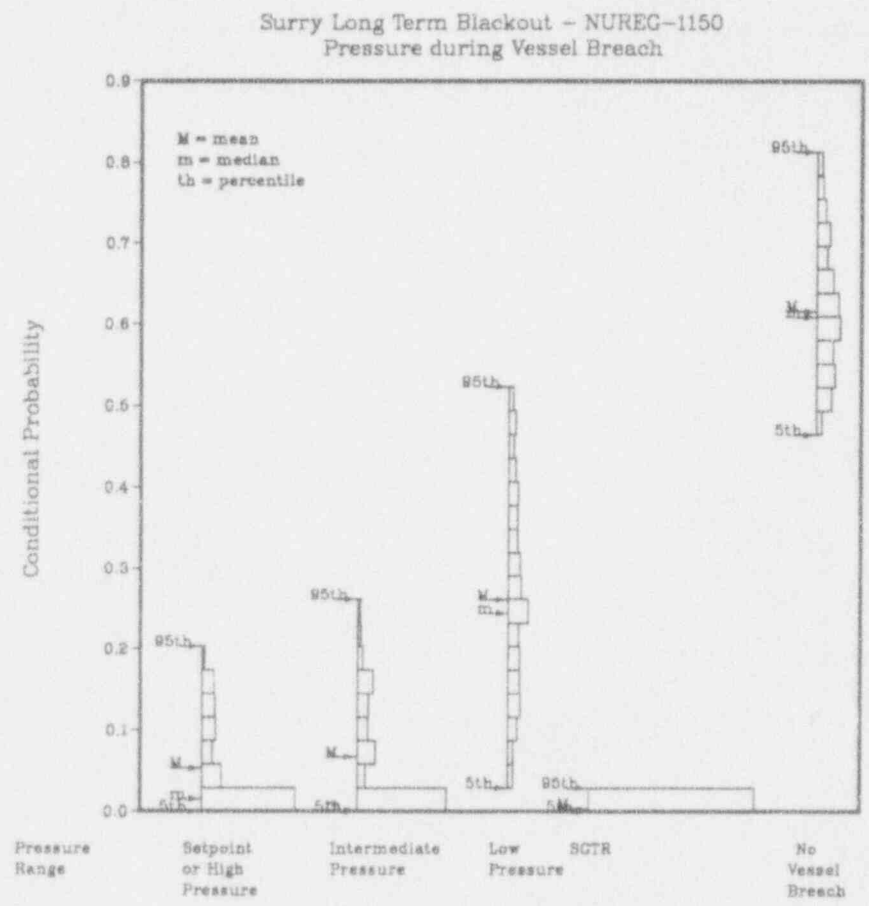

Figure A-1 Pressure at vessel breach - NUREG-1150

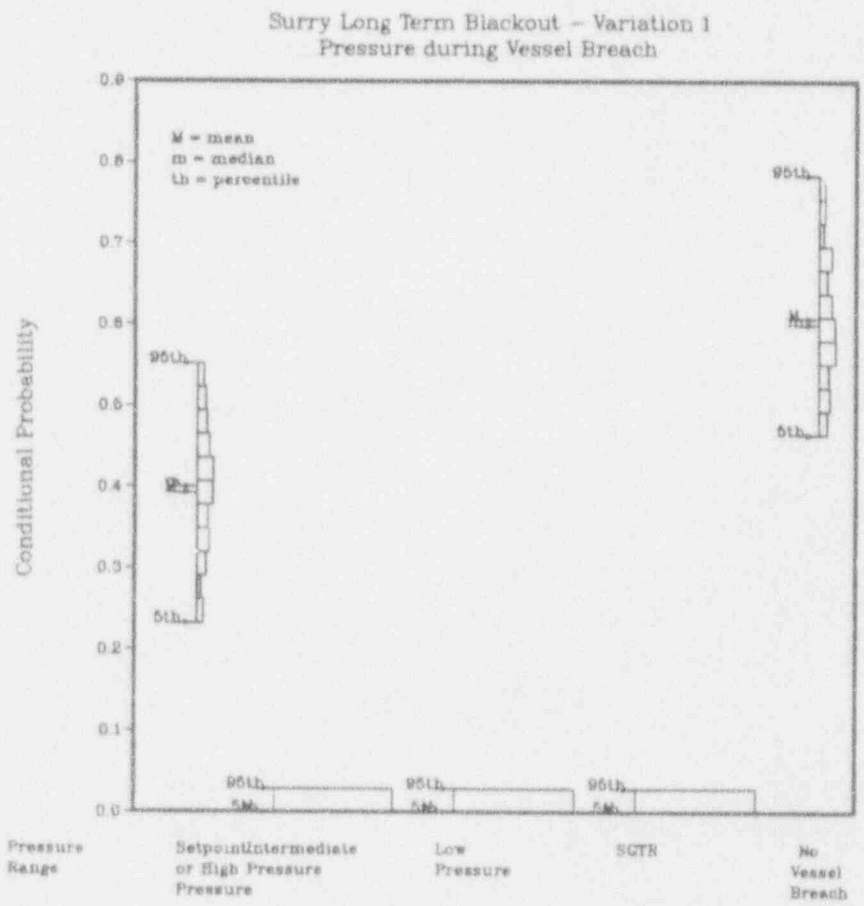

Figure A-2 Pressure at vessel breach - Variation 1

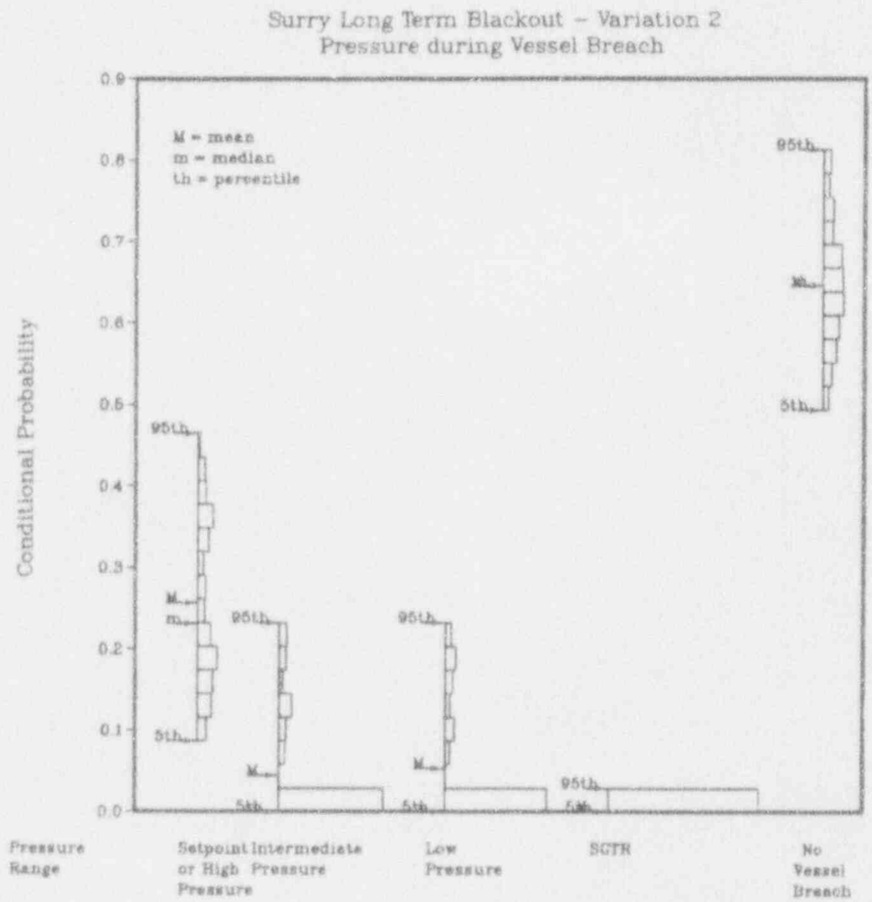

Figure A-3 Pressure at vessel breach - Variation 2

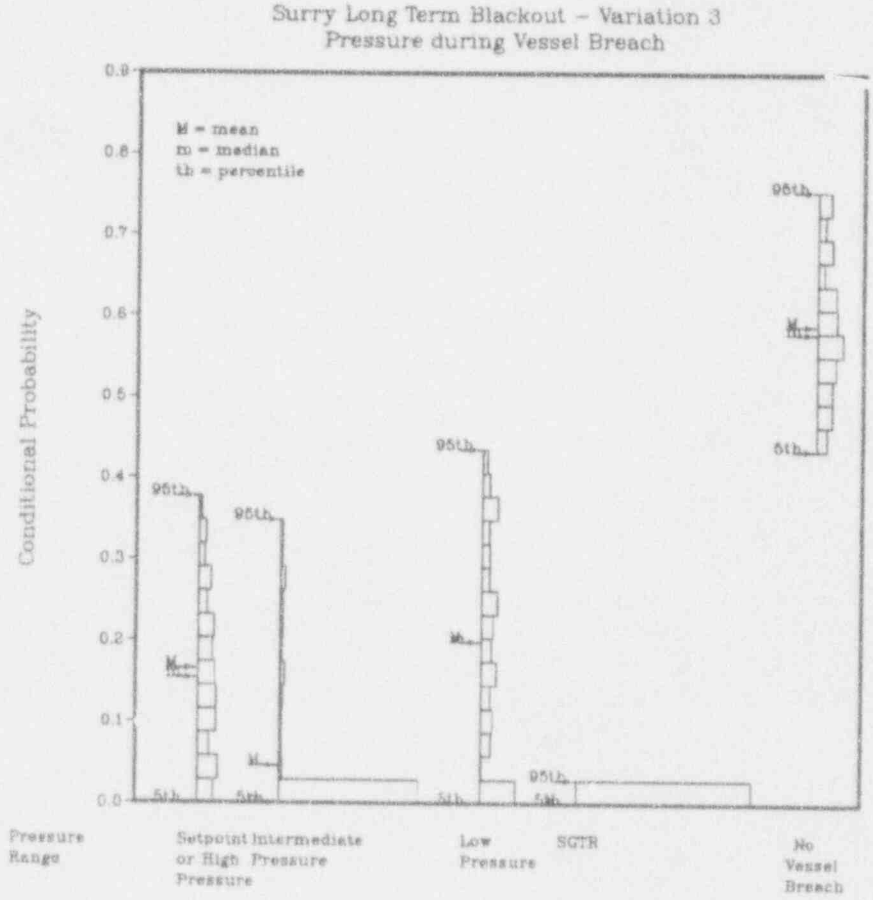

Figure A-4 Pressure at vessel breach - Variation 3

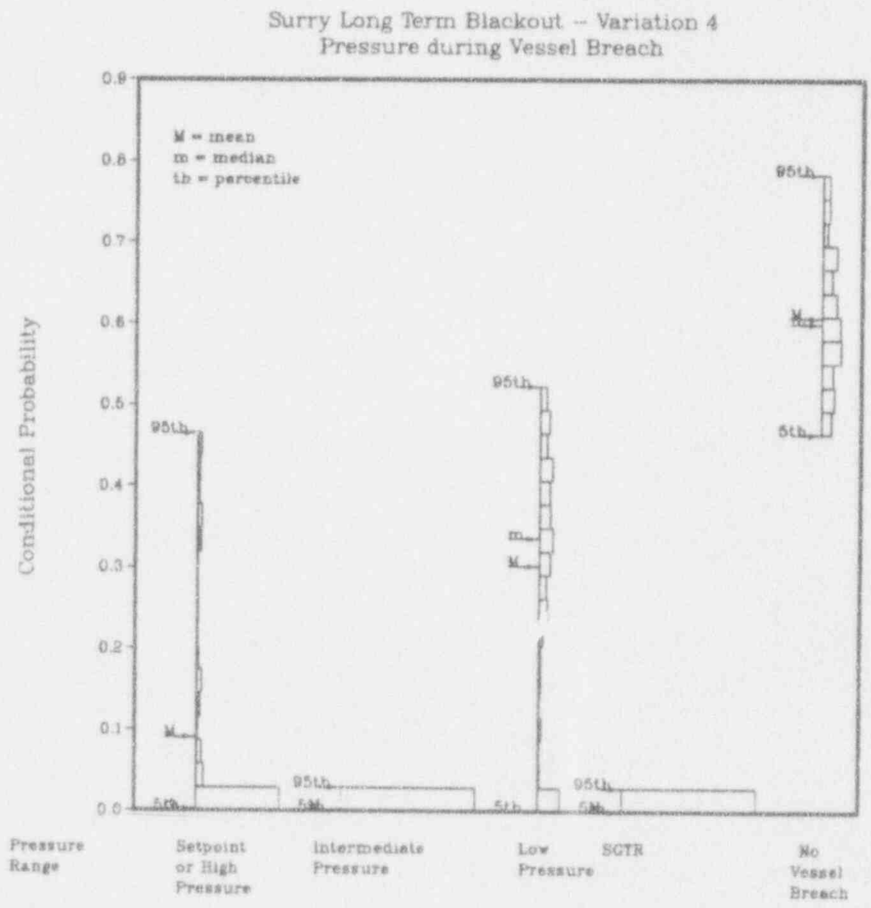

Figure A-5 Pressure at vessel breach - Variation 4

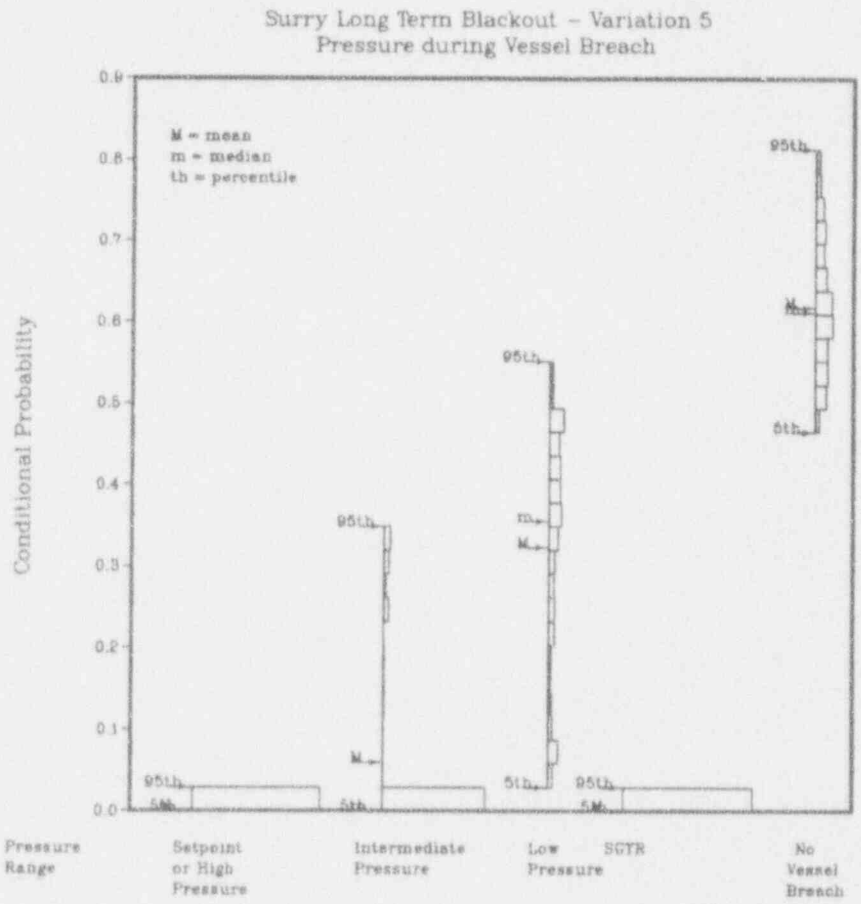

Figure A-6 Pressure at vessel breach - Variation 5

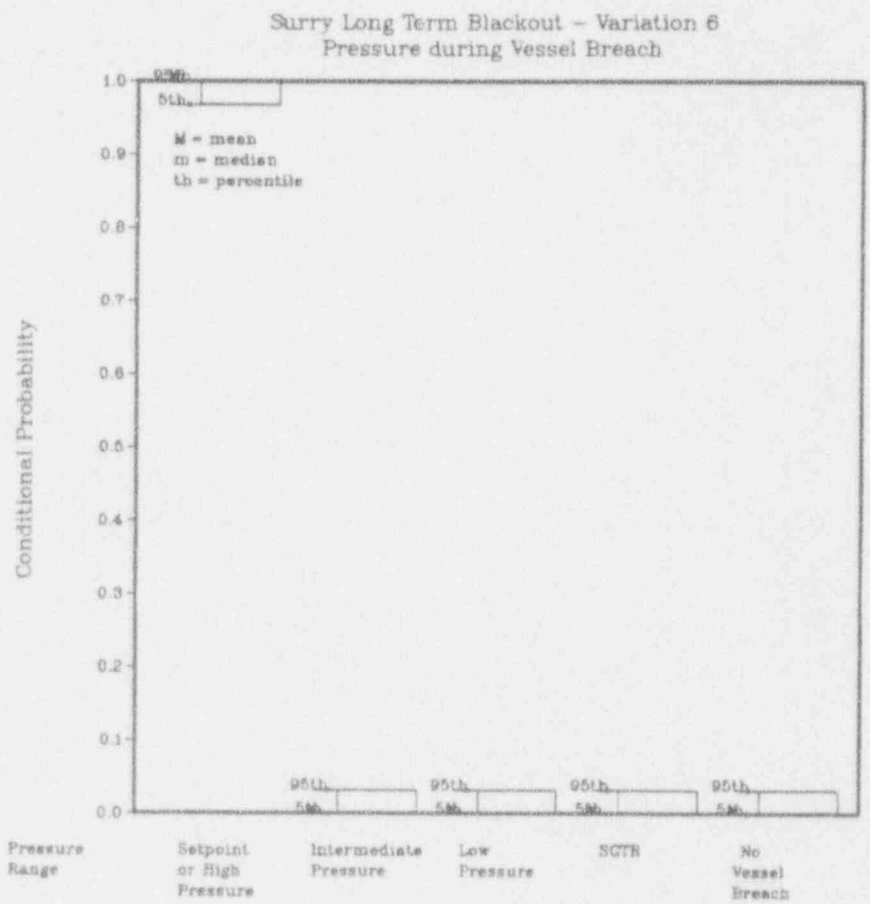

Figure A-7 Pressure at vessel breach - Variation 6

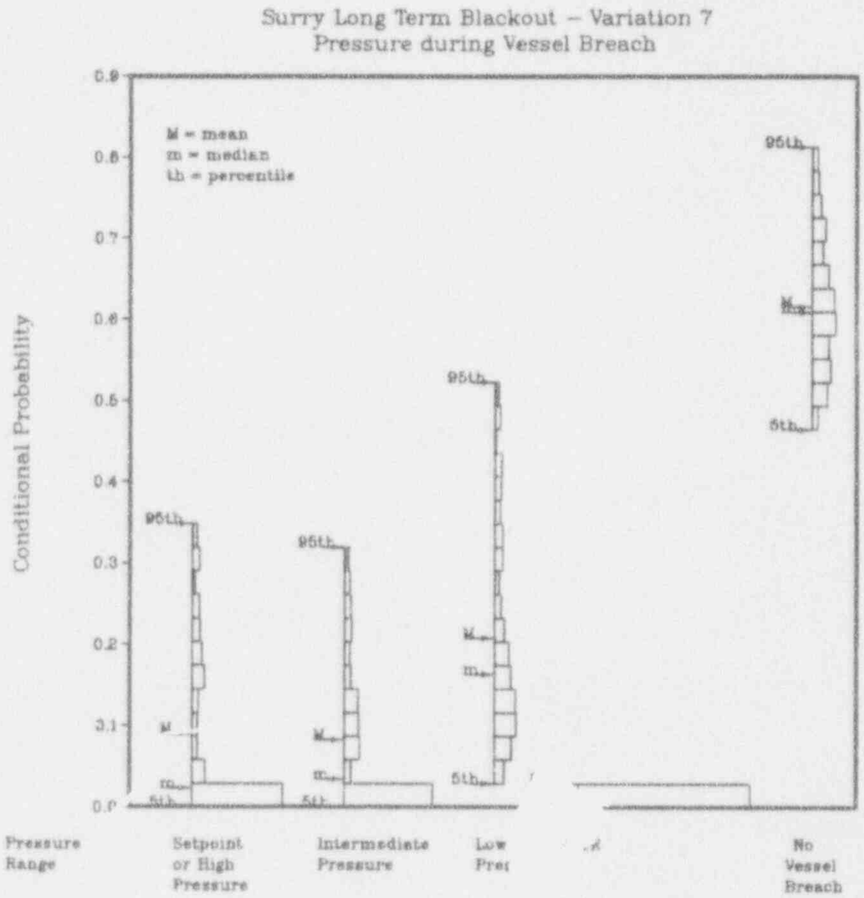

Figure A-8 Pressure at vessel breach - Variation 7

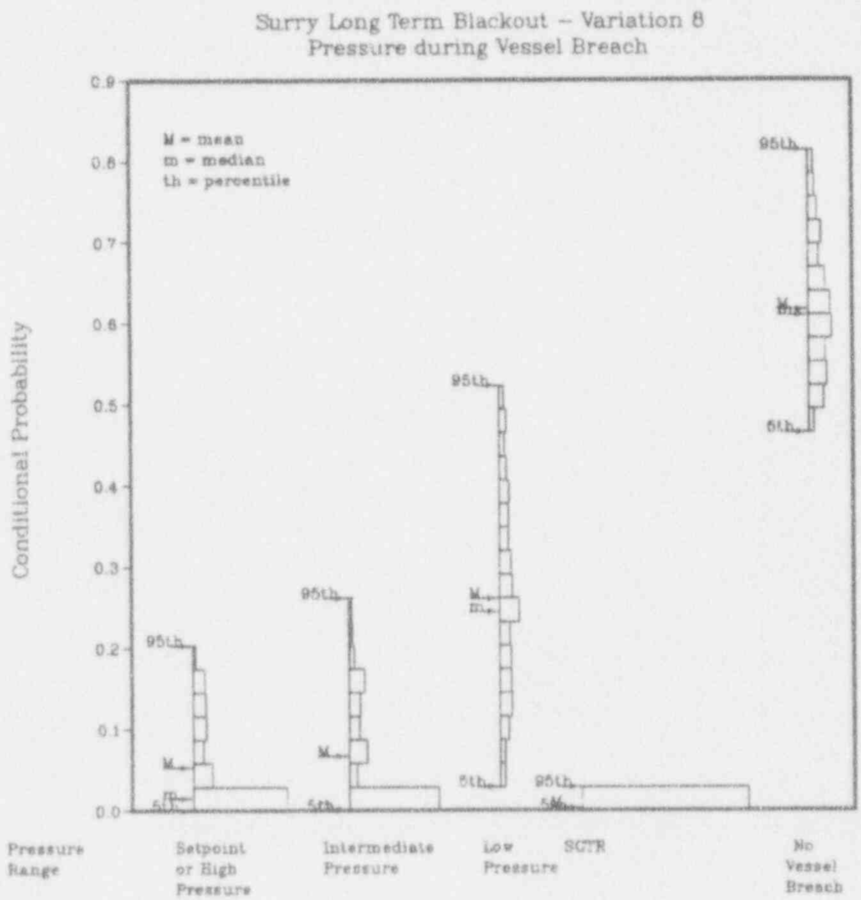

Figure A-9 Pressure at vessel breach - Variation 8

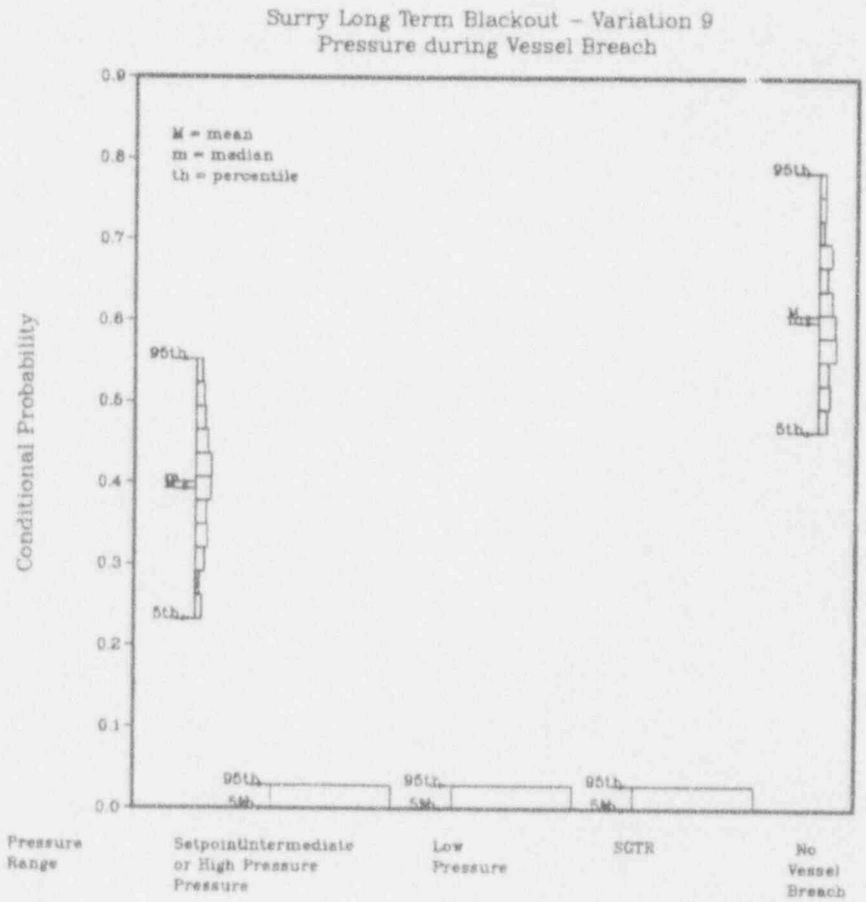

Figure A-10 Pressure at vessel breach - Variation 9

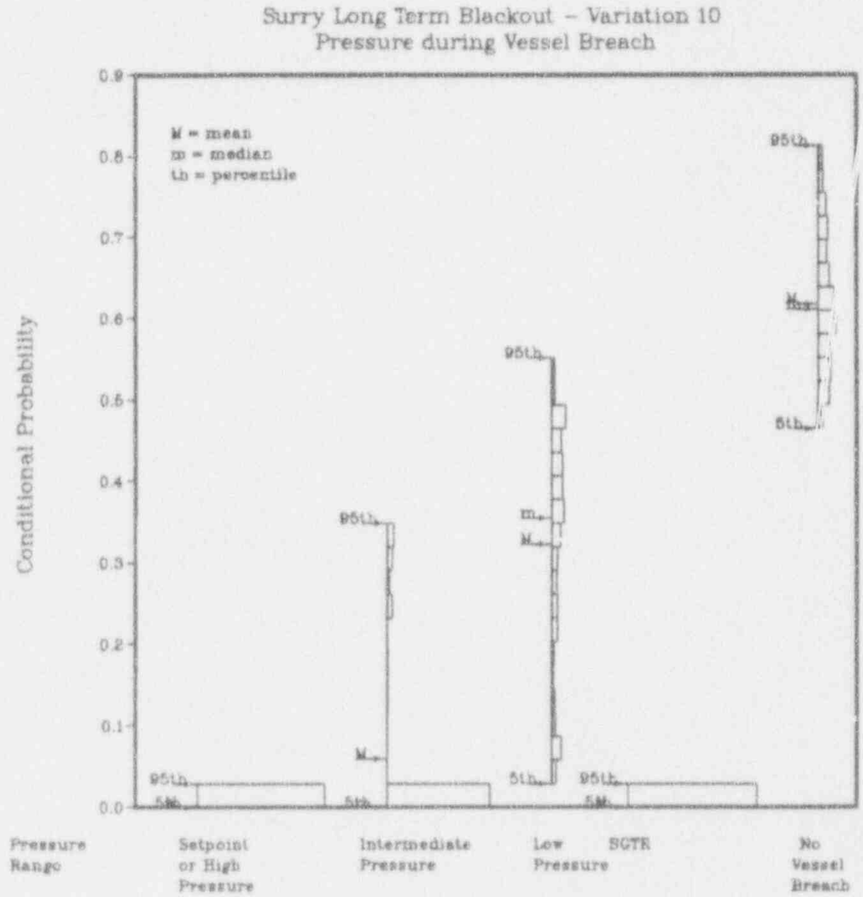

Figure A-11 Pressure at vessel breach - Variation 10

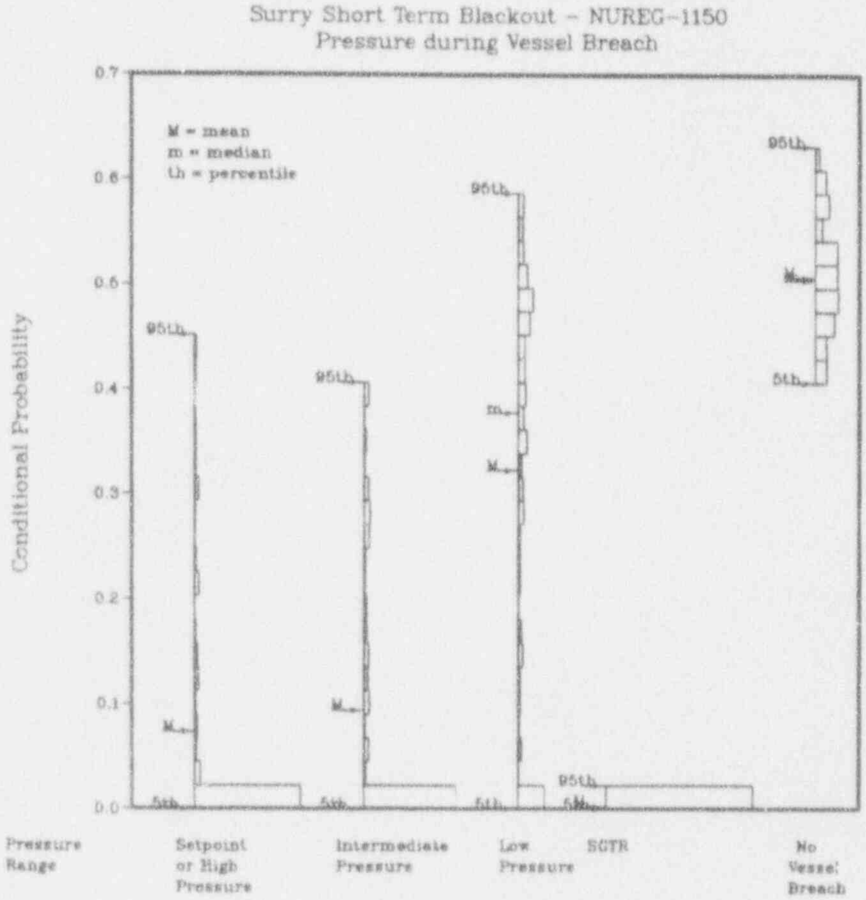

Figure A-15 Pressure at vessel breach - NUREG-1150

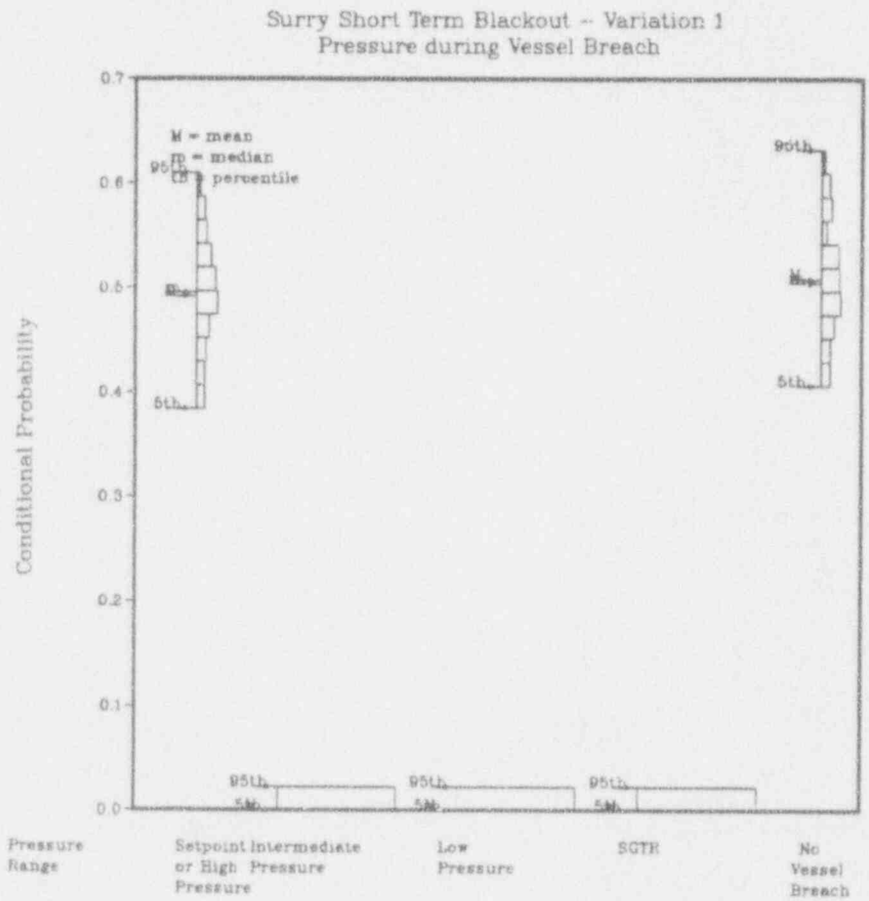

Figure A-16 Pressure at vessel breach - Variation 1

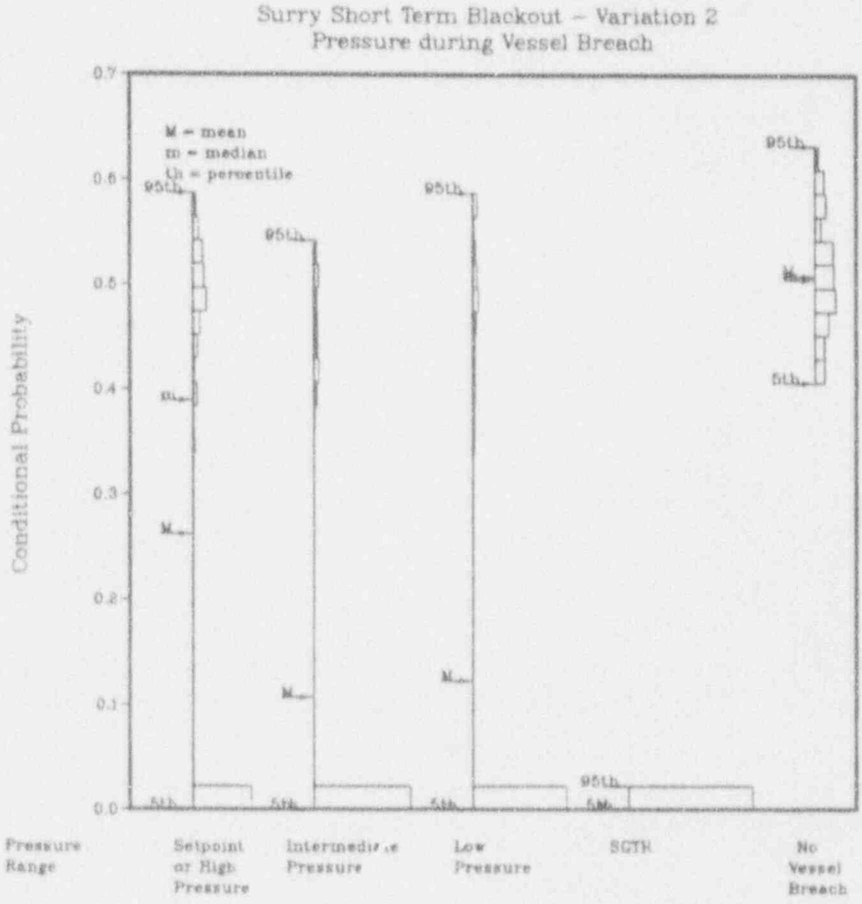

Figure A-17 Pressure at vessel breach - Variation 2

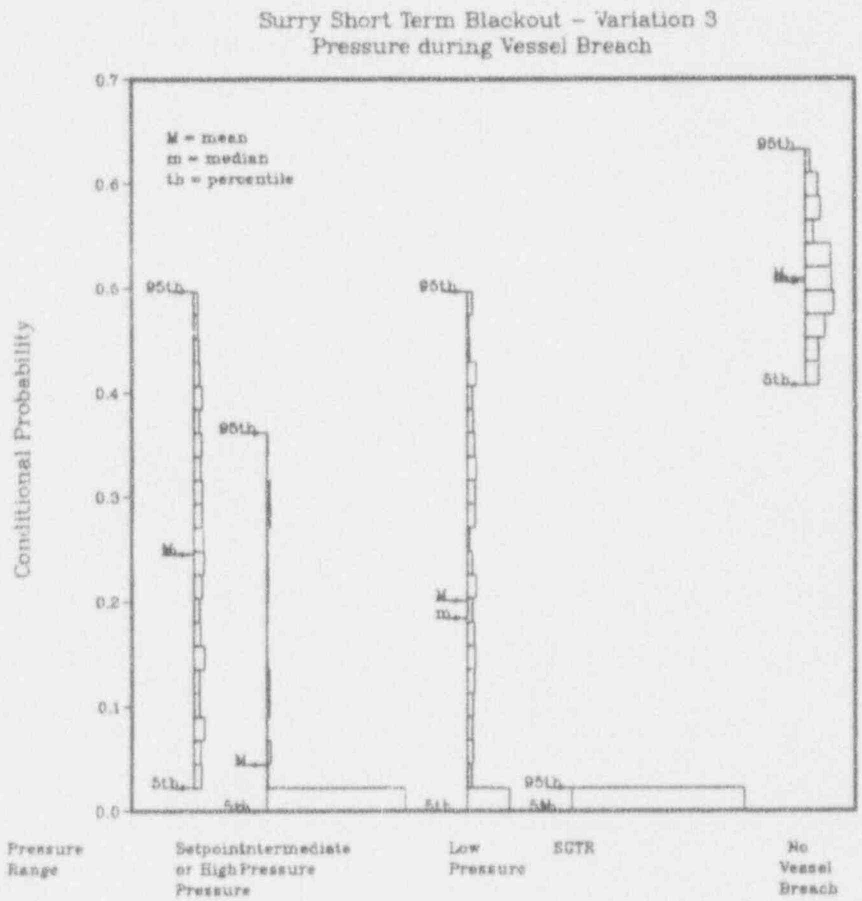

Figure A-18 Pressure at vessel breach - Variation 3

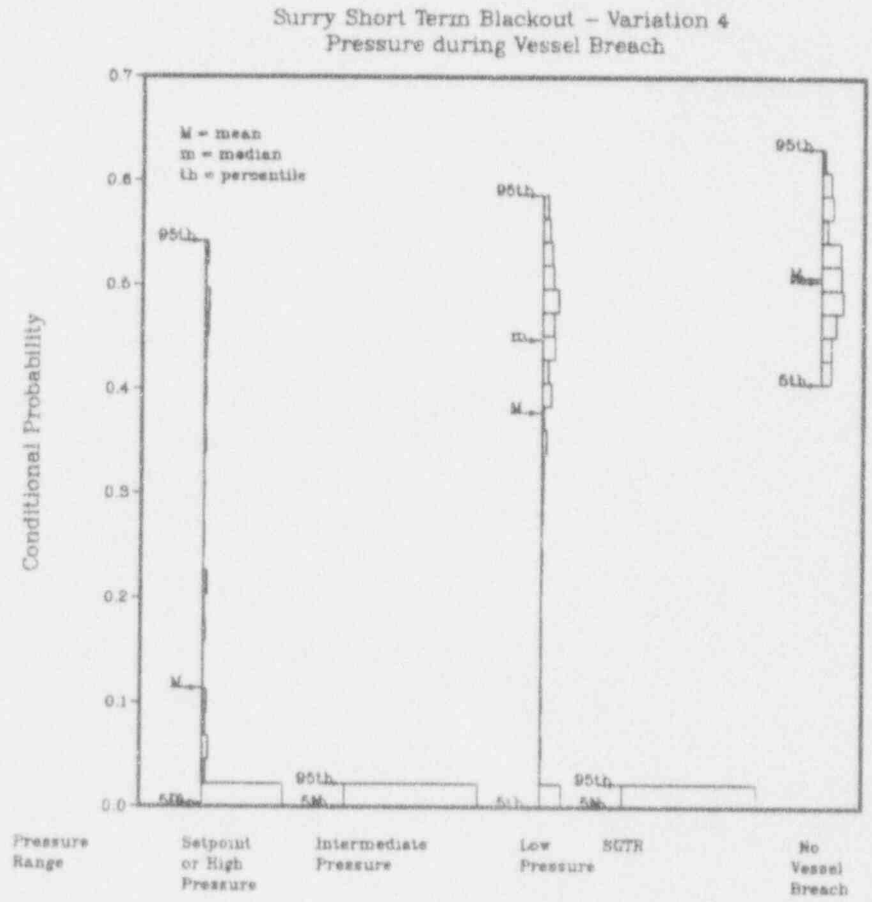

Figure A-19 Pressure at vessel breach - Variation 4

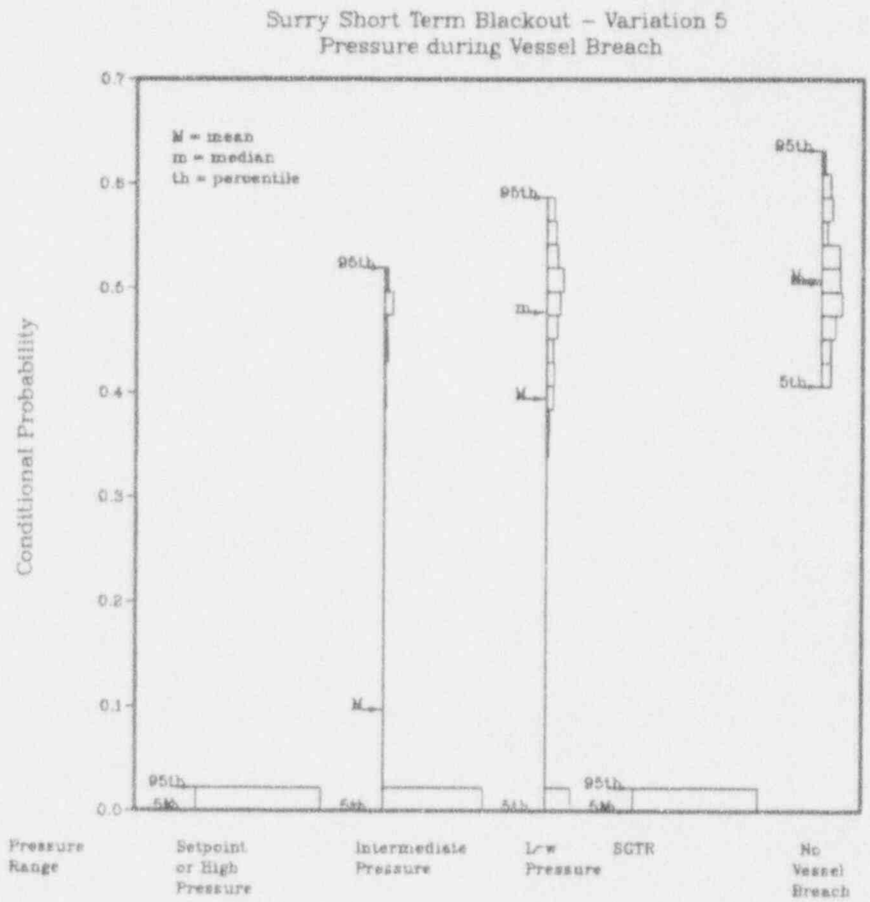

Figure A-20 Pressure at vessel breach - Variation 5

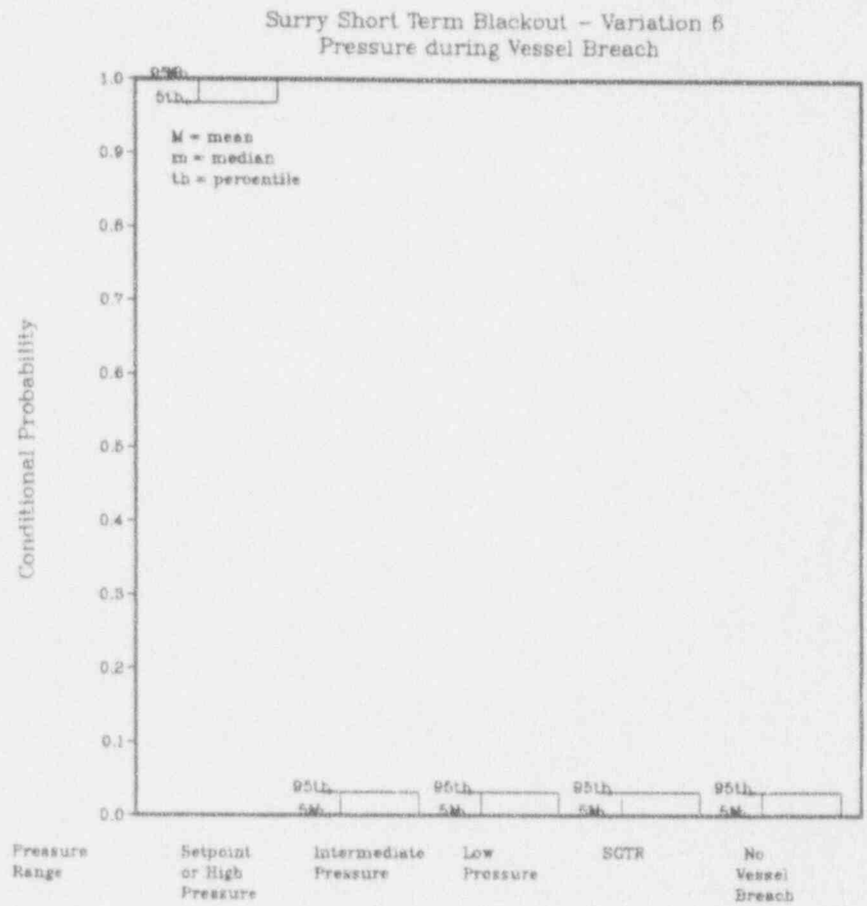

Figure A-21 Pressure at vessel breach - Variation 6

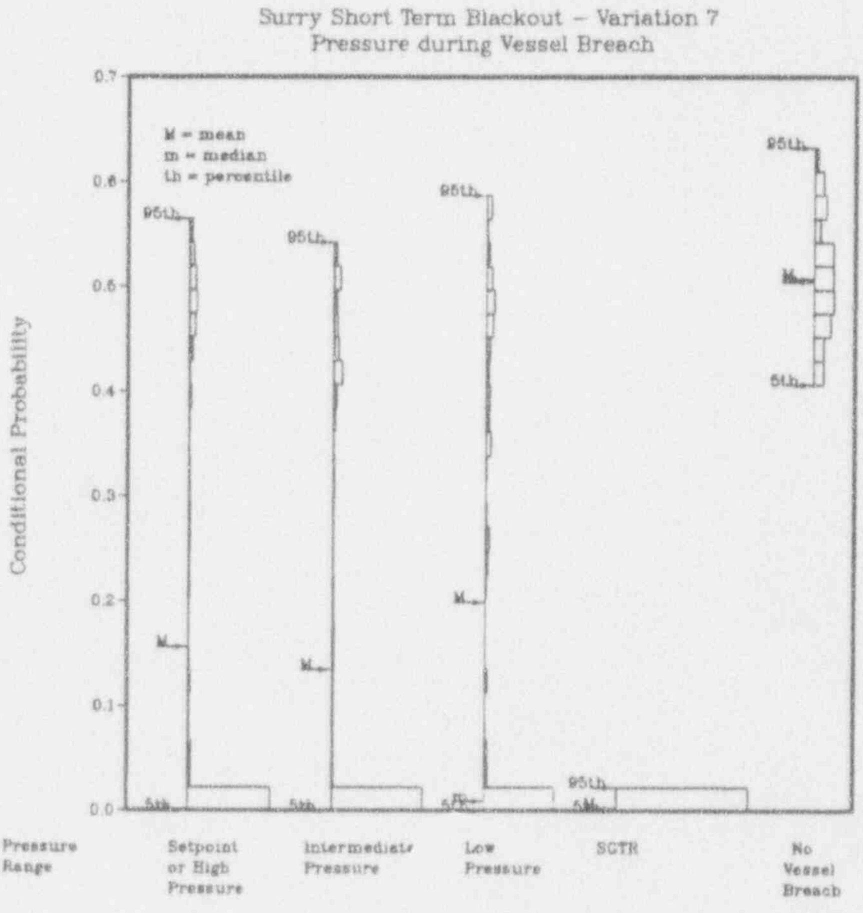

Figure A-22 Pressure at vessel breach - Variation 7

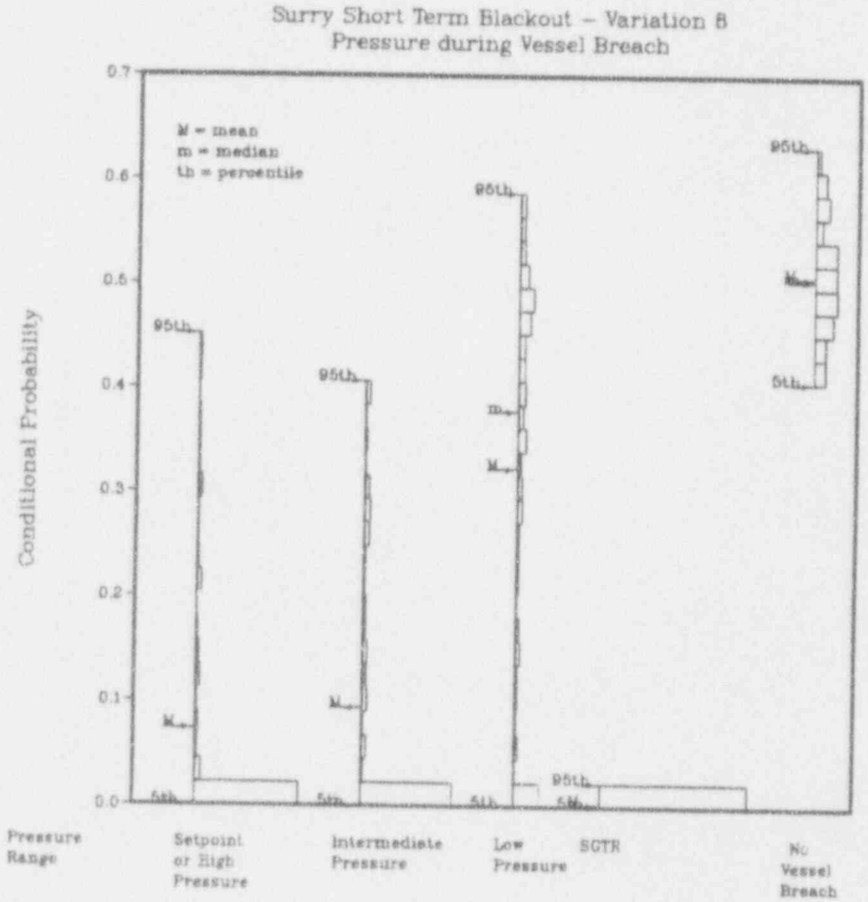

Figure A-23 Pressure at vessel breach - Variation 8

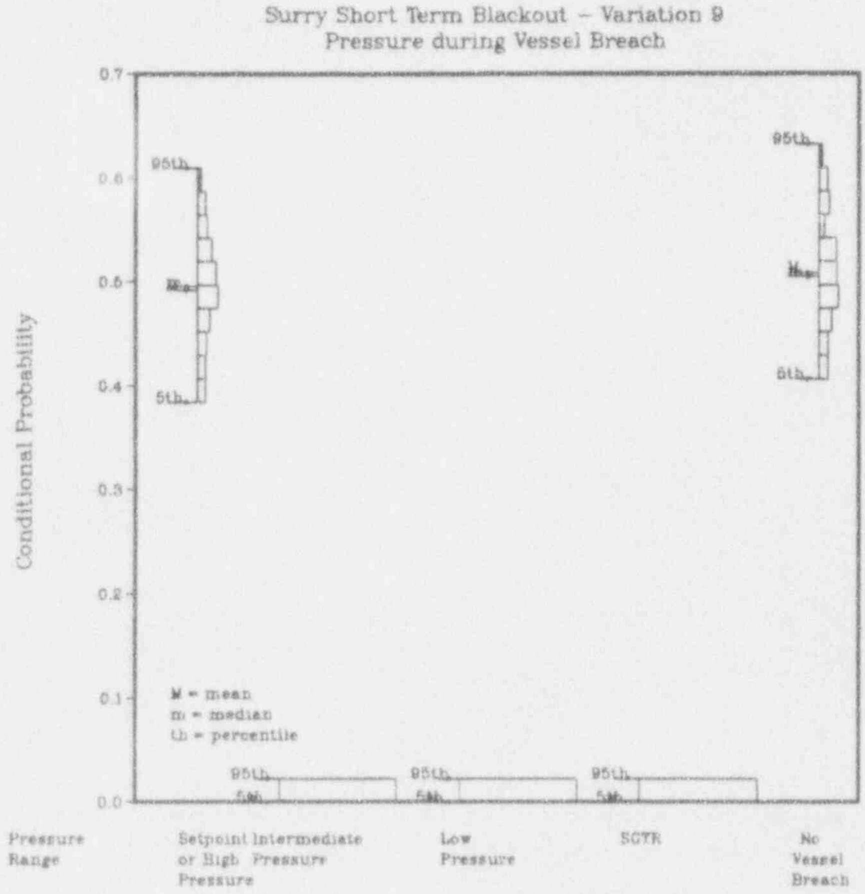

Figure A-24 Pressure at vessel breach - Variation 9

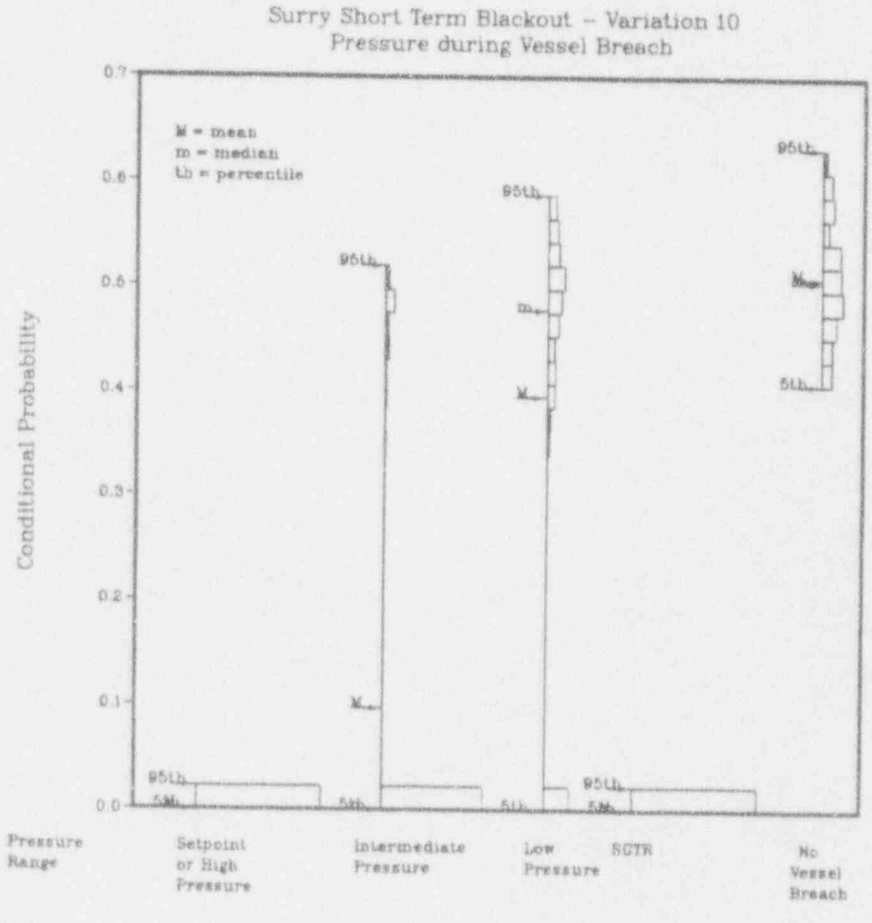

Figure A-25 Pressure at vessel breach - Variation 10

Appendix B: Additional Plots for Updated Evaluation

This appendix contains additional plots for the updated evaluation of intentional depressurization. Included are probability distributions for vessel breach occurring at the various reactor coolant system (RCS) pressure levels.

Histograms are used to show the probability distributions, in which the length of the histogram rectangles in the y direction represents the fraction of the samples that fall within the range indicated on the x axis. Also noted on the histogram plots are the mean, the median, and the 5th and 95th percentiles of the distributions.

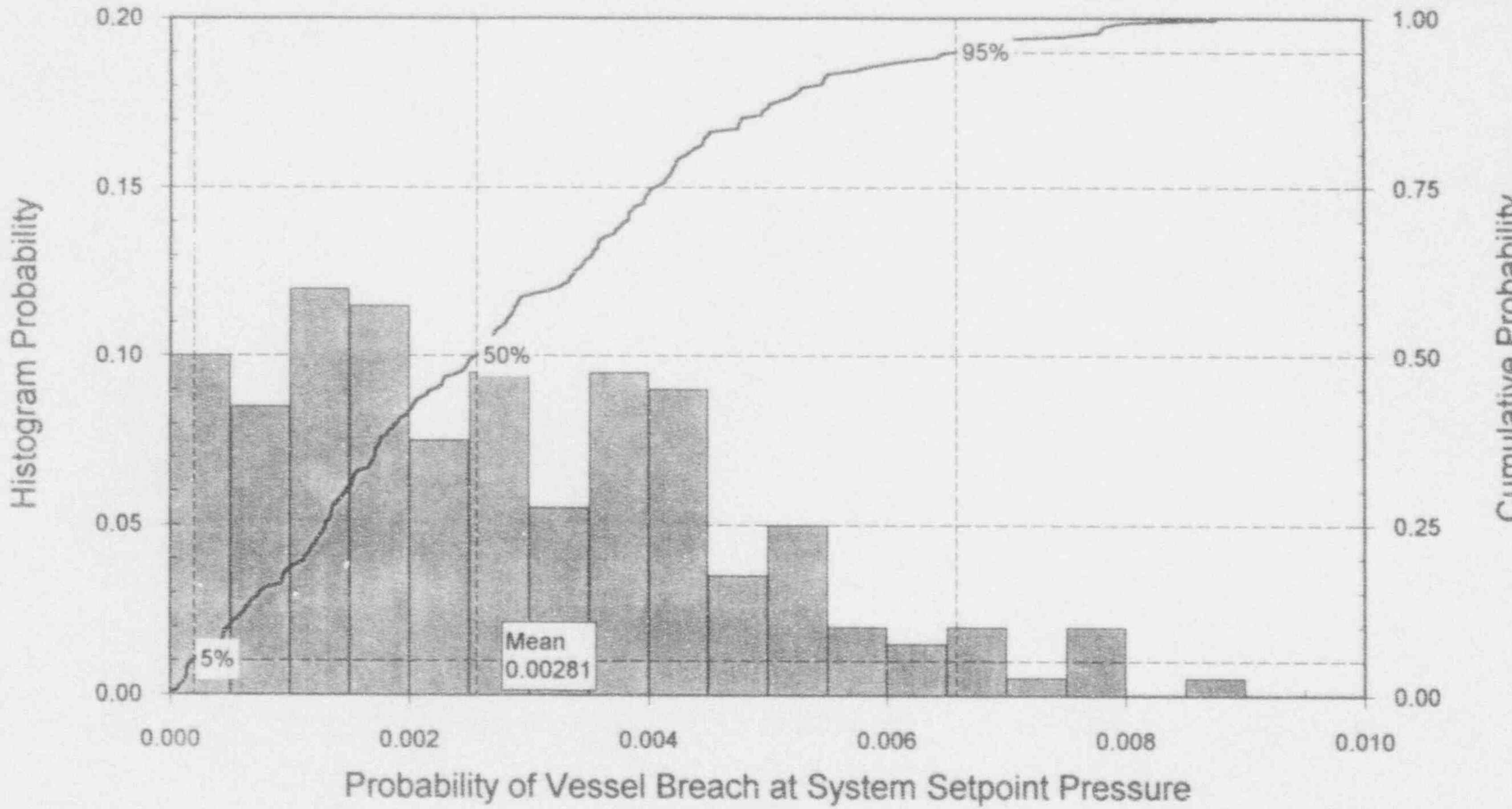

Figure B-1 Probability of vessel breach at system setpoint pressure for long-term station blackout without intentional depressurization

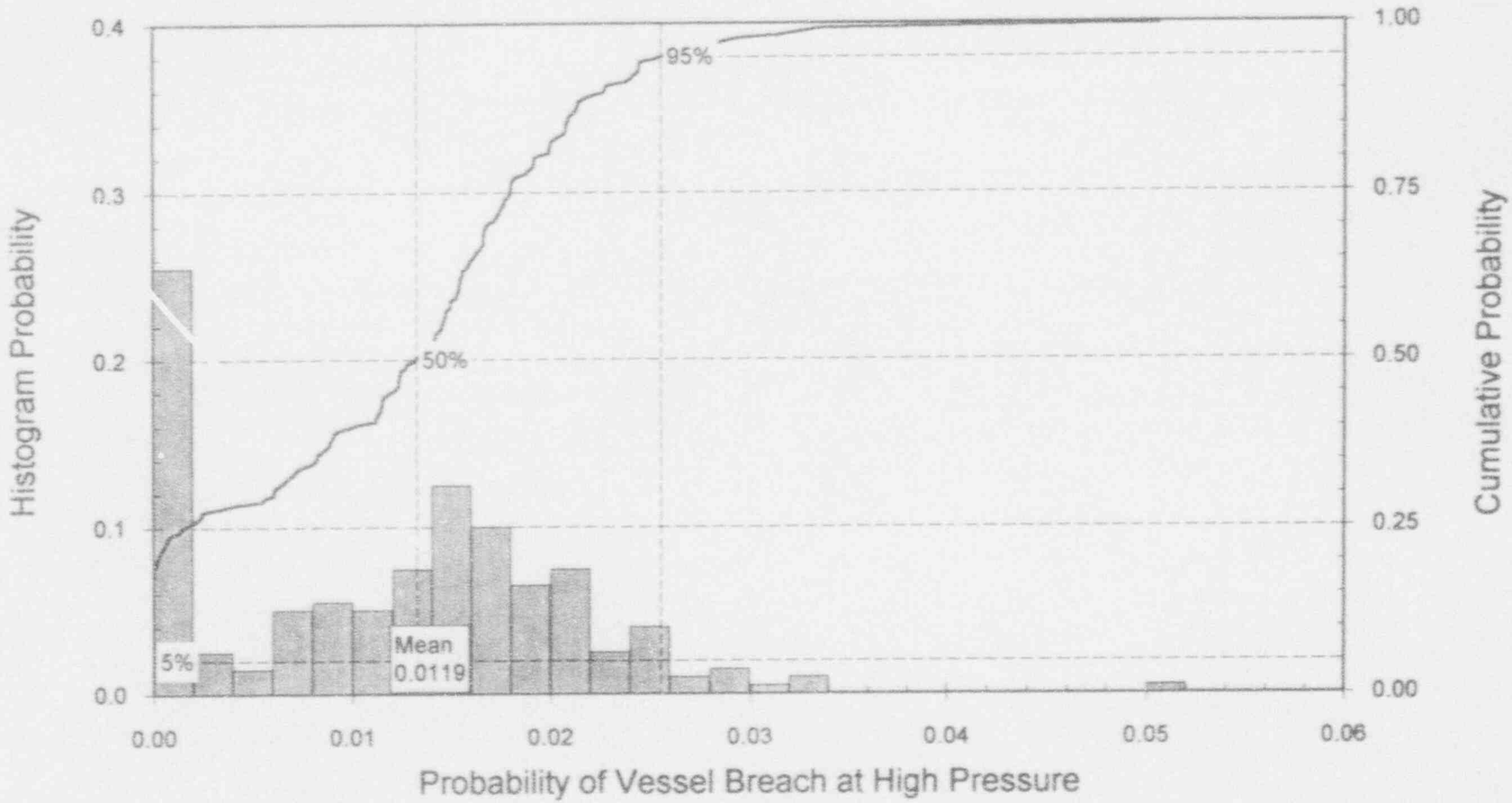

Figure B-2 Probability of vessel breach at high pressure for long-term station blackout without intentional depressurization

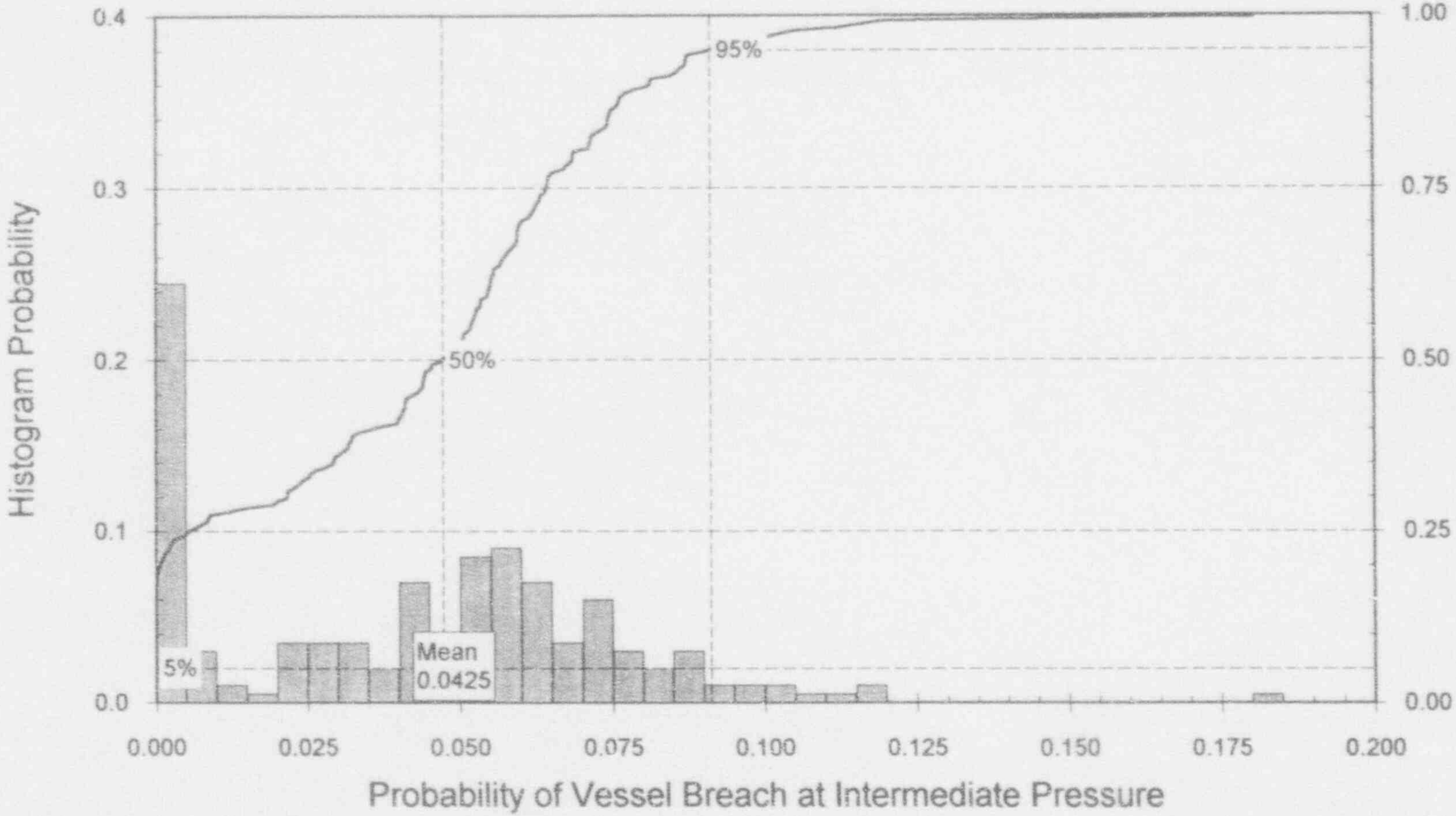

Figure B-3 Probability of vessel breach at intermediate pressure for long-term station blackout without intentional depressurization

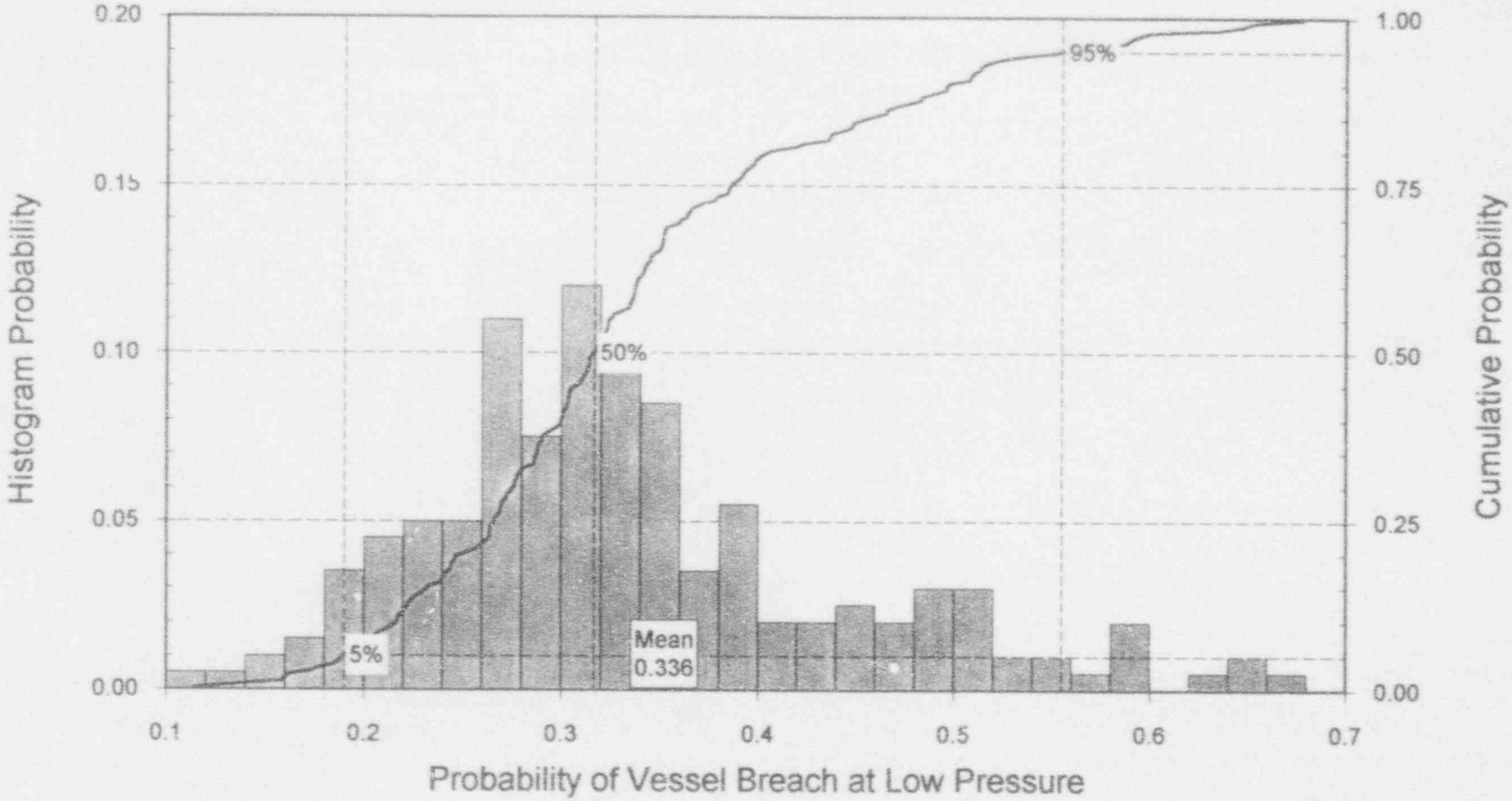

Figure B-4 Probability of vessel breach at low pressure for long-term station blackout without intentional depressurization

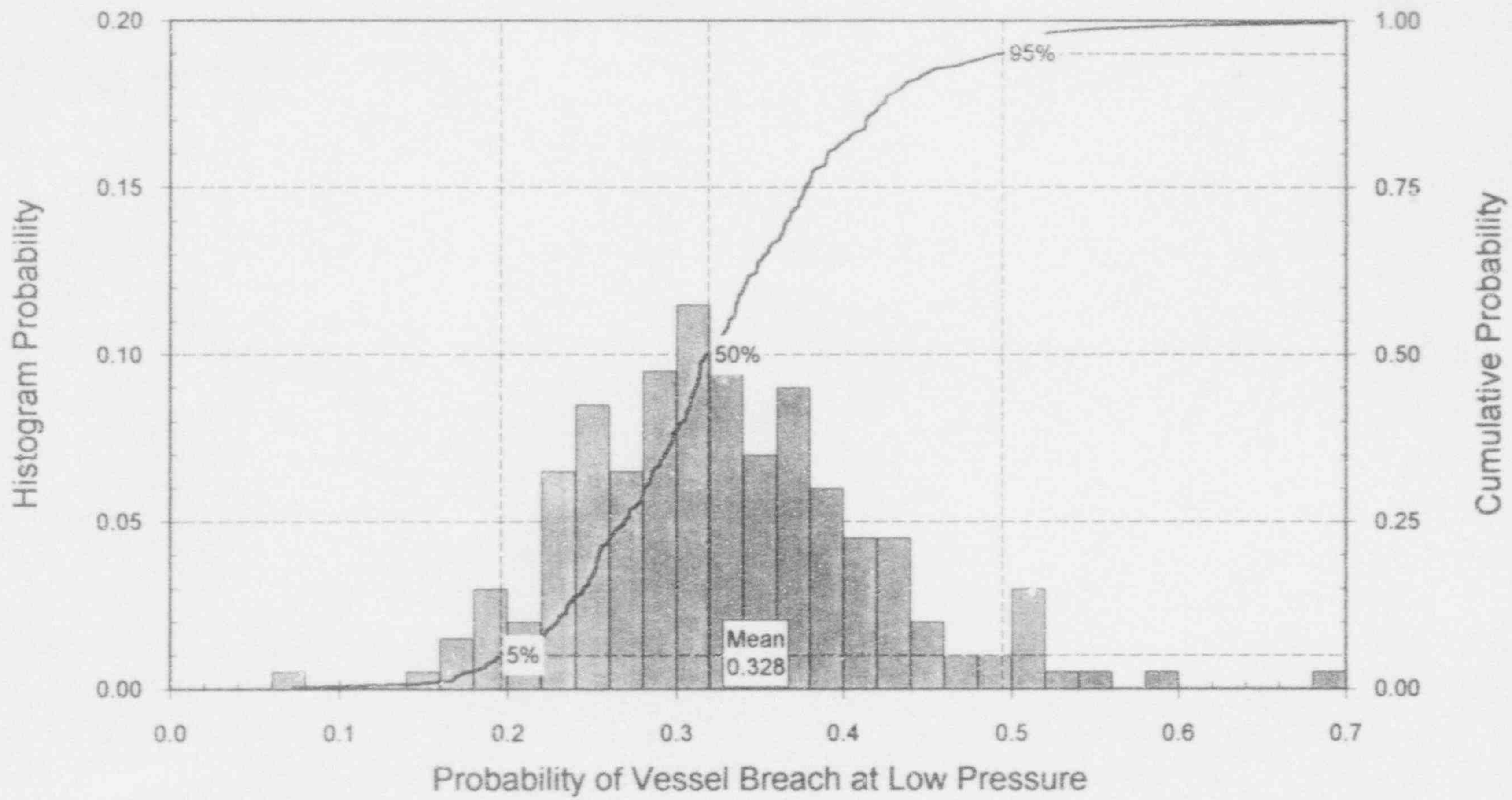

Figure B-5 Probability of vessel breach at low pressure for long-term station blackout with intentional depressurization

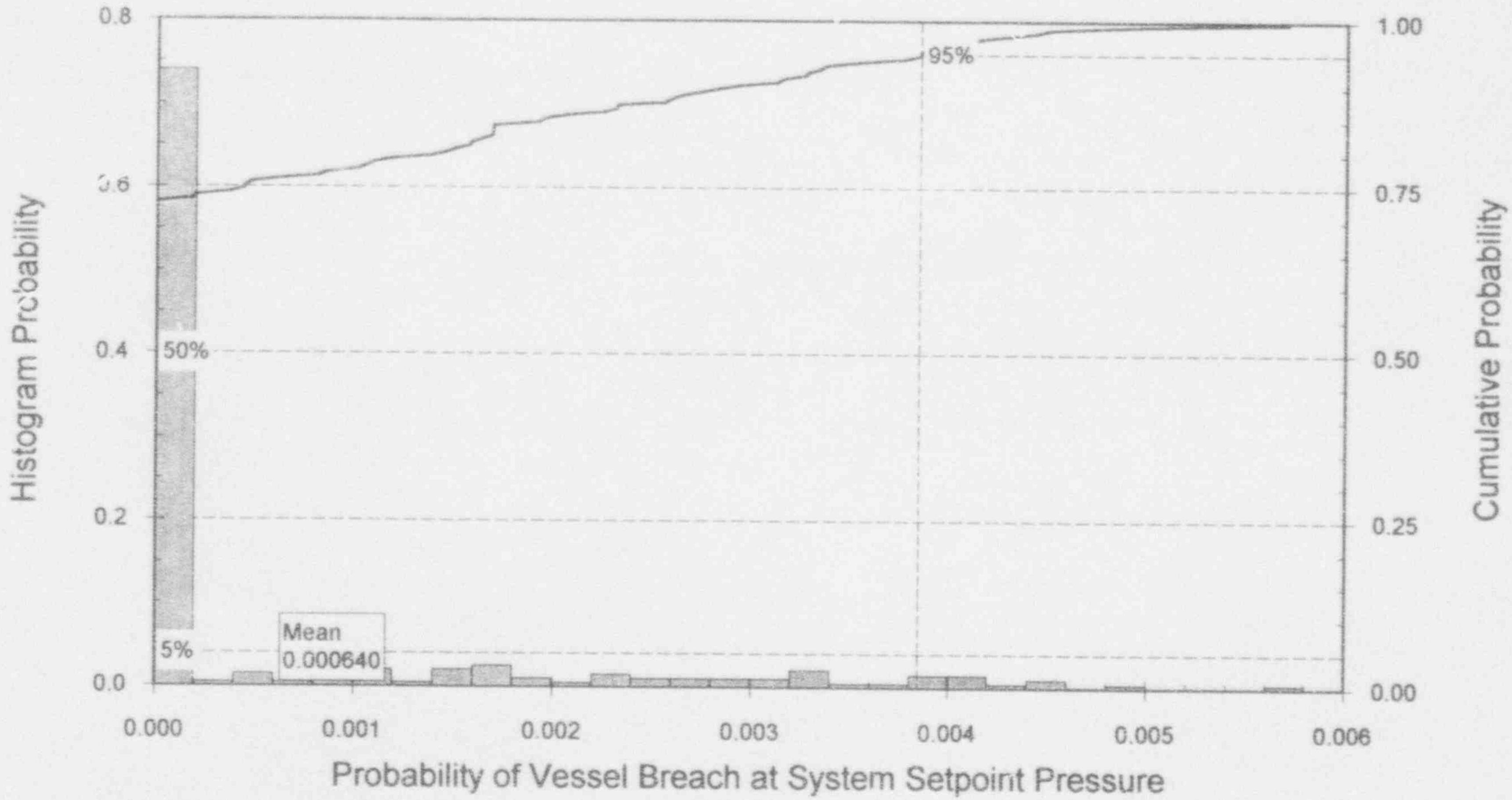

Figure B-6 Probability of vessel breach at system setpoint pressure for short-term station blackout without intentional depressurization

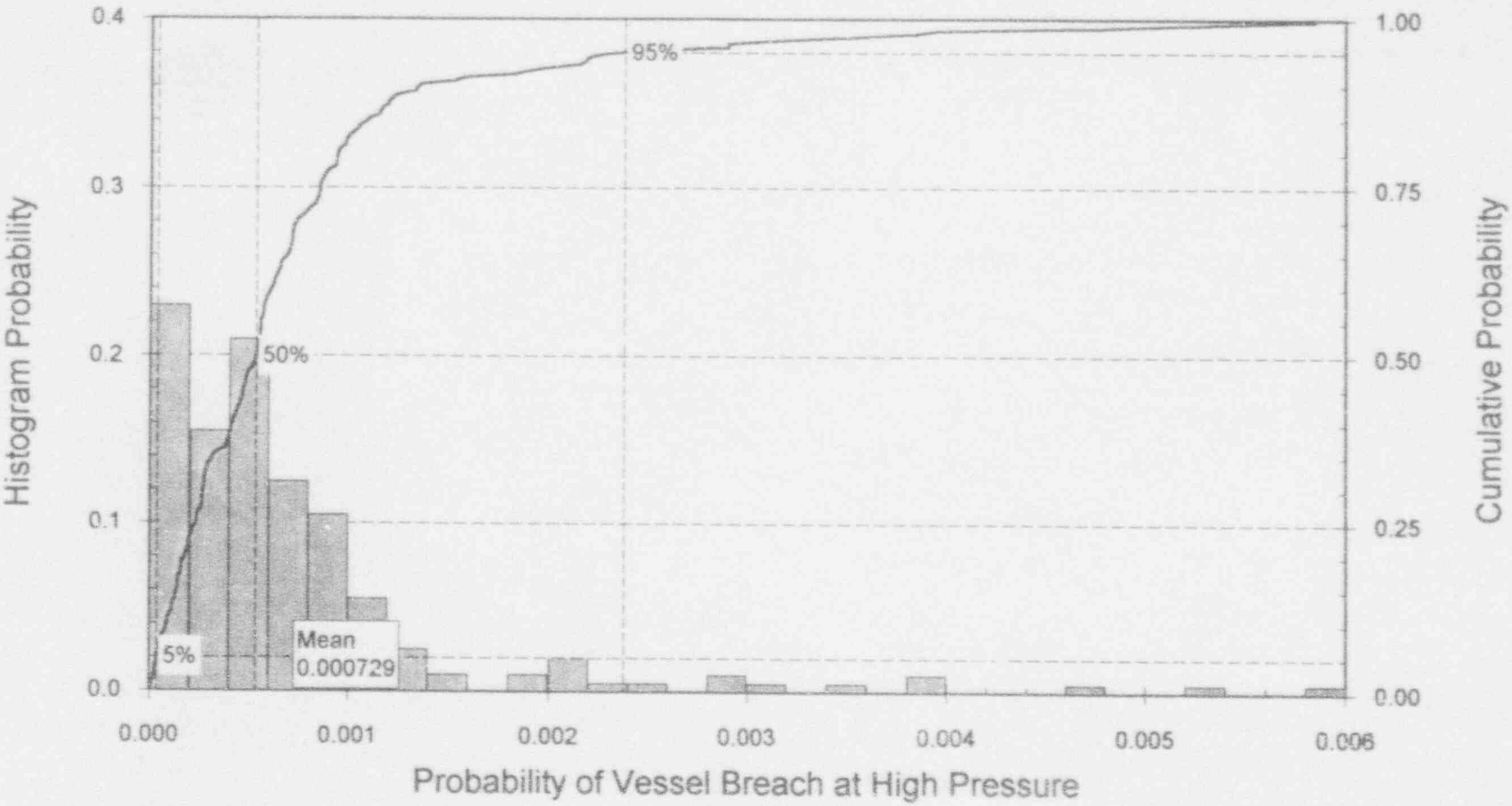

Figure B-7 Probability of vessel breach at high pressure for short-term station blackout without intentional depressurization

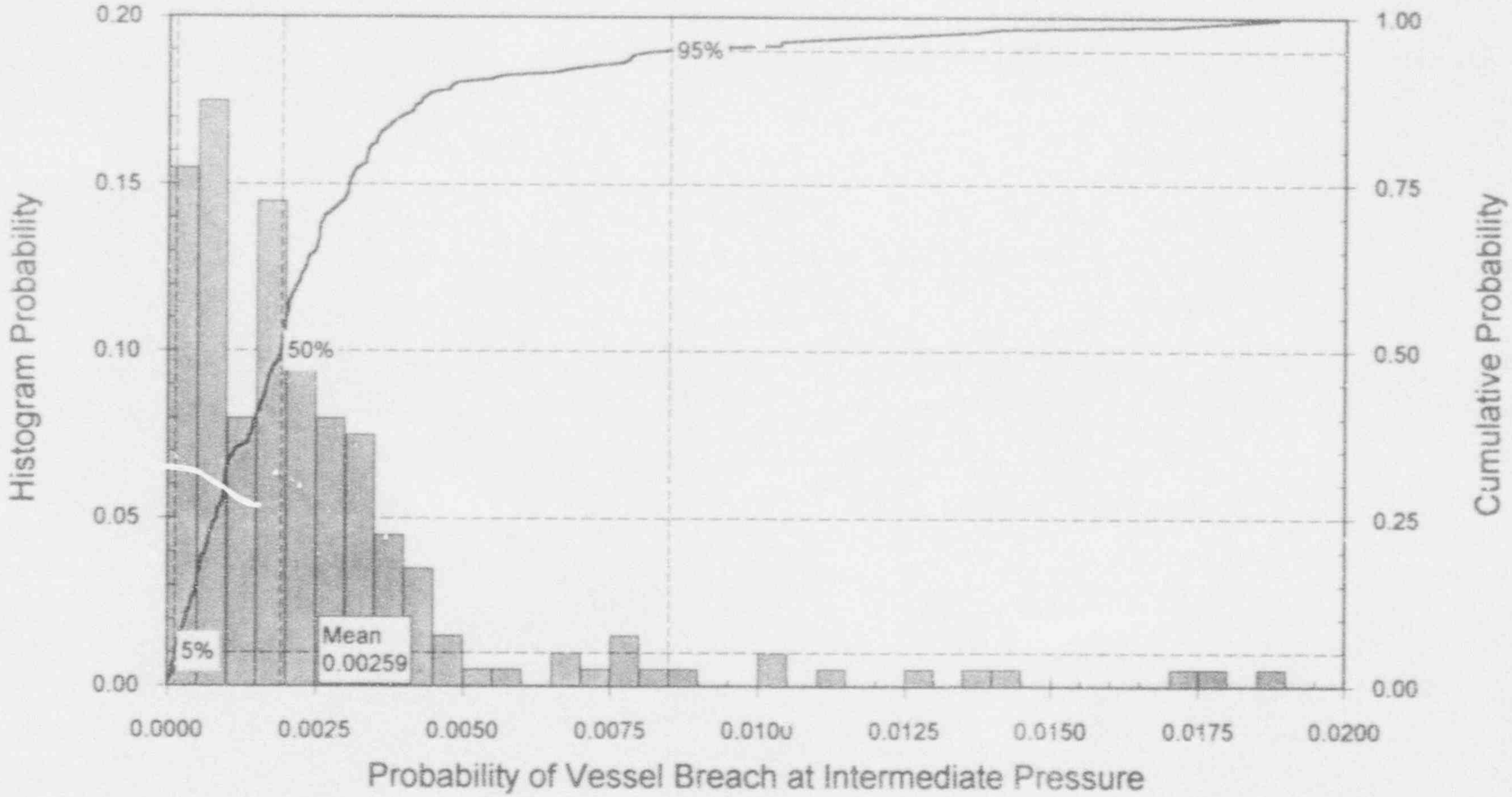

Figure B-8 Probability of vessel breach at intermediate pressure for short-term station blackout without intentional depressurization

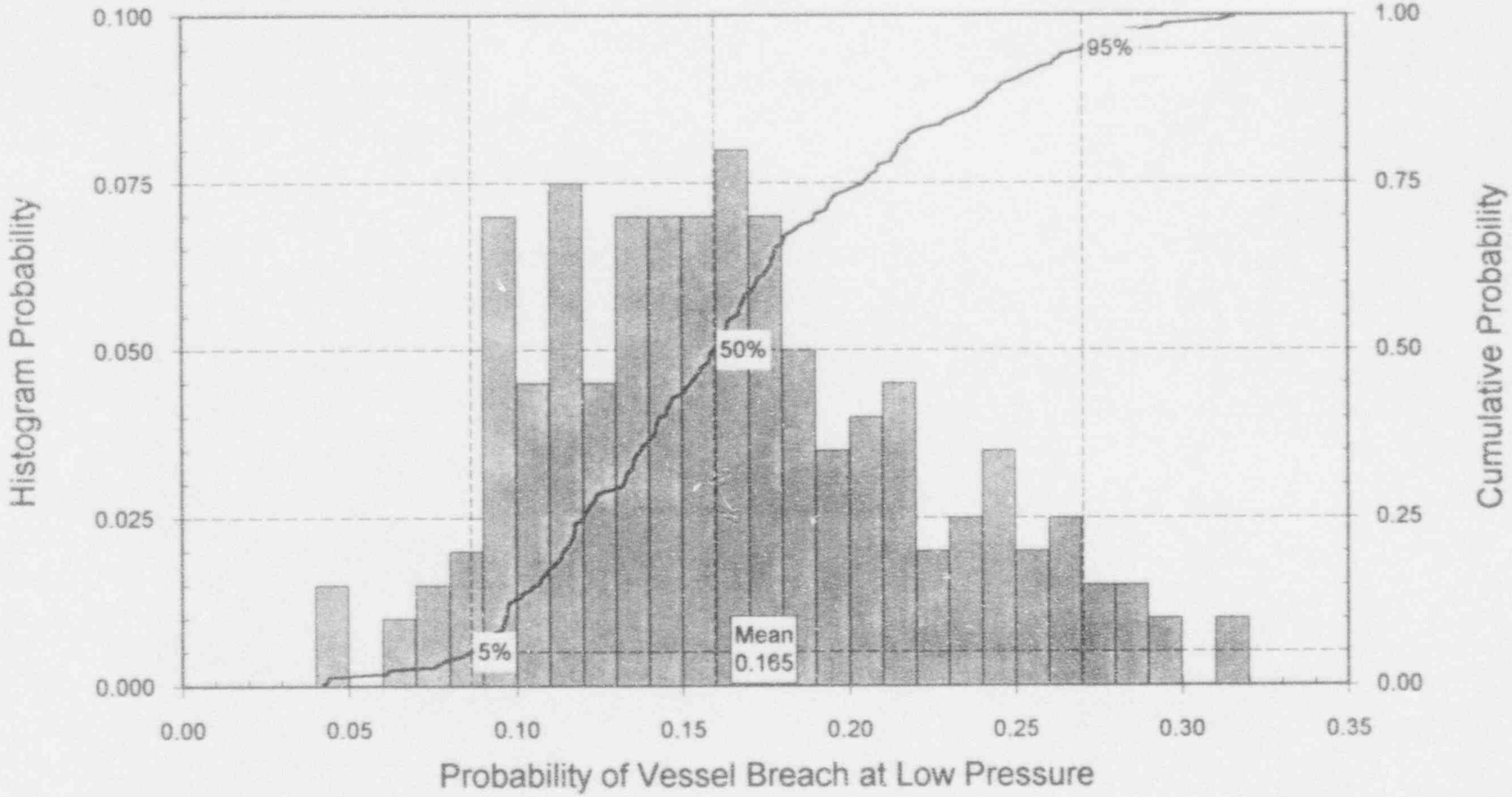

Figure B-9 Probability of vessel breach at low pressure for short-term station blackout without intentional depressurization

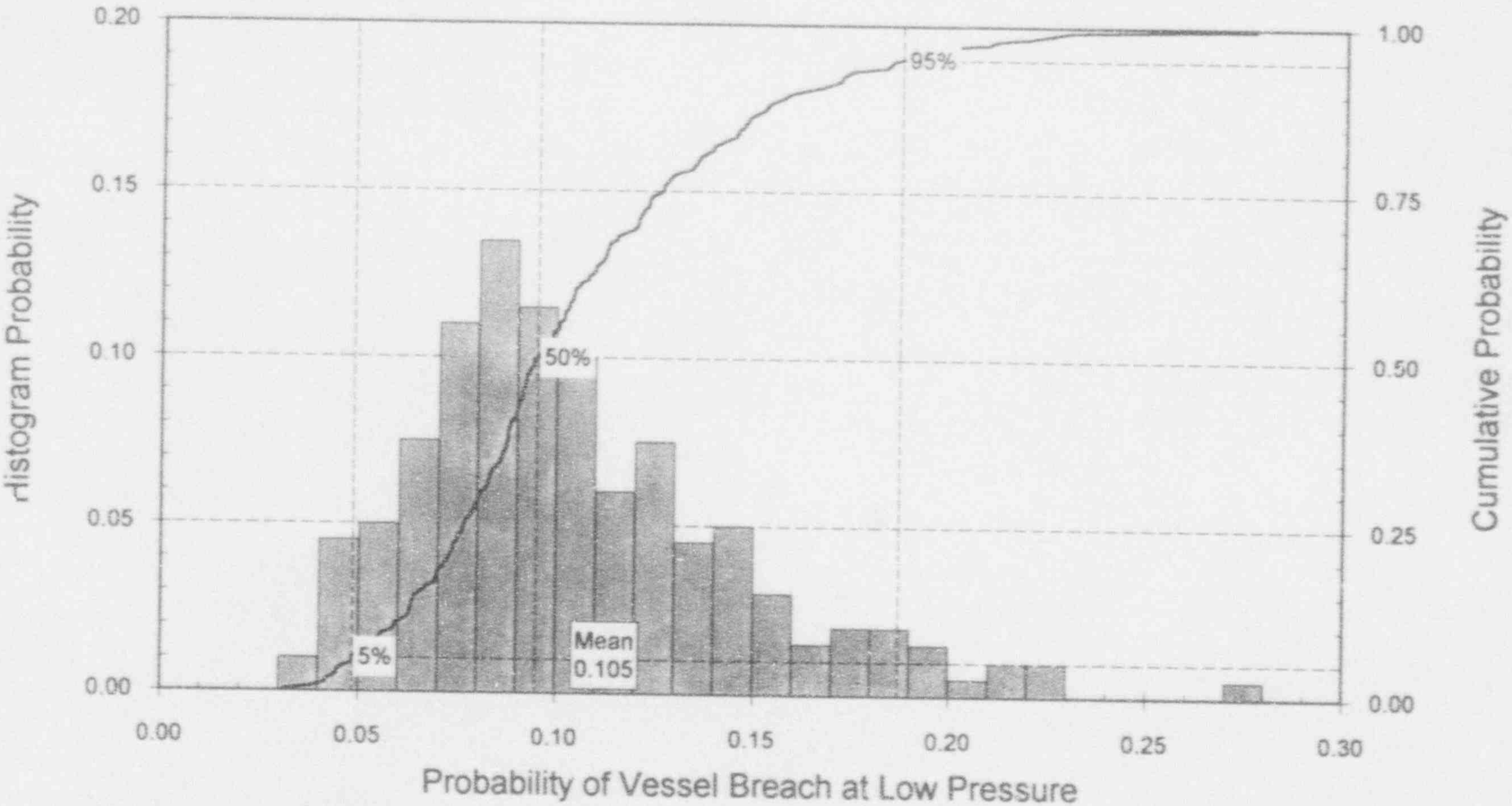

Figure B-10 Probability of vessel breach at low pressure for short-term station blackout with intentional depressurization

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

An accident management strategy has been proposed in which the reactor coolant system is intentionally depressurized during an accident. The aim is to reduce the containment pressurization that would result from high pressure ejection of molten debris at vessel breach. Probabilistic risk assessment (PRA) methods were used to evaluate this strategy for the Surry nuclear power plant. Sensitivity studies were conducted using event trees that were developed for the NUREG-1150 study. It was found that depressurization (intentional or unintentional) had minimal impact on the containment failure probability at vessel breach for Surry because the containment loads assessed for NUREG-1150 were not a great threat to the containment survivability. An updated evaluation of the impact of intentional depressurization on the probability of having a high pressure melt ejection was then made that reflected analyses that have been performed since NUREG-1150 was completed. The updated evaluation confirmed the sensitivity study conclusions that intentional depressurization has minimal impact on the probability of a high pressure melt ejection. The updated evaluation did show a slight benefit from depressurization because depressurization delayed core melting, which led to a higher probability of recovering emergency core coolant injection, thereby arresting the core damage.

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