NUREG-4956

DRAFT FOR REVIEW

CONTAINMENT EVENT ANALYSIS AND ESTIMATION OF SOURCE TERM FREQUENCIES

by

A. S. BENJAMIN, V. L. BEHR, S. E. DINGMAN,
J. M. GRIESMEYER, F. E. HASKIN, D. M. KUNSMAN,
and W. B. MURFIN

SANDIA NATIONAL LABORATORIES
ALBUQUERQUE, NEW MEXICO 87185

MARCH 22, 1985

8508220035 850322
PDR NUREG
0956 C PDR

Correspondence concerning this nureg rpt. Identify Nureg No on File Lead

TABLE OF CONTENTS

1. INTRODUCTION

The intent of this document is to provide a limited risk perspective for the fission product source terms reported in the BMI-2104 documents.

One might ask, "Why is there a need for a risk perspective?" To answer the question, let us consider as an example, the BMI-2104 analysis of the sequence S_2D at Surry. Two calculations were made. In the first, a break of a specified size (2-inch diameter) was presumed to occur in the cold leg through the reactor coolant pump seals, and the primary system coolant was presumed to end up in the containment sump. The containment was assumed to fail from rapid pressurization due to events occurring just after meltthrough of the reactor vessel, and the containment sprays were assumed to fail at that time. The release from containment was assumed to bypass the auxiliary building.

In the second, a break of the same size was assumed to occur in the hot leg piping. Both the containment and the containment sprays were assumed to survive the events following vessel meltthrough, but the water flow to the reactor cavity was assumed not to prevent the core-concrete interaction from occurring. Containment remained intact until the calculation was terminated.

While the events assumed in the BMI-2104 calculations are plausible, one might ask whether they are the most likely series of events that could occur following an S_2D, or whether they produce the highest source terms that could occur within the realm of reasonable probability. Is it more likely, for example, that the size and location of the break would be different from what was assumed in BMI-2104; that the containment sprays would fail prior to containment failure, because of debris in the containment sump plugging the pump intakes; that containment might fail by some means other than early overpressurization; that the release pathway would be through the auxiliary building, where further reduction of the source term would take place?

These questions raise the need for a systematic identification of the various pathways that the accident can take and assessment of the likelihood, or probability, of each. Such an analysis provides a basis for evaluating whether the source terms developed for a particular accident sequence cover the range of risk-significant source terms for that accident sequence.

In addition, it is important to recognize that the BMI-2104 analyses, by virtue of limited time and funding,

did not consider all the accident sequences that are thought to be potentially important to risk. They did not treat, for example, station blackout accidents with delayed failure of high pressure coolant injection (TB) for either of the two BWR containments, Peach Bottom and Grand Gulf, whereas the Accident Sequence Evaluation Program (ASEP) identified this accident as being potentially important from a probabilistic viewpoint. To determine whether the BMI-2104 source terms cover the range of significance in risk space, it is necessary also to address accident sequences not treated in BMI-2104.

The overall objectives of this study are (1) to identify accident pathways (i.e., combinations of accident sequences and containment events) that delineate source terms which may be important to risk, (2) to estimate the frequencies of those pathways and hence the frequencies of the source terms they attend, (3) to ascertain how well the BMI-2104 source terms cover the accident pathways that are important to risk, and (4) to identify accident pathways for which additional source term calculations are needed. These objectives extend to the five reference plants addressed in BMI-2104: Surry (PWR subatmospheric containment), Zion (PWR large, dry), Peach Bottom (BWR Mark I), Grand Gulf (BWR Mark III), and Sequoyah (PWR ice condenser).

In Section 2, we will describe the method we used to achieve the objectives stated just above. First we will provide a general description of our containment event trees and the procedure we use for quantifying the branches. Then we will provide a detailed example, illustrating the application of the method for a particular accident sequence in one of the reference plants. The example will show how we used information from recently developed sources to quantify the containment event tree. Then we will discuss some of the special considerations we made for the other accident sequences and reference plants. In Section 3, we will present the results for all the sequences and plants we analyzed. In Section 4, we will summarize the results and indicate accident pathways for which additional source term calculations are needed.

As a final comment, it is important to differentiate between the objectives stated above and those of a risk assessment. We do not calculate risk here, only the frequencies of source terms. Evaluation of risk requires two additional steps: (1) estimation of source terms for important sequences and accident pathways not treated in BMI-2104, and (2) determination of the mean consequences associated with each source term. These objectives are part of the Severe Accident Risk Reduction Program (SARRP), which will use the results reported here to calculate risks for the references plants. That analysis is scheduled to be completed and documented by end of 1985.

2. METHOD

2.1 General Description and Observations

In traditional probabilistic risk assessments (PRAs), the accident pathways that contribute to risk are described by two types of event trees. "System event trees" are used to define the spectrum of accident sequences (i.e., the combinations of accident initiators and subsequent system failures) that can lead to core melting. "Containment event trees" are used to define the containment failure modes which lead to fission product releases beyond the containment boundary.

In our analyses, we take the accident sequences to have been previously defined by the existing PRAs, and we obtain estimates of their frequencies of occurrence from the Accident Sequence Evaluation Program (ASEP). The sequence frequencies are provided in the form of central estimates, first for the plants as configured in the original PRAs, and then for the plants as they currently exist. The second set of ASEP estimates includes consideration of plant changes that are planned but not yet implemented. The basis for these estimates are given in another appendix.

Our primary focus is upon the containment event trees. We have developed a general containment event tree for each reactor plant analyzed in this study. We apply the event tree to particular sequences by selecting the appropriate branches within the event tree.

Our containment event trees are considerably expanded beyond those considered in previous PRAs. We ask the following types of questions:

(1) <u>Phenomenological events.</u> What are the phenomena that could affect the progression of severe accidents, at what points during the accident timeline do they occur, and what are their subsequent effects on the accident development?

(2) <u>Reactor coolant system failure modes.</u> What is the size and location of the reactor coolant system breach and the pressure in the system at the time of breach?

(3) <u>Containment system survivability.</u> Do the containment sprays, fan coolers, and suppression systems survive the conditions occurring during severe accidents that exceed their design bases?

(4) <u>Containment failure modes.</u> What are the loads that challenge containment, does containment survive these loads, what is the nature of the failure

(approximate size and location), and what is the subsequent pathway for fission product release to the environment?

The questions on the containment event trees are posed in ways that require the answers to be expressed in terms of likelihoods. For a loss-of-coolant accident in a PWR, for example, we might ask how likely it is that the reactor coolant system breach will be in the cold leg piping as opposed to the hot leg piping; or how likely that containment will fail due to a hydrogen burn following reactor vessel failure. Answers to such questions require information about the reactor design, the phenomenology of reactor accidents, and the capabilities of containment. For example, to answer the two likelihood questions just posed, one would need to know about the characteristics of the cold leg versus hot leg piping, the amount of hydrogen generated prior to vessel breach, the availability of ignition sources, and the failure pressure of the containment.

Some of the issues addressed by the containment event trees are listed in Table 2.1. We point out that these are not the events themselves, but rather some of the issues that must be addressed in order for the event trees to be quantified. Also shown is an indication of the issues that have been considered in some of the recent PRAs as a basis for defining fission product release categories. Observe that none of the PRAs account for all the issues we consider for binning source terms, but the most recent one (Seabrook) accounts for more than the others.

We have utilized a large number of sources to obtain the needed information, including the following:

(1) Containment Loads Working Group (CLWG), References 1-3.

(2) Containment Performance Working Group (CPWG), Reference 4.

(3) Battelle calculations for Accident Source Term Project Office (BMI-2104), Reference 5.

(4) Quantitative Uncertainty Estimation for the Source Term (QUEST), Reference 6.

(5) Industry Degraded Core (IDCOR) program, Reference 7.

(6) Severe Accident Sequence Analysis (SASA) program, References 8-12.

(7) Severe Accident Uncertainty Analysis (SAUNA), Reference 13.

(8) Accident Sequence Evaluation Program (ASEP), References 14-15.

(9) SARRP Phenomena Assessment Task Force (PATF), Reference 16.

(10) Available probabilistic risk assessments (PRA), References 17-22.

(11) Final Safety Analysis Reports (FSAR), Reference 23.

(12) Architect-engineer (AE) and other estimates of containment failure pressure, References 24-25.

(13) Filtered-Vented Containment System (FVCS) reports, References 26-27.

(14) Others, References 28-36.

We encountered several difficulties in attempting to utilize information from these various sources. One of the biggest was incomplete coverage. Table 2.1 illustrates the relationship between some of the issues addressed by the event trees and the information provided by the two containment working groups, the two ASTPO studies, and the IDCOR program. It is clear from the table that the results from these studies address only a fraction of the questions asked.

When a question was addressed by one of the studies, the information provided often required us to make extrapolations. For example, the Containment Loads Working Group provided estimates of the size of steam spikes and direct heating for only the large-dry PWR reference plants, and then with preconditions appropriate for only one accident sequence. We had to extrapolate this information to other plants and other sequences. The same was true for the analyses of global hydrogen burns, diffusion flames, and containment temperatures achieved from core-concrete interactions. Similar statements apply to other studies.

Furthermore, the information provided to us often did not include a best estimate but rather a range of possible values. In particular, the CLWG provided low, medium, and high estimates for containment loading, whereas the CPWG provided only high estimates for containment leakage. In the CLWG, consensus was generally reached more often on the low and high estimates than on the medium estimates.

When we quantified our containment event trees, therefore, we propagated three separate estimates -- optimistic, central, and pessimistic. Thus, we derived three sets of accident outcome probabilities for each sequence. The one labeled "pessimistic" tends to provide

higher probabilities for the pathways that lead to higher source terms and lower probabilities for the lower source term pathways. The ones labeled "central" and "optimistic" are analogously interpreted. We do not propose weighting factors or distributions for these estimates, nor purport that one is better than another. Rather we present them as a reflection of the information that is available. (Please refer to Section 4. for additional comments on our interpretation of the words "optimistic", "central", and "pessimistic".)

Finally, our containment event analysis is based on the plant configurations that currently exist. We have not attempted to account for plant changes that are planned but not yet implemented. For example, in our analysis for Peach Bottom, we account for the symptomatic emergency procedures that include containment venting, alternate injection sources, and procedures for anticipated transients without scram. We do not credit the planned change in logic that would bypass the high drywell pressure permissive for actuation of the automatic depressurization system. Basically, our reference plant configurations are consistent with IDCOR's "baseline" profiles, as opposed to their "committed" profiles.

It is worth noting again that we have not calculated source terms, but rather the frequencies of accident pathways that can lead to distinct source terms. One can sometimes qualitatively judge that certain pathways are similar enough in character to permit them to be binned to the same source term. Often, however, this is not the case.

2.2 Specific Example: Surry S_2D and S_3D

We will illustrate the application of the method by providing the details for two accident sequences which are similar in many respects but dissimilar in some - Surry S_2D and S_3D. These are sequences which are initiated by small reactor coolant system breaks (in one case a pipe break, in the other a pump seal leak). The emergency core cooling system fails to inject water onto the core. Containment sprays are operative as the accident develops toward a core meltdown.

The general event tree for large, dry PWR containments includes a total of 59 questions. These questions cover all the accident sequences which we analyzed for the Surry and Zion plants. Table 2.2 provides a list of the questions together with references to prior questions upon which the answers depend, in effect giving the structure of the event tree.

About half of the questions in Table 2.2 are applicable to the S_2D and S_3D sequences in Surry. For those

questions, we have provided in Table 2.3 an itemization of the branches that are applicable and of the corresponding answers that we assigned for the specific sequences being evaluated. It should first be observed that for some of the questions there are more than two branches. In these cases, the answers are not expressed as "yes" or "no", but rather as a choice among possible break sizes, break locations, or degrees of damage.

Next it should be noticed that some of the answers are expressed numerically and others are expressed verbally. This distinguishes the fact that for some questions the available information is sufficient to make quantitative estimates of likelihood, while for others the data supports only qualitative likelihood descriptors. Ultimately, we will assign numerical values to the qualitative descriptors in order to evaluate source term frequencies. However, we will recognize that this assignment of numbers is highly subjective and will accordingly evaluate the sensitivity of the results to the numerical choices. For the present, we need not be concerned about this aspect of the work; it will be discussed further in Sections 2.3 and 3.6

For many of the questions addressed in Table 2.3, information is needed about containment loading and containment capacity. In Table 2.4, we provide the results of our evaluations of containment loading for various accident situations. We give containment capacities (both for structural failure and for induced leakage) in Table 2.5.

The remainder of this subsection provides the rationale for our assignment of values in Tables 2.3, 2.4, and 2.5:

<u>Question 3: Likelihood of Various Sizes of Preexisting Containment Leakage or Isolation Failure</u>

We considered four levels of containment leakage, as follows: (i) leakage within technical specifications, (ii) leakage greater than technical specifications but insufficient to preclude gradual overpressurization later in the accident, (iii) leakage insufficient to depressurize containment in about 2 hours, and (iv) leakage sufficient to depressurize containment in about 2 hours or less. Based on MARCH code calculations (BMI, Ref. 5), the corresponding leakage areas for each of these levels are (i) 0-.004 in^2, (ii) .004-4.0 in^2 (iii) 4.0 in^2 - 1.0 ft^2, (iv) >1.0 ft^2.

Weinstein (NRR, Ref. 28) studied the frequency of observed leaks at nuclear power plants based on licensee event reports. We performed an analysis of his data using a binomial distribution. This yielded 5th, 50th, and 95th percentiles of .017, .033, and .058, respectively, for Level (ii) leaks and 4×10^{-5}, 5×10^{-4}, and 2×10^{-3},

respectively, for Level (iii) leaks. Most of Weinstein's data, however, are for plants which are operated at atmospheric pressure. Preexisting leaks at subatmospheric containments are limited to the capacity of the vacuum pumps (.07 in^2). Therefore, for Surry we have excluded all observations of preexisting leakage areas greater than .07 in^2.

The RSS (Ref. 17) estimated a probability of containment isolation failure of $2x10^{-3}$. The major contributor was inadvertant opening of a containment purge valve with coincident failure of the radiation interlock. RSSMAP (Ref. 18) increased the probability to 7×10^{-3} to reflect concerns about a recently discovered large leak at another plant. On the other hand, IDCOR (Ref. 7) estimated coincident opening of double purge valves to be no greater than 10^{-6}. We have taken the probability of isolation failure to be negligible (optimistic) or the RSS value (central and pessimistic).

We recognize that the probability of containment isolation failure could be a function of ac power availabiliy. However, we did not have sufficient information to quantify the dependence. Further, we recognize that external events, such as earthquakes, could cause the likelihood of initial containment leakage or bypass to be higher than what we have estimated. We did not attempt to quantify that effect, likewise, because there does not exist an external event analysis for the plant in question.

We recently received a draft report by Pacific Northwest Laboratory (PNL), entitled "Reliability Analysis of Containment Isolation Systems." This report provides a detailed evaluation of applicable licensee event reports and integrated leak rate test reports. We have not had time to incorporate PNL's results into our event trees or to study them in any detail. It appears, however, that PNL's results imply a higher frequency for Level (ii) leaks than what we present and a commensurate frequency for Level (iii) leaks.

Question 4: Likelihood of Various Locations for the Initial Reactor Coolant System Break

Examination of P&ID's showed approximately equal numbers of pipe connections in the range of 1-2 inches on the hot and cold legs. However, the cold legs are penetrated by a large number of instrumentation tubes. We have therefore estimated that a pipe break in the S_2 size range is "likely" to be in a cold leg rather than a hot leg. S_3 LOCAs (.5 - 1 inch equivalent) have been taken to be pump seal failures and are hence always on the cold leg.

Question 9: Likelihood of Containment Spray Failure to Actuate Before Core Degradation

At a subatmospheric plant like Surry, where the fan coolers are non-emergency and trip on safety actuation, the containment spray set point of 10.3 psig is reached for small breaks in about 1 to 2 hours (BMI, Ref. 5). Only when there is a large preexisting containment leak or isolation failure will the spray setpoint not be reached prior to core melting. Hence, for S_2D and S_3D, we took spray actuation to be a certainty, unless there was a large preexisting leak or isolation failure.

Question 11: Likelihood of Various Levels of Auxiliary Building Breakthrough or Bypass, Given Preexisting Containment Leakage or Isolation Failure

The RSS (Ref. 17) allowed no credit for deposition of fission products in the auxiliary building, whereas BMI (Ref. 5) allowed full use of the auxiliary building for holdup of fission products released through an isolation failure. We have also considered the possibility that only part of the auxiliary building would be effective, either because the release occurs in the upper, more open part of the building, or because the blowdown from containment damages the building.

Preexisting leaks or isolation failures in the larger range (Level iii) are considered "likely" to pass through the auxiliary building, because most containment penetrations lead into the auxiliary building. The only major exception would be the complete opening of a purge line, which would cause a total bypass of the auxiliary building. Such an event, however, was assessed by IDCOR (Ref. 7) to be extremely unlikely.

Question 12: Likelihood of Temperature-Induced Failure of the Reactor Coolant System in Various Locations

Temperature-induced failures of the reactor coolant system have not been considered in most risk assessments (Ref. 17-22). However, recent analyses (Ref. 31) show pressurizer gas temperatures as high as 2000 F for the TMLB' sequence, and similarly high temperatures are expected for S_3D. RCS pressure is expected to be at the pilot-operated relief valve set point for both TMLB' and S_3D (BMI, Ref. 5; IDCOR, Ref. 7; SASA, Ref. 8). Based on these observations, CLWG and others have raised the possibility of induced failure in some part of the RCS. IDCOR (Ref. 7) has assumed that the induced failure is a seal LOCA, which would convert an S_3D sequence to an S_2D sequence. Full rupture of a hot leg is also considered possible.

Induced failure is optimistic, in that later direct heating scenarios can be prevented or mitigated. We have

taken induced failure to be "certain" (optimistic), "indeterminate" (central), or "unlikely" (pessimistic), consistent with the observations made above.

We also considered the possibility of an induced steam generator tube rupture (SGTR) for the central and pessimistic walkthroughs. The Zion-Indian Point Study (Ref. 32) raised the question of induced SGTR, and there have been reports of SGTRs even without the high pressures and temperatures expected in sequence S_3D. On the other hand, IDCOR (Ref. 7) considered SGTR to be "not risk significant." We have estimated the probability of an induced SGTR to be "impossible" (optimistic), "remotely possible" (central) or "unlikely" (pessimistic), based on these observations.

<u>Question 13: Likelihood of Various Sizes of Induced RCS Failure</u>

Rupture of a hot leg would be a large break. Temperature-induced seal failures and SGTRs would be equivalent to a S_2 LOCA.

<u>Question 17: Likelihood of Containment Spray Failure to Actuate After Induced LOCA</u>

The logic for answering this question is similar to that given for Question 9.

<u>Question 21: Likelihood of Occurrence and Magnitude of a Hydrogen Burn Before Vessel Breach</u>

The likelihood of a hydrogen burn before vessel breach depends upon whether a combustible mixture develops and whether an ignition source is present.

The hydrogen in containment will not be combustible if the containment is steam inerted or the hydrogen concentration is less than the lean combustion limit. For S_2D and S_3D, the operation of the containment sprays precludes steam inerting. For the atmosphere to be combustible, however, an amount of hydrogen equivalent to about 40% Zircaloy oxidation must be in containment.

The accumulation of hydrogen in containment before vessel breach depends upon two factors - the amount that is produced in-vessel and the fraction of that amount which escapes to containment. Let us consider each of those elements separately.

The RSS (Ref. 17) estimated that hydrogen produced before vessel breach would be equivalent to 75% ± 25% Zircaloy oxidation. CLWG and SASA calculations for a different Westinghouse reactor (Sequoyah) ranged in in-vessel hydrogen generation from 35% to 100% Zircaloy

oxidation (Ref. 2 and 9). IDCOR calculated about 25% Zircaloy oxidation in-vessel for S_2D in Sequoyah. A separate CLWG submittal (Ref. 3) used a lower bound of 25% and upper bound of 100% for atmospheric and subatmospheric PWR containments. BMI-2104 calculations for Surry S_2D showed about 50% oxidation occurring during core heatup and another 10% to 40% occurring during core slumping, amounting to a total of 60% to 90%. The SARRP Phenomena Assessment Task Force (Ref. 16) set the total lower and upper bounds at 10% and 100%, respectively, for a TMLB' accident in Surry. SNLA (Ref. 33) has estimated that steel oxidation could add another 50%. Given this variety of possible choices, we selected 25% Zircaloy oxidation (400 lb. hydrogen) as our optimistic estimate, 50% (800 lb.) as our central estimate, and 100% (1600 lb.) as our pessimistic estimate.

The fraction of these amounts escaping from the RCS before vessel breach depends on the break size. On the basis of MARCH calculations (BMI, Ref. 5; SASA, Ref. 8), we estimated a 75% escape fraction for large breaks, 50% for S_1, and 25% for S_2 and S_3. We also estimated (from MARCH calcula- tions) that isolation failure or large containment leaks would reduce the amount in containment by 25% (large breaks), 15% (S_1) and 10% (S_2, S_3). From these results, we concluded that hydrogen burns before vessel breach could occur for S_2D and S_3D only in the event of an induced large LOCA with no large preexisting containment leakage or isolation failure, and then only for the pessimistic walkthrough.

Because hydrogen burns before vessel breach can only occur in the pessimistic walkthrough and because it is more pessimistic to have the burn occurring at vessel breach than before vessel breach, we have estimated hydrogen burns before vessel breach to be "unlikely".

<u>Question 22: Likelihood of Containment Failure from Hydrogen Burn Before Vessel Breach</u>

The RSS (Ref. 17) estimated the mean failure pressure of the Surry containment structure to be 85 psig, based on an assessment that the most probable failure mechanism was tearing of the liner. A standard deviation of 15 psi was assigned to this estimate. More recently, Stone and Webster (Ref. 24) calculated a failure pressure of 119 psig, corresponding to general yielding of the reinforce- ment. They gave no estimate of uncertainty; however, an analogous estimate for the Zion containment (Ref. 19) produced a standard deviation of about 2.5 psi. This latter standard deviation accounted for material property uncertainties but not for uncertainties in the modeling of the structural response or for possible "as-built" structural deviations from design.

For our pessimistic estimate, we used the RSS failure pressure of 85 psig and standard deviation of 15 psi and assumed a normal distribution. For the optimistic estimate we used the Stone and Webster failure pressure of 119 psig and the Zion standard deviation of 2.5 psi. For the central estimate, we combined the Stone and Webster failure pressure of 119 psig with the RSS standard deviation of 15 psi. For rapid pressurizations caused by hydrogen burns, we assumed that the containment would fail structurally as opposed to developing leaks (CPWG, Ref. 4).

To obtain failure likelihoods, we evaluated the pressure increments associated with burning of the amounts of hydrogen specified in Question 21. The pressure before the burn was established from BMI calculations (Ref. 5), which incidentally agree quite well with calculations by IDCOR and SASA. Tables 2.4 and 2.5 summarize the containment loading and capacity estimates leading to the evaluation of the failure likelihoods.

Question 23: Likelihood of Various Levels of Auxiliary Building Breakthrough or Bypass, Given Containment Failure from a Hydrogen Burn Before Vessel Breach

As discussed for Question 11, we considered three levels of auxiliary building bypass. The auxiliary building at Surry covers approximately 10% of the containment sidewall (FSAR, Ref. 23). However, we estimated the most probable point of gross containment failure following a hydrogen burn to be at the intersection of the cylinder and upper dome, i.e., above the auxiliary building. For gross containment failure we believed that complete bypass of the auxiliary building would be "likely" (optimistic), "almost certain" (central), or "certain" (pessimistic).

Question 25: Likelihood of Containment Spray Failure, Given a Hydrogen Burn Before Vessel Breach

If the spray set point had not been reached previously (Question 17), an early hydrogen burn would definitely actuate them. On the other hand, a hydrogen burn could also cause the sprays to fail. To address this question, the case where containment survives the hydrogen burn must be considered separately from the case where containment fails.

If no containment failure occurred, it would still be possible for the pressure-temperature pulse of a hydrogen burn to cause spray failure by damaging instrumentation or burning insulation. However, it is to be noted that containment sprays survived a hydrogen burn at TMI-2 and operated normally. We estimated spray failure, given hydrogen burn without containment failure, to be "remotely possible."

The RSS (Ref. 17) assumed that sprays operating in the recirculation mode would inevitably fail following containment failure. The rationale was that the containment sump would become saturated as containment pressure fell and that the pumps would fail because of cavitation. More recent research has shown this assessment to be unduly pessimistic. Containment pressure is expected to decline gradually for the size of openings expected (1-2 ft^2), allowing the sump to become stabilized after evaporation of a small fraction of the contents. Research conducted under Task Action Plan A-43 (Ref. 30) has also shown that pumps can operate for an extended period with net positive suction head well below the design requirement. IDCOR (Ref. 7) also considers spray success to be likely.

We estimate spray failure following containment failure to be "unlikely" for the optimistic and central cases, based on A43 and IDCOR. For the pessimistic case, we considered spray failure to be "likely" in accordance with the more pessimistic RSS judgement.

<u>Question 27: Likelihood of Various Modes of Vessel Breach</u>

The RSS (Ref. 17) assumed that the reactor pressure vessel would fail either as a result of a large steam explosion or by a meltthrough of the skirt allowing the hemispherical head to detach. The ZIP study (Ref. 32) pointed out that other failure modes were equally or more likely. ZPSS (Ref. 19) and IDCOR (Ref. 7) argued that the most probable point of failure would be a local failure at a core instrumentation tube, and that the core debris would emerge as a jet in high pressure sequences or would dribble out in low pressure sequences.

We considered four modes of reactor vessel failure: a steam explosion failing the upper head, a steam explosion failing the lower head, a high pressure ejection, and a low pressure meltthrough.

The probability of a steam explosion failing the upper head and driving the upper head through the containment dome was estimated in the RSS to be 0.01. More recent estimates (for example ZPSS and IDCOR) have disputed this estimate as being unrealistic, or as being physically impossible at high pressures. SAUNA (Ref. 13) reported that triggering an explosion at high pressures might be more difficult, but that the probability of an adequate trigger being available was unknown. We have accordingly discounted the supposed high-pressure limitation.

Berman et al. (IVSE, Ref. 34) conducted a Monte Carlo study of the probabilities of steam explosions causing failure of the reactor vessel, and for those cases where upper head failure occurred, the subsequent probability of

a large missile with sufficient velocity to fail containment. Several geometric and physical parameters used in the study were uncertain and were given uniform distributions which were sampled by the Monte Carlo procedure. To assess the sensitivity of the results to the assumed distributions, three different distributions were postulated for each uncertain parameter. These were labeled "low", "middle", and "high." For example, the three distributions for conversion ratio for thermal to kinetic energy were 0-1.7%, 1.7-3.3%, and 3.3-5.0%.

For our central estimate, we used the results of Berman et al. based on "middle" distributions for all the uncertain parameters. Out of 10,000 samples, it was determined that 2126 resulted in failure of the bottom head, while only one led to a failure of the upper head. That one also created a missile with velocity judged sufficient to threaten containment. The probability of one in 10,000 for α-mode containment failure is consistent with ZPSS and other industry-sponsored risk assessments.

For the pessimistic estimate, we took an average of the results obtained by Berman et al. when one parameter was assigned its "high" distribution and the others were kept at their "middle" distributions. Since five parameters were varied in this fashion, the result corresponded to the average of 5 cases, each involving 10,000 samples. The failure fractions were .47 (bottom head) and .014 (upper head), with all the upper head failures creating a missile judged sufficiently energetic to threaten containment. The latter figure is consistent with the RSS.

For the optimistic estimate, we took the lowest of the results obtained by Berman et al. when one parameter was assigned its "low" distribution and the others were kept at their "middle" distributions. This resulted in no failures. (If we had taken the average of the five "low" cases, we would have obtained a .05 probability of bottom head failure and zero probability of upper head failure.)

All high or intermediate pressure sequences other than in-vessel steam explosions were assumed to give pressurized jet ejections. All low pressure sequences other than in-vessel steam explosions were assumed to give a passive meltthrough.

The results discussed above must be qualified in light of a recent experiment by Berman et al. in which the conversion ratio was estimated to be much higher than the 3.3-5.0% "high" distribution quoted above. Since this is the only experiment to date in which the explosion has been intentionally confined, the results of this experiment are noteworthy and may need to be factored into the analysis. It should be noted that based on Ref. 34, a conversion

ratio distribution of 5.3-10.7% would lead to an upper head/α-mode failure probability of 0.4. Although our tables do not reflect this experiment, because of the preliminary nature of the data, we wish to leave open the possibility of higher failure probabilities due to in-vessel steam explosions.

Question 28. Likelihood of Occurrence and Magnitude of Direct Heating/Steam Spike at Vessel Breach

The core debris ejected from the reactor vessel after vessel breach can be quenched either by water or by direct heat transfer to the atmosphere. Both can produce a pressure spike, the former by generating steam (hence the term "steam spike") and the latter by adding thermal and/or chemical energy directly to the atmosphere (hence the term "direct heating"). The two phenomena are linked because the occurrence of one subtracts from the amount of core debris energy available for the other.

Because direct heating is a basically unexplored issue which can have a large effect on our results, we considered two cases in our analyses. In the first case, we excluded direct heating and included only steam spikes. In the second case, we allowed for both to occur concurrently.

Both CLWG (Ref. 2) and IDCOR (Ref. 7) agreed that at least a fraction of the core will be involved in a steam spike if water is present. In Surry S_2D and S_3D, a large amount of water will exist both in the reactor cavity and on the containment floor because of the continuous operation of the containment sprays before vessel breach. (If this were not enough, sufficient water would be provided by the portion of accumulator water not discharged prior to vessel breach, which would discharge into the reactor cavity after vessel breach.) Hence, we took a steam spike to be assured and focused upon its size.

IDCOR calculated a steam spike of 15 psi (1.0 bar) for the S_2D sequence, assuming that 50% of the core debris quenched. CLWG (Ref. 2) reported that the spike from core debris quenching in water could be nil or as high as 26 psi (1.9 bar) corresponding to quenching of 100% of the core debris. The CLWG figures were obtained without consideration of containment cooling; however, BMI-2104 calculations indicated that the effect of sprays on the steam spike would be small if one assumed, for the pessimistic case, that the debris was highly fragmented. Thus, we took the size of the steam spike to be 0 psi (optimistic), 15 psi (central), and 26 psi (pessimistic).

Direct heating may occur if the reactor vessel fails at high pressure, causing a highly aerosolized jet of core debris, and if the geometry of the reactor cavity area is such that there is a pathway for the aerosolized debris

to disperse into the containment atmosphere. We postulated that direct heating can also occur as a result of a steam explosion that fails the upper head and provides a direct pathway to the upper containment. This second possibility, however, did not play a significant role in our final results.

IDCOR discounted the occurrence of direct heating by arguing that the pathways were too tortuous for the core debris to be dispersed rapidly into the containment atmosphere. CLWG, on the other hand, was sharply divided on the issue. One subgroup (Group A) maintained that less than 2% of the core debris could participate in direct heating, while fully 80% might participate in a steam spike. The other subgroup (Group B) had as much as 50% of the core debris participating in both thermal and chemical heating of the atmosphere, with the remainder of the core debris contributing to a steam spike.

The data we used to generate estimates of pressure increments from concurrent direct heating and steam spike are provided in Table 2.6. The first entry is the fraction of core melt material ejected from the reactor vessel at the time of vessel breach. For this question, we took 50% melt ejection (IDCOR) to be optimistic, 100% melt ejection(BMI) to be pessimistic, and 75% to represent a central estimate. (CLWG did not provide estimates for the quantity of melt ejected; it was instead stipulated as part of the standard problem definition.) The remaining entries were obtained from CLWG. The "optimistic" values are from Subgroup A, which provided only a single estimate, whereas the "central" and "pessimistic" values correspond to Subgroup B's "medium" and "high" estimates. We did not use Subgroup B's "low" estimate in this analysis, since it would not have been characterized as "optimistic" by Subgroup A nor as "central" by Subgroup B.

<u>Question 29: Likelihood of Occurrence and Magnitude of Hydrogen Burn at Vessel Breach</u>

The total amount of hydrogen generated before vessel breach is assumed to be released to containment at the time of breach, if it has not previously been released. If hydrogen has burned before vessel breach, a burn is considered impossible at this time because the hydrogen concentration would be less than the lower combustion limit. If there is no containment heat removal, the atmosphere would be steam inerted and a burn would likewise be impossible.

If the atmosphere is flammable, many experts (CLWG, BMI-2104, SASA) consider a hydrogen burn to be highly probable when the core debris is first discharged from the reactor vessel. The ignition source is the hot core debris

itself. Others (IDCOR) contest this supposition on the basis that the interaction would produce such large amounts of steam as to inert the atmosphere locally. The possibility of local inerting has been addressed by making the burn probability "likely" rather than certain.

The quantity of hydrogen produced for the three walkthroughs has been addressed in the discussion for Question 21. According to a CLWG submittal (Ref. 3), the containment pressure increments corresponding to these amounts of Zircaloy oxidation would be 20 psi (optimistic), 35 psi (central), and 70 psi (pessimistic), as listed in Table 2.4. The same submittal showed that the effect of sprays on the pressure increment would be small if the burn time were equal to that which occurred during the TMI-2 accident (i.e., about 8 seconds).

Question 30: Likelihood of Containment Failure from Steam Spike, Direct Heating, and/or Hydrogen Burning at Vessel Breach

The containment loading at the time of vessel breach results from the accumulation of pressure increments due to depressurization of the reactor vessel, steam spiking, direct heating, and/or hydrogen burning, all of which may occur within a short period of time. Except for the increment due to vessel depressurization, the pressure increments for the various contributors have been discussed in Questions 28 and 29 and are summarized in Table 2.4.

For vessel depressurization, BMI (Ref. 5), IDCOR (Ref. 7), SASA (Ref. 8), and CLWG (Ref. 2) show containment pressure increase of 8-15 psi for high pressure sequences such as S_3D. Smaller pressure rises are seen in S_2D, and essentially none for large breaks. This information is also summarized in Table 2.4.

Containment failure pressures have been discussed under Question 22. For rapid pressure spiking events, we assumed that the containment would fail structurally rather than by a more benign leakage mode. We evaluated the likelihood of failure for each walkthrough by comparing the pressure loading with the failure pressure, assuming a normal distribution for the latter.

Question 32: Likelihood of Various Levels of Auxiliary Building Breakthrough or Bypass, Given Containment Failure at Vessel Breach

For α-mode failures, a complete bypass is assumed, because the containment failure must be in the dome. Otherwise, the logic is identical to Question 23.

Question 34: Likelihood of Containment Spray Failure After Vessel Breach

In addition to the possibility of failure by a hydrogen burn damaging instrumentation or insulation or by cavitation due to containment depressurization (Question 25), containment sprays could fail because of debris in the sump clogging the screens or passing through the screens and damaging the pumps. One of the sources of debris might be an energetic fuel-water interaction that sweeps core debris, tubing, ductwork, and insulation out of the reactor cavity. Even without a steam spike, some debris could be expected. It may be observed, for example, that the sump water at TMI-2 was laden with particulate matter.

If there is no containment failure, we estimated containment spray failure after vessel breach to be "remotely possible" (optimistic), "unlikely" (central), and "indeterminate" (pessimistic), taking into account the results of TAP A-43 research (Question 25). In the event of containment failure, spray failure is given the same likelihoods as in Question 25.

Question 35: Likelihood of an Oxidation Release

Either direct heating or any steam explosion that fails the reactor vessel is judged to give an oxidation release.

Question 42: Likelihood of Significant Core-Concrete Interactions after Vessel Breach

Significant core-concrete interactions (i.e., those which could result in vaporization releases from the fuel) may be precluded by either of two occurrences. First, if there is a sufficient amount of water in the reactor cavity, there is a possibility that a coolable debris bed will form. Second, if there is an event which disperses the core debris, such as a high pressure ejection or an in-vessel steam explosion which fails the reactor vessel, the core debris may be scattered so sparsely through containment that significant concrete attack cannot occur. Conversely, if there is insufficient water and no dispersive event, the core debris will mostly remain in the reactor cavity, the debris bed will dry out, and a core-concrete interaction will occur with high certainty.

For S_2D-type accidents, ZPSS (Ref. 19) and IDCOR (Ref. 7) expected a coolable debris bed to be formed as long as water could be provided ("top-down quench"). SAUNA (Ref. 13) regarded coolability to be uncertain, because the particle size was unknown. BMI (Ref. 5) has performed calculations with and without a coolable debris bed with very different results.

The uncertainty of the phenomenology and the paucity of experimental evidence led us to consider top-down quench to

be somewhat less certain than the ZPSS or IDCOR, more in line with the SAUNA estimate. For a benign meltthrough with water in the reactor cavity, we took the probability of core-concrete interactions to be "indeterminate" (optimistic), "likely" (central), or "almost certain" (pessimistic).

For an intermediate pressure ejection, we expected moderate dispersal and we estimated the probability of core-concrete interactions to be "unlikely" (optimistic), with the other walkthroughs unchanged.

For a high pressure ejection or steam explosion, we expected thorough fragmentation and dispersal, and we estimated the probability of core-concrete interactions to be "remotely possible" (optimistic), "unlikely" (central), or "indeterminate" (pessimistic).

<u>Question 44: Likelihood of Large Containment Leakage and/or Structural Failure from Pressurization Due to Gradual Boiloff of Water in the Reactor Cavity</u>

If there is a coolable debris bed and no containment heat removal (i.e., no late containment sprays), overpressurization is expected to occur due to vigorous steaming (ZPSS, Ref. 19; CLWG, Ref. 1). If the debris bed is not coolable but is covered by an overlying water layer and there is no containment heat removal, the pressure achieved in containment depends upon the degree to which heat is transferred upward to the water versus the rate at which steam is condensed via natural heat sinks in containment. MARCH calculations (extrapolated from BMI, Ref. 5) indicate a peak containment pressure of about 70 psig for that case.

We considered three levels of containment failure, as follows: (i) no failure, (ii) leakage sufficient to preclude gradual overpressurization but insufficient to depressurize containment within a period of about two hours, and (iii) gross structural failure causing depressurization in less than two hours. To evaluate the likelihood of each, we relied upon information from various sources.

The Containment Performance Working Group (Ref. 4) provided "upper bound" estimates of leak area versus pressure for all of the ASTPO reference plants, and of leak area versus time-at-temperature for one of the plants (Peach Bottom). For Surry, the primary source of leakage was the personnel air lock. The pressure-dependent leak areas are listed in Table 2.5. Although CPWG did not provide "lower bound" or "central" estimates, there is an implication that the leak areas for these cases would be negligible.

Even the "upper bound" leak areas for Surry are very small compared to those which would be required to preclude

structural failure from gradual pressurization. The leak area which would preclude overpressure failure is about 20-26 in^2 if one does not consider internal heat sinks and condensation on containment walls. With consideration of heat sinks and condensation, MARCH runs have sometimes shown failure precluded with leaks as small as 4-6 in^2 (SASA, Ref. 8; BNL, Ref. 29). If a minimum of 4 in^2 is considered necessary to preclude overpressure failure, the CPWG upper bound leak model would not preclude structural failure unless containment survived well beyond its mean failure pressure. Hence, the CPWG leakage model leads to the conclusion that the development of a pressure-relieving leak is at best "remotely possible."

A recent Sandia test on a 1/8-linear-scale steel containment model produced results which would at first glance appear to corroborate the CPWG model; that is to say, significant leakage did not occur prior to structural failure. However, it is very risky to extrapolate these results to the present containment, or for that matter to any of the reference plants. First, the Surry containment is steel-lined, reinforced concrete, not a steel structure. (Even the reference plants that have steel containments also have a concrete shield, whereas the experiment did not.) Second, the deformation patterns in the experiment indicate that leakage around a penetration seal may have been imminent.

Finally, IDCOR (Ref. 7) presented a case that for Zion, pressure-induced leakage would be a more likely failure mode than gross structural failure.

In terms of source terms and consequences, it is more optimistic for containment to fail via a pressure-relieving leak than a gross rupture. Hence, we adopted IDCOR's position that "leak-before-break" is the "likely" outcome for the optimistic walkthrough.

For the "central" and "pessimistic" walkthroughs, we utilized the conclusion, based on the CPWG model, that leak-before-break was at best "remotely possible."

Question 48: Likelihood of Occurrence and Magnitude of a Late Hydrogen Burn

We define a late burn as one occurring sufficiently long after vessel breach to allow virtually all of the fission products from the melt release to have settled out in containment or to have been washed out by the sprays, but not necessarily long enough for the vaporization release to have been entirely depleted.

Because ignition sources are available during S_2D and S_3D, we considered a late burn to be inevitable if there is adequate hydrogen, adequate oxygen, and if there is

containment heat removal, (i.e., the containment sprays have not failed). In the absence of containment heat removal, the atmosphere would initially be steam-inerted. Condensation of steam on the walls may eventually "de-inert" the atmosphere. However, there are so many uncertainties that we have been unable to determine whether or not the atmosphere would become flammable before an overpressure failure occurred. We considered a late hydrogen burn in the absence of containment heat removal to be "indeterminate" for all walkthroughs.

The maximum quantity of hydrogen or other flammable gases could exceed 100% of that due to zirconium oxidation, if hydrogen due to steel oxidation and combustible gases from core-concrete interactions are included. However, the oxygen available could only burn about 150% zirconium equivalent. The amount of combustibles available depends on whether core-concrete interaction releases flammables and whether a prior burn has occurred.

IDCOR (Ref. 7) calculated a few hundred pounds of hydrogen generated for Zion. At the other end of the spectrum, SNLA (Ref. 33) calculated up to 2500 lbs due to Zircaloy oxidation, 1000 lbs due to steel oxidation and several hundred pounds from other sources, for a representative large, dry PWR.

We estimated that a late hydrogen burn could involve as much as 75% Zircaloy equivalent (optimistic) or 150% Zircaloy equivalent (central and pessimistic) if core-concrete interactions have occurred or if there has been direct heating. Otherwise, we limited the amount to 110% (central and pessimistic). From these total quantities, we subtracted the amount of hydrogen consumed in prior burns. The associated pressure rises were taken from a CLWG submittal (Ref. 3) and are given in Table 2.4.

Question 49: Likelihood of Containment Failure from a Late Hydrogen Burn

Containment pressures prior to a late burn were obtained from BMI (Ref. 5), SASA (Ref. 8), IDCOR (Ref. 7), and MARCH calculations previously performed at Sandia. To these were added the pressure increments obtained from Question 48. The values are given in Table 2.4. The likelihood of containment failure was evaluated as in Question 22 and 30.

Question 50: Likelihood of Various Levels of Auxiliary Building Breakthrough or Bypass, Given Containment Failure from a Late Hydrogen Burn

The logic for this question is similar to Questions 23 and 32.

Question 52: Likelihood of Late Containment Spray Failure, Given No Earlier Failure

The logic for this question is similar to Question 34. If the sprays had not failed earlier, we believed they could fail later due to continued pumping of debris-laden water or clogging of intake strainers. We took the likelihood of this event to vary from "remotely possible" to "indeterminate," taking into account the TAP A-43 experiments (Question 25).

Question 54: Likelihood of a Vaporization Release

The vaporization release may be precluded by either of two occurrences. First, if there are no significant core-concrete interactions (Question 42), there will not be a vaporization release from the fuel. Second, if core-concrete interactions occur but the reactor cavity is filled with water, the vaporization release from fuel will be scrubbed and mitigated. Otherwise a vaporization release is assured.

Question 55: Likelihood of Late Containment Leakage due to High Temperature

It is the current understanding of the CPWG (Ref. 4) that temperature-induced leakage would not be expected at Surry. We corroborated this understanding on the basis of BMI calculations (Ref. 5), which predict a maximum atmospheric temperature of 280 F for a sequence without containment heat removal (TMLB'). Elastomeric penetration seals do not begin to degrade below temperatures of about 400 F.

Question 56: Likelihood of Basemat Meltthrough Before Containment Overpressurization

If the core-concrete interaction occurs (Question 42), there is some uncertainty as to whether the core debris will penetrate completely through the basemat. The models in core-concrete interaction codes such as CORCON are not considered to be as accurate when the core debris freezes and starts to attack the concrete as a heat-producing solid. We thus have to rely on the limited experimental evidence that exists.

Experiments at Sandia appear to indicate that considerable erosion of concrete continues to occur after the melt solidifies. If water is supplied to a core debris layer which is already attacking concrete, the penetration continues but the debris layer cools down more quickly.

The RSS (Ref. 17) assumed that basemat meltthrough was inevitable. A race was envisioned between meltthrough (which was assumed to lead to depressurization) and

overpressure failure. The ZIP study (Ref. 32) questioned whether meltthrough was either assured or likely. IDCOR (Ref. 7) stated that if half or more of the core is dispersed, core-concrete interactions would be prevented.

If core-concrete interactions do occur, but there is an abundance of water overlying the melt, we expected meltthrough to be delayed, and meltthrough before overpressure was assessed to be "unlikely". If core-concrete interaction occurs without water, we took meltthrough before overpressure to be "indeterminate." If there is no core-concrete interaction, meltthrough is impossible.

Question 57: Likelihood of Containment Depressurization Given Basemat Meltthrough

If the core debris melts through the basemat, late overpressure failure is not necessarily precluded, because pressure relief through the ground might be too slow to prevent overpressurization. Calculations in BMI-2104 presume that basemat meltthrough leads to a depressurization of containment as a result of venting of the gases through the ground; however, the authors of that document recognized this to be an area of high uncertainty.

The RSS also assumed that meltthrough would depressurize the containment. However, calculations reported in the RSS indicate very slow flow out of containment, so that prompt depressurization is not assured.

The RSS also asserted that sump water could not escape containment. The presence of overlying water could thus delay depressurization. However, other studies (e.g., Ref. 35) concluded that sump water escape might be possible.

If there is no overlying water, we have taken rapid depressurization (given prior meltthrough) to be "likely" (optimistic), "indeterminate" (central), and "unlikely" (pessimistic). With overlying water we took depressurization to be "likely" (optimistic), "unlikely" (central), or "remotely possible" (pessimistic).

Question 58: Likelihood of Various Modes of Very Late Containment Failure, Given No Prior Containment Failure

If there is containment heat removal (i.e., late spray operation) or a large containment leak, overpressure failure cannot occur. If in addition there is significant core-concrete interaction (Question 42), the occurrence of basemat meltthrough is considered to be highly uncertain. Based on the information sources described in Question 56, we took the probability of meltthrough to range from "unlikely" (optimistic) to "likely" (pessimistic). If

there is no core-concrete interaction but there is containment heat removal, the only possible result is "no failure".

For cases where late containment heat removal is not available, there is a spectrum of possible answers given in Table 2.3. If there is no basemat meltthrough, it is possible (but unlikely) that containment would not fail because steam generation could be balanced by condensation on the walls. If there is basemat meltthrough but containment does not depressurize, subsequent overpressurization is considered to be "likely" (optimistic) or "certain" (central and pessimistic). If containment does depressurize, basemat meltthrough is the only failure mode because depressurization precludes overpressure failure.

For overpressurization failures, the split between pressure-relieving leakage and gross structural failure is the same as for Question 44.

Question 59: Likelihood of Various Levels of Auxiliary Building Breakthrough or Bypass, Given Very Late Containment Failure

The logic for this question is similar to Questions 23 and 32.

2.3 Treatment of Verbal Descriptors

Interpretation of words such as "likely", "indeterminate", "unlikely", or "remotely possible" is subjective. In cases where we have used these words, we did so because there was no clearcut way to quantify the likelihoods of the questions being asked. Still, some assignment of numerical values is necessary if the frequencies of the outcomes are to be estimated.

Table 2.7 shows 4 plausible assignments of values for the verbal descriptors we have used. In most cases, we used Alternative 1 to quantify the outcome frequencies; however, we also investigated the sensitivity of some of the results to the choice of quantification alternatives. The results of the sensitivity study are described in Section 3.6.

2.4 Treatment of Other Sequences and Other Plants

The questions asked on the containment event tree and the utilization of information to quantify them vary from sequence to sequence and from plant to plant. Below we shall provide a brief description of some of the important differences. For the present document, we will limit our

discussion to the main points only and will not attempt to give any of the details regarding the construction or quantification of the event trees. We will provide details for other sequences and plants, commensurate with Section 2.2, in documentation to be provided later in 1985.

2.4.1 Surry

Sequences evaluated for Surry in addition to S_2D and S_3D are TMLB' (station blackout) and AB (large LOCA with station blackout). We did not perform a containment event tree analysis for sequence V (interfacing systems LOCA) because it is a sequence defined by a unique containment failure mode (i.e., the bypass of containment).

In TMLB' and AB, containment sprays are not available early, but may be available late if ac power is recovered. We allow for restoration of power (and hence sprays) in accordance with data developed for EPRI (Ref. 36).

Temperature-induced failure of the reactor coolant system is somewhat more likely for TMLB' than for S_3D, for two reasons. First, loss of ac power results in loss of reactor coolant pump (RCP) cooling which can lead to a failure of the RCP seals. Second, loss of auxiliary feedwater results in dryout of the steam generator secondary side which can increase the likelihood of a steam generator tube rupture. We have no analyses to support using different values for TMLB' than for S_3D. Thus, despite these differences, we treated the likelihood of temperature-induced failure of the reactor coolant system to be the same for both sequences.

2.4.2 Zion

Sequences evaluated for Zion are S_2D, S_3D, and TMLB'. There are several differences between the Zion and Surry analyses. The Zion containment has fan coolers; hence, one must ask about survivability of the fan coolers as well as survivability of the sprays. The actuation setpoints are different: 4.5 psig for the fans and 23 psig for the sprays, compared to 10.3 psig for the sprays in Surry. Hence, the likelihood of the containment sprays not actuating before vessel breach is higher for Zion than for Surry. The containment is atmospheric rather than subatmospheric; hence the likelihood of undetected pre-existing leakage is somewhat higher. The containment failure pressure is higher than at Surry. The model for pressure-induced containment leakage (CPWG, Ref. 4) allows for more leakage than in Surry, but not enough to make induced leakage a significantly more important containment failure mode.

2.4.3. Peach Bottom

Five sequences were analyzed for Peach Bottom: AE (large LOCA with failure of core cooling), TC (transient with failure to scram), TW (transient with failure of containment heat removal), TQUV (transient with failure of core cooling), and TB (station blackout with long-term battery depletion).

Many of the questions we posed for the BWRs were quite different from those we posed for the PWRs. This is to be expected, since the BWR containment and reactor coolant system designs are very different from the PWR designs. Below is a list of some of the questions that are specific to Peach Bottom, together with some of the observations we used for quantifying the likelihoods:

(1) Will there be unmitigated leakage through the main steam isolation valves? Some leakage through MSIVs is often observed during integrated leak rate tests (SASA, Ref. 11). During accidents involving steam line isolation, such leakage could be controlled if the turbine gland valves remained sealed and the condenser vacuum was maintained. Leakage control could not be maintained with loss of offsite power.

(2) Is there a significant likelihood of preexisting leakage? Data from Weinstein (NRR, Ref. 28) indicate that BWRs have been operated with unintentionally open containments more frequently than PWRs. We have been told that Weinstein's data does not apply to currently operating BWRs; furthermore, the fact that Mark I containments are pre-inerted provides some assurance that inadvertant openings would be detected after some time. Without a more current data base, we had to rely on the Weinstein data. We assigned a much lower likelihood to pre-existing drywell leaks, compared to wetwell leaks, because the Mark I drywells are operated at a slightly positive pressure.

(3) Can containment be bypassed during an ATWS? SASA (Ref. 11) has identified a scenario in which the current operating procedures for anticipated transients without scram could, under pessimistic assumptions, lead to a reactivity excursion causing a LOCA outside containment. We accounted for this possibility for the TC accident, although we considered it unlikely.

(4) For TW and TC sequences, will the containment fail before the core melts? The current emergency procedure guidelines (EPGs) call for the operator to vent containment under conditions that would occur during a TW or TC accident. Provisions are also made for using high pressure service water to make up lost suppression pool inventory and for actuating the drywell sprays as a containment

cooling alternative. The estimated frequency and failure modes for the TW and TC sequences strongly depend upon the assumptions made about the operator's capability to implement the procedures and about the effects of the procedures on the sequence. There are at least 5 possibilities. We considered the ones itemized below, giving them different weights for each of the two sequences and the three walkthoughs:

(a) The operator may succeed in venting containment and providing core water makeup. The accident is terminated successfully.

(b) The operator may succeed in venting containment, but the pumps drawing from the suppression pool may fail due to cavitation (as they are not qualified for saturated water) and the operator may be unable to maintain water delivery indefinitely from an external source (such as control rod drive, condensate, or high pressure service water). The core melts down in either an open or a closed containment, depending upon whether the operator recloses the vent.

(c) Same as (b), except the pumps fail from overheating (as the pump rooms are near the suppression pool and the doors are normally kept closed).

(d) The operator may fail to initiate venting, the containment fails due to overpressurization, and core water makeup may fail as a result of pump cavitation and/or dislocation of the flow lines. The core melts in a failed containment. (This is the RSS scenario.)

(e) The operator may fail to initiate venting, and the pumps fail prior to containment overpressurization. High drywell pressure may defeat the operation of the automatic depressurization system valves (SASA, Ref. 10 & 11), so that the operator cannot provide makeup water from low pressure external sources. The core melts in a closed containment.

(5) <u>Are there any other sequences for which containment failure could precede core melting?</u> It is possible for containment to fail during an AE sequence after the core has become severely degraded but not yet completely molten. This could occur if the amount of hydrogen produced in-vessel exceeds about 70% of the Zircaloy equivalent (BMI, Ref. 7).

(6) <u>Will there be a temperature-induced failure in the reactor coolant system before vessel breach?</u> Temperature-induced failures in BWR reactor coolant systems are much less likely than for PWRs because the recirculation flow paths are restricted by the upper internals (dryers and separators) and there is a much larger overall structural mass that serves as a heat sink.

(7) <u>Does core slumping occur?</u> Differences in the BWR vessel structure design lead to different estimates of melt progression and subsequent containment loading phenomena than in the PWRs. The BWR fuel is supported in the core differently from the PWR fuel, which is supported en masse by the bottom support plate. The canister support design and structures beneath the BWR core make a coherent slump of large portions of the core less likely than for a PWR (SASA, Ref. 10 & 11).

(8) <u>Do energetic in-vessel steam explosions occur?</u> Because core slumping is less likely for BWRs than for PWRs, the occurrence of a large in-vessel steam explosion is also less likely. Even if the core slumps and a large steam explosion occurs, the BWR vessel itself has features which tend to mitigate the effects. For example, while the three PWR vessels examined in this study are supported at the nozzles, the BWR vessels are supported by a lower head skirt which in turn is supported by a pedestal. This makes a lower head failure less likely. Above the core the BWR vessels have steam dryers and separators, structures which would tend to absorb the impact of an upward directed steam explosion. Because we had no calculations for the likelihood or effects of in-vessel steam explosions in BWRs, we tried to account for the mitigative factors mentioned above by adjusting some of the PWR estimates. This approach leads to numbers which are not rigorously defensible but at least are qualitatively reasonable.

(9) <u>Will the reactor coolant system be pressurized at vessel breach?</u> High reactor coolant system pressures during core melting affect the likelihood of early containment failure through the potential for a rapid vessel depressurization and direct heating after vessel breach. The reactor pressure will be high if the automatic depressurization system (ADS) fails to actuate, or if it actuates but subsequently fails due to adverse conditions. Failure to actuate is generally caused by operator error. Adverse conditions that can lead to delayed ADS failure include (1) depletion of dc batteries following a loss of all ac power or (2) defeat of the air-operated solenoids for the safety/relief valves caused by high drywell pressure (SASA Ref. 11). Usually, the reactor pressure is high for transient and small LOCA sequences involving core melting before the containment fails or is vented, and is low otherwise.

(10) What is the likelihood and effects of direct heating or an ex-vessel steam explosion following vessel breach? Direct heating and ex-vessel steam explosions are judged to be less likely for BWRs than for PWRs because of the less coherent nature of the meltdown. That is, the possibility of a large mass of core debris exiting the reactor vessel at the time of breach is considered less likely for a BWR. To account for the possibility of these phenomena, however, we had to estimate the magnitude of the interactions and ask whether the suppression pool vents clear sufficiently fast to relieve the pressure increment in the drywell. To determine the magnitude of the steam explosion, we based our analysis on the amount of water that could exist within the reactor vessel pedestal. For direct heating, we adopted the fractions of thermal and chemical energy transfer developed by the CLWG for PWRs. The question of vent clearing has not been analyzed for these phenomena, and we had to make subjective judgments for the three walkthroughs.

(11) Will the containment breach be in the drywell? The RSS originally predicted that containment failure would occur just above the midplane of the toroidal suppression chamber (i.e., in the wetwell). A more recent analysis (AMES, Ref. 25) predicted that the failure point would be in the drywell. In BMI-2104, a drywell failure was assumed, but the authors discussed the possibility that the location of failure could be different. For TC, dynamic loads in the suppression pool could increase the likelihood of a wetwell breach. However, the suppression pool structure has recently been strengthened by the torus integrity improvement program, and so we estimated that for all sequences, the point of failure would likely be in the drywell.

(12) Will containment failure lead to failure of the suppression pool function? If the failure occurs in the drywell, suppression pool bypass is guaranteed. If it occurs in the wetwell, there is considerable uncertainty about whether the suppression pool would survive. Because the containment is a free-standing steel shell structure with a high failure pressure, the forces associated with the failure could be violent (FVCS, Ref. 27). On the other hand, the suppression chamber has been considerably strengthened by the torus integrity improvement program.

(13) Will leakage paths develop in the drywell after vessel breach? Three causes of containment leakage induced during severe accidents are considered for Peach Bottom. The first is high pressures which could cause the drywell head seal to separate from the structure (CPWG). The second is high temperatures which could lead to degradation of the elastomeric electrical penetration seals (CPWG, SASA). The third is the direct attack of core debris on

the drywell structure, causing failure of the shell at a location where it is not directly backed by concrete (CLWG). Pressure-induced leakage is found to be more likely than temperature-induced leakage when the CPWG models are applied to accident situations. (This observation is corroborated by recent SASA analyses which indicate that the outboard electrical penetration seals would not overheat.) Direct core debris attack on the drywell structure is considered unlikely because of the noncoherence of the meltdown, the interfering effect of the reactor vessel pedestal, and the existence of two large floor sumps that would tend to collect the core debris.

(14) Is the secondary containment effective in removing fission products from the atmosphere? Most releases following drywell or wetwell failure go through the secondary containment. (An exception is the "direct" release considered in the RSS, where the failure occurs in the wetwell near a truck ramp.) In virtually every scenario, the secondary containment panels blow out, providing a potential direct pathway to the environment. A hydrogen burn in the secondary containment could cause further damage and possible bypass. If large bypass does not occur, the efficacy of the secondary containment depends upon whether the fire sprays are actuated and whether the standby gas treatment system (SGTS) continues to operate. The fire sprays are designed to actuate when temperature-sensitive plugs reach about 145 F. Dampers to the SGTS close at 170 F, making it unavailable for further use. In most cases involving drywell failure, we estimated that the fire sprays would actuate and the SGTS dampers would close.

2.4.4 Grand Gulf

The sequences we analyzed for Grand Gulf were TC (transient with failure to scram), TQUV (transient with failure of core cooling), TB (station blackout with long-term battery depletion), S_2E (small LOCA with failure of core cooling), and TPI (transient with stuck-open relief valve and failure of containment heat removal).

The questions we posed for Grand Gulf were similar to Peach Bottom in some respects and quite different in others. Many of the important differences stem from two features: (1) the Mark III drywell is contained within the wetwell, rather than being separate from it, and (2) neither the drywell nor the wetwell is initially pre-inerted. In place of pre-inerting, Grand Gulf has a hydrogen igniter system which is dependent on ac power and which is manually actuated by the operator.

For Grand Gulf, a number of questions were posed to address issues associated with hydrogen burning. Burning above the suppression pool could cause the following significant events to occur: (1) The pressure increase in the wetwell could cause water to flow over the weir wall onto the drywell floor and into the pedestal area, thus affecting subsequent steam spikes, debris bed coolability, and vaporization releases. (2) The high temperatures produced by diffusion flames could induce leakage through drywell penetrations (CLWG). (3) The flow of hot, combusting gases through the vacuum breakers could degrade their ability to reclose. (4) Igniter unavailability or the occurrence of a steam spike that rapidly forces hydrogen into the wetwell could lead to a global deflagration or local detonation that could threaten the containment structure.

An important consideration in the Grand Gulf analysis is the dependency of many mitigative systems on ac power provided by offsite sources and the two emergency diesel generators (but not the high pressure core spray dedicated diesel). In addition to the igniters, the containment sprays, standby gas treatment system, vacuum breakers, and suppression pool makeup system all are ac dependent, the latter two through normally closed, ac-powered valves. Thus, even in sequences not involving total station blackout, many systems are unavailable if offsite power is lost, the two diesels fail, and power is not restored. (We should note that if the suppression pool makeup water has not dumped, we allow it to act like a missile shield for in-vessel steam explosions that fail the upper head.)

As mentioned, the Mark III drywell is contained within the wetwell; thus, drywell leakage results only in suppression pool bypass, not release from containment. We allow for six causes of suppression pool bypass in excess of technical specifications: (1) preexisting small cracks and leaks, (2) induced leakage around the personnel air locks or other penetrations caused by high drywell temperatures and pressures (CPWG model), (3) induced leakage through vacuum breakers caused by hydrogen combustion events in the wetwell, (4) leakage through failed piping or drywell failure caused by local or global detonations in the wetwell, (5) gross failure of the drywell caused by an ex-vessel steam explosion or direct heating that does not clear the vents, and (6) gross failure of the drywell caused by core debris attack on the reactor vessel pedestal and subsequent vessel movement. Induced leakage through instrument penetrations due to high drywell temperatures is considered less credible for Grand Gulf than for Peach Bottom (CPWG), because the penetration seals are steel welded.

Another important difference is the containment structure itself. The Grand Gulf containment is a fairly low-pressure concrete structure whose expected point of failure is at the upper springline. The most likely failure mode is equivalent to a vent release; hence, we do not consider the intentional venting of containment according to the EPGs to be a separate containment release mode. The suppression pool most likely survives, partly because of the more benign containment failure which would occur well above the pool, and also because of the increased thickness of concrete in the region of the pool.

The Grand Gulf high pressure core spray (HPCS) system has its own cooling subsystem, unlike the high pressure coolant injection (HPCI) system at Peach Bottom, and its pump is more capable than a HPCI pump of taking suction from a saturated pool. Thus, for TPI and TC at Grand Gulf, core cooling does not fail prior to containment failure. We assume, consistent with IDCOR and previous PRAs, that containment failure can lead to failure of the high pressure core cooling systems. We acknowledge, however, that it is quite possible that core cooling would not fail and that the sequences would not necessarily develop into meltdowns.

Finally, the standby gas treatment system at Grand Gulf is a low volume system (2300 cfm), compared to that at Peach Bottom (20,000 cfm), and we therefore do not credit it with having an appreciable effect on any of the sequences.

2.4.5 Sequoyah

Five sequences were analyzed for Sequoyah: S_2H (small LOCA with loss of core cooling recirculation), S_2HF (small LOCA with loss of core cooling and containment spray recirculation), S_3D (small pump-seal LOCA with loss of core cooling injection), TML (transient with loss of core heat removal), and TMLB' (station blackout).

Because it has an ice condenser containment with hydrogen igniters, we added several questions to the Sequoyah event tree that were not in the trees for either Surry or Zion. For example, we asked about the availability of the igniters, which are ac dependent and manually actuated. We differentiated between whether a burn occurred in the upper compartment or the lower compartment (including the ice region); burns that involved the upper compartment generally provided higher pressure spikes. We accounted for the energy consumed by the ice to estimate the time of ice depletion due to melting; further, we allowed for the possibility that preferential melting near the source of steam and hydrogen could lead to early bypass paths. We considered the possibility that hydrogen

burns could defeat the air return fans; burns in the upper compartment were deemed to provide the greater threat in this regard because the fans take suction from the upper compartment.

Two sequence specific points about Sequoyah warrant discussion. First, in the S_2HF sequence, we used the recent IDCOR and ASEP assessments that the failure of the operators to remove the drain plugs is negligible. Second, except for TMLB', the accident sequences analyzed for Sequoyah had negligible probability of ending in containment failure due to direct heating. This resulted from consideration of the reactor cavity design at Sequoyah. The cavity is basically a large room with the keyway located at some distance from the reactor vessel. In the S_2H, S_2HF, S_3D, and TML sequences, by the time of vessel breach, this room would be flooded from the combination of the LOCA, coolant injection (if successful), spray injection, and melting of the ice. The lower portion of containment would also be partially flooded, covering the reactor cavity exit. Thus, to have direct heating, the ejected melt would have to sweep all this water away, an event we deemed not credible. For TMLB' however, the reactor cavity would not be flooded, and direct heating remains a possibility. For this case, direct heating was handled similarly to Surry and Zion.

3. RESULTS

The results of the containment event analysis are presented in this section. A total of twenty-two sequences were analyzed for the five plants: Surry, Zion, Peach Bottom, Grand Gulf, and Sequoyah. The results for each sequence are described briefly and tabulated. Each sequence table has three parts. The first is the sequence frequency as estimated by ASEP; a baseline value is given as well as the potential frequency of the sequence in light of recent plant modifications. The second part of the table gives the containment failure mode probabilities for the optimistic, central, and pessimistic walkthroughs. Two cases are presented for the central and pessimistic analyses. Case 1 excludes in-vessel steam explosion and ex-vessel direct heating phenomena, and case 2 includes them. (In the optimistic walkthrough, these either do not occur or have negligible affect so the two cases are not differentiated.) The third part of each table describes the principal containment pathways and whether a calculation was made for them in BMI-2104. For Grand Gulf, we include a fourth part which depicts the probabilities of various degrees of drywell-to-wetwell leakage causing bypass of the suppression pool.

3.1 Results for Surry

3.1.1 <u>Sequence S_2D</u>

The results for sequence S_2D at Surry are shown in Table 3.1 for the three different walkthroughs. The results range from no containment failure in the optimistic case to nearly assured failure at the time of vessel breach in the pessimistic case.

In the optimistic case, virtually all of the sequence frequency is associated with no containment failure. In this walkthrough, the formation of coolable debris beds is considered very likely and the failure of sprays during the accident progression considered very unlikely. Thus a safe stable state is achieved wherein the decay heat is removed by the containment sprays, and the containment does not fail.

In the central estimate analysis, spray failure after vessel breach due to debris in the containment recirculation sump is considered unlikely, but formation of a coolable debris bed is also considered unlikely. This scenario results in the shift from no containment failure to late containment failure due to either basemat meltthrough or gross overpressurization from production of noncondensibles. Centrally, we estimate that more core debris is involved in the ejection process at vessel breach, thus giving rise to a very small chance (~1%) of

containment failure due to direct heating of the containment atmosphere at the time of vessel breach in Case 2.

In the pessimistic walkthrough, containment failure at vessel breach, either due to direct heating or a combined steam spike and hydrogen burn, dominates the results. The estimate of 100% zirconium oxidation by the time of vessel breach in this walkthrough produces the high probability of failure due to a steam spike and hydrogen burn in Case 1. These phenomena, in combination with larger fractions of the core participating in direct heating, give rise to nearly assured containment failure in Case 2.

3.1.2 Sequence S_3D

The results for sequence S_3D at Surry are shown in Table 3.2. The results are similar to those for sequence S_2D, discussed in the previous section, but there are some differences. The S_3D sequence remains at higher RCS pressures than does S_2D. The higher RCS pressure provides more severe direct heating loads but increases the chances of attaining a coolable debris bed because the higher driving forces involved during the ejection process disperse the debris more finely. The lack of RCS depressurization also retains more hydrogen in the RCS, resulting in larger amounts of hydrogen being released from the RCS at the time of vessel breach. Thus, the higher RCS pressure results in a higher probability of early containment failure for the central case, but also in a higher probability of no containment failure and less chance of basemat meltthrough. In the pessimistic analysis there is more chance of failure at vessel breach due to a combined steam spike and hydrogen burn either with or without direct heating. In addition, because the RCS remains pressurized, an induced small steam generator tube leak is possible.

3.1.3 Sequence TMLB'

The results for sequence TMLB' at Surry are shown in Table 3.3. They are similar in character to the results discussed earlier for sequence S_2D, but there are some differences. The differences arise due to ac power considerations. In sequence TMLB', ac power is initially unavailable. However our analyses considered that it was likely that ac power would be recovered late in the accident. This has several effects. First, since power is unavailable early, containment heat removal is also unavailable early, so the containment becomes inert early in the accident. This prevents any hydrogen combustion at the time of vessel breach, thus eliminating the possibility of failing containment at this time due to a combined steam spike and hydrogen burn. A second effect is that when ac

power and containment heat removal are restored late in the accident, the containment is eventually deinerted, so the hydrogen which has accumulated in the interim can then combust. This provides the opportunity for late containment failure due to a hydrogen burn. The third effect is that if power is not restored, containment failure cannot be prevented. The containment will ultimately fail either from basemat meltthrough or overpressurization. In addition, pessimistically there is a small probability (~10%) that a small steam generator tube leak will be induced.

3.1.4 Sequence AB

The results for the three walkthroughs (optimistic, central, and pessimistic) for sequence AB at Surry are summarized in Table 3.4. The results are dominated by either no containment failure or late containment failure. With the exception of failure of containment due to a missile from an in-vessel steam explosion, the results are identical for Case 1 and Case 2. This is to be expected since in sequence AB (a large LOCA) the RCS is at nearly ambient pressure so there is no driving force for the direct heating phenomenon.

In the optimistic analysis the majority of the event tree outcomes result in no containment failure. This ensues from considerations that a coolable debris bed will likely be formed and that ac power, and hence containment heat removal, will be restored. However, there is still some chance that the debris will not be coolable. This possibility coupled with the chance that power will not be restored leads to moderate chances of late containment failure due to basemat meltthrough and late pressure induced leakage.

In the central walkthrough, pressure induced containment leakage sufficient to preclude overpressure is considered less likely and larger amounts of zirconium are oxidized. This leads to a shift away from the no containment failure and pressure induced leakage categories of the optimistic walkthrough toward the containment failure categories of late hydrogen burns and gross overpressure.

In the pessimistic analysis the results show practically no chance of avoiding containment failure, but the failures are still dominated by late containment failure modes. The most notable change from the results of the central case is that containment failure due to gross overpressurization dominates. This arises because both coolable debris bed formation and pressure induced leakage sufficient to preclude overpressurization are considered as very unlikely.

3.2 Results for Zion

3.2.1 Sequence S_2D

The results of the optimistic, central, and pessimistic walkthroughs for sequence S_2D at Zion are shown in Table 3.5. The most probable result for all three walkthroughs is no containment failure or late containment failure modes, with the exception in the pessimistic case when direct heating was considered.

In the optimistic analysis, nearly all of the outcomes of the event tree result in no containment failure. Since the debris is sufficiently dispersed at vessel breach to be coolable and since the containment sprays are optimistically assessed to survive, a safe stable state is reached wherein the decay heat is removed by the containment sprays and containment is never breached. There is a very small chance (~2%) of small preexisting leakage or isolation failure. In addition, there is some small chance that the debris will not be coolable after ejection from the RCS, concrete attack will begin, and ultimately basemat meltthrough will occur. However, in the optimistic case this only represents ~1% of the sequence frequency.

The central results show a partial shift to either basemat meltthrough or late overpressure, but the majority of the sequence end states still result in no containment failure. A higher likelihood of not obtaining a coolable debris was assessed in the central case giving a moderate chance (~33%) of containment breach by basemat meltthrough. In addition, the probability of containment spray failure at the time of vessel breach was increased over that used in the optimistic estimate. This leads to a moderate chance (~10%) of late overpressurization failure of containment. A small chance (0.1% - 0.3%) of early containment failure was predicted, and also a small chance (~3%) of a small preexisting leak or isolation failure was estimated.

In the pessimistic walkthrough, containment failure was nearly assured. In the this case, 100% zirconium oxidation before vessel breach was used in the analysis. In the case without direct heating, this resulted in a moderate chance (~16%) of containment failure at the time of vessel breach due to a coincident steam spike and hydrogen burn. The majority of the outcomes for Case 1 result in late overpressurization of containment because the containment sprays fail at the time of vessel breach. When direct heating and steam explosion effects are included in Case 2, the majority of the outcomes result in containment failure due to direct heating at the time of vessel breach. The other major mode of containment failure in Case 2 is late overpressurization due to failure of the containment sprays at the time of vessel breach (just as in the Case 1 results).

3.2.2 Sequence S_3D

The results for sequence S_3D at Zion, summarized in Table 3.6, are similar to the results described previously for S_2D at Zion with a few exceptions. The optimistic results are identical for the two sequences. In the central analysis, we found a higher (but still small) probability of early containment failure from direct heating, but also a higher probability of no containment failure and a lower probability of basemat meltthrough for S_3D than for S_2D. In the pessimistic results for Case 2, our results showed a slightly higher probability of containment failure at vessel breach due to direct heating in S_3D than in S_2D. Both of these effects are due to higher RCS pressures in S_3D than in S_2D (see similar discussion for Surry in Section 3.1.2). Finally, because the RCS remains at elevated pressures in S_3D, there is a possibility of inducing a small steam generator tube leak.

3.2.3 Sequence TMLB'

Optimistic, central and pessimistic results are provided for sequence TMLB' at Zion in Table 3.7. The results for this sequence are similar to those described for the S_2D sequence with some differences.

In the case of TMLB', the plant initially undergoes a complete loss of ac power. This causes the containment heat removal systems to be unavailable and renders the containment atmosphere inert early in the accident. Thus combustion is prevented at the time of vessel breach which reduces (below that predicted for S_2D) the probability of containment failure due to the combination of hydrogen combustion, steam spike, and direct heating at vessel breach. Our analysis included the likelihood of restoring ac power late (~4 hrs) in the accident. This provided the opportunity for restoring containment heat removal and thus preventing late containment overpressurization. However, restoring containment heat removal condenses steam from the atmosphere, deinerting it. This allows hydrogen burns to occur at a very high initial concentration of hydrogen which threatens containment integrity and increases the probability of containment failure due to a late hydrogen burn. For the remainder of the TMLB' sequence in which power is not restored, containment failure occurs due to either overpressure or basemat meltthrough.

As for the S_3D sequence, there is a possibility that a small steam generator tube leak is induced in TMLB'.

3.3 Results for Peach Bottom

3.3.1 Sequence AE

The optimistic, central and pessimistic results for sequence AE at Peach Bottom are summarized in Table 3.8. Late overpressurization failures of the containment after vessel breach dominate the probabilities in the optimistic and central cases. The failure is always a leak for the optimistic case, but is a large rupture half of the time for the central case.

Both the central and pessimistic walkthroughs predict that accumulation of hydrogen due to metal oxidation in-vessel can cause overpressurization failure of the containment before vessel breach. For the central case, there is a moderate chance of this happening (~10%), with the failures evenly split between leaks and ruptures. In the pessimistic case, overpressurization due to hydrogen always leads to rupture and is the dominant mode of containment failure. The pessimistic case also predicts a small chance (~6%) that direct heating and steam spike will fail the containment at the time of vessel breach.

3.3.2 Sequence TB

The optimistic, central and pessimistic results for sequence TB at Peach Bottom are summarized in Table 3.9. The most probable outcome for the optimistic and central cases is late failure of the containment. The failure is always a leak in the optimistic case; in the central case gross containment rupture is the predicted mode of late failure half of the time. There is a moderate chance (~10%) of drywell overpressurization at the time of vessel breach for the central case, while it is the dominant mode of failure in the pessimistic case. Containment always fails by gross rupture in the pessimistic case.

The primary system is pressurized because high drywell pressure or battery depletion fails the ADS and high pressure injection. The RCS pressure provides a dispersal mechanism so that consideration of direct heating in the central and optimistic cases results in more failures due to overpressurization near the time of vessel breach.

Restoration of power and, hence, mitigative systems, is the main reason that containment failure is prevented for a moderate fraction (~27%) of the optimistic cases.

3.3.2 Sequence TC

The optimistic, central and pessimistic results for sequence TC at Peach Bottom are summarized in Table 3.10.

The estimated frequency of the TC sequence depends upon assumptions regarding the likelihood and effects of venting the containment. In the optimistic and central cases venting will usually prevent core melt at least until coolant injection fails from loss of net positive suction head. If the containment is not vented, high lube oil or room temperatures or ADS failure due to high containment pressure can fail coolant injection. For these cases, the core melts before containment overpressurization. In the pessimistic case, venting usually will not be able to prevent overpressurization of the containment. Containment failure occurs before core melt and causes the failure of coolant injection.

We estimated that in half of the cases in which venting prevents early overpressurization (but not core degradation), the vents are reclosed. For these cases, the containment fails late. For the optimistic case, the late failure is a leak that prevents further pressurization of the containment; half of the late failures are large for the central analysis, while all are large for the pessimistic walkthrough.

The pessimistic analysis is dominated by overpressurization of the containment before core melt, but a LOCA is induced outside the containment a small fraction of the time (~10%). For both the pessimistic and central cases, the containment also can fail at the time of vessel breach or shortly thereafter due to the dynamics of the breach or the direct attack of the drywell wall by molten core debris. In- vessel steam explosion and direct heating phenomena have little effect on the containment analysis.

3.3.4 Sequence TQUV

The optimistic, central and pessimistic results for sequence TQUV are summarized in Table 3.11. Late overpressurization is the containment failure mode which occurs most frequently for all cases. In the optimistic case the failure is a leak that prevents further pressurization. Half of the late overpressurization failures are ruptures in the central case while all are ruptures in the pessimistic case. These include small probabilities of thermal-induced failures of the drywell in the central and pessimistic cases.

The vessel fails by high pressure meltthrough most of the time. In the pessimistic case, about half of the time a large amount of debris is collected in the vessel bottom head before vessel breach. This increases the probability that the containment will fail by melt structure attack. In the Case 2 results, which consider direct heating,

overpressurization at the time of vessel breach is predicted for about one-fourth of the core melts.

3.3.5 Sequence TW

Table 3.12 summarizes the optimistic, central, and pessimistic containment failure modes for sequence TW at Peach Bottom. As discussed in Section 2.4, the estimated frequency of the TW sequence depends upon assumptions regarding the likelihood and effects of venting the containment. In the optimistic analysis, venting will prevent core melt and thus, the frequency in the table is for unvented melts. In the central case, venting prevents early pressurization of the containment and reduces the probability of core melt, but does not always prevent core melt, so about half of the core melts are vented. While venting prevents early containment failure in the pessimistic case, it does not prevent core melt so the estimated frequency is mainly for vented melts. In half of the vented melts, the vents are reclosed. Reclosing the vents usually results in late containment failure, but in a fraction of the pessimistic cases the containment fails by melt structure attack near the time of vessel breach.

If the containment is not vented, high temperature lube oil or ADS failure due to high containment pressure can still fail coolant injection before containment failure in the optimistic and central cases. For the optimistic case, late failure is a leak that prevents further pressurization of the containment; half of the late failures are ruptures for the central case, while the failures are always ruptures in the pessimistic case.

3.4 Results for Grand Gulf

3.4.1 Sequence TC

Table 3.13 summarizes the optimistic, central, and pessimistic containment failure mode and suppression pool bypass fractions for TC core-melt accidents at Grand Gulf. The only significant containment failure mode in all three quantifications is failure before core melt due to over-pressure following suppression pool overheating. As a result of the containment failure, ECC injection fails due to either loss of net positive suction head or deformation of piping. This containment failure mode is the one analyzed in BMI-2104 for the Grand Gulf TC sequence. Since containment fails before core melt, the containment failure mode is not affected by direct heating or steam explosions. Also, the containment will be steam inert in the TC scenario; thus, hydrogen burns or detonations will not cause containment failure or suppression pool bypass.

In the TC sequence, we estimated that the containment failure would be sufficiently energetic to force water from the suppression pool over the weir wall, flooding the drywell floor. This flooding provides scrubbing of vaporization releases before the releases enter the drywell. Thus, for the TC scenario, any suppression pool bypass that is induced late in the accident will not significantly affect the releases from containment. However, early suppression pool bypass will still be important.

The fraction of cases involving early suppression pool bypass due to leakage through the drywell wall is affected by direct heating and ex-vessel steam explosions. Without direct heating or steam explosions, no leakage was induced early, so early leakage only existed in cases with pre-existing leakage. With direct heating and ex-vessel steam explosions considered, we found a very small probability (~1%) of early leakage through the drywell wall in the central case and a slightly higher probability (~10%) of leakage in the pessimistic case.

3.4.2 Sequence TPI

Table 3.14 summarizes the conditional containment failure mode and suppression pool bypass probabilities for TPI based on our optimistic, central, and pessimistic analyses.

The results for TPI are similar to TC, Section 3.4.1. However, the containment will pressurize more slowly following suppression pool overheating than in the TC sequence, giving a small chance that a pre-existing leak will prevent containment overpressurization before core melt. Since ECC must fail if the scenario is to proceed to a core melt, we assumed that ECC would fail due to high temperature for these leakage cases.

3.4.3 Sequence TQUV

In the TQUV sequence, hydrogen burning in the containment will not be prevented by steam-inerting as it had been in the TC and TPI sequences. When the igniters are operating, controlled burning of hydrogen will most likely prevent a hydrogen burn from occurring that would be large enough to fail the containment. However, the igniters will not be operating in most of the TQUV cases due to loss of offsite power and the two emergency diesel generators or their support systems (see Section 2.4). Without a reliable ignition source, it is most likely that fewer hydrogen burns will occur, but that the resulting burns, when they do occur, will be larger than those

occurring with the igniters operating. There is a small chance that the hydrogen will accumulate to detonable levels before ignition occurs. Since the timing of burns is unpredictable when the igniters are not operating, the containment failure probability is very sensitive to assumptions regarding hydrogen production and combustion.

Table 3.15 summarizes the optimistic, central, and pessimistic containment failure mode and suppression pool bypass fractions for TQUV core-melt accidents. These results reflect the uncertainties in hydrogen production rates and ignition threshholds. In the optimistic analysis, little hydrogen is generated during the melting of the core; in the central analysis, approximately one-quarter of the zirconium is oxidized; and in the pessimistic analysis, almost half of the zirconium in the vessel reacts to form hydrogen before vessel breach. Thus, optimistically, we found a moderate chance (~45%) of the accident being terminated without containment failure. The remaining outcomes consist of failures due to either late overpressurization or late hydrogen burns. No early containment failures were predicted in the optimistic case. However, in the pessimistic analysis, a large percentage (~80%) of the outcomes resulted in early containment failures. Most of these failures were due to hydrogen burns. The remaining portion in the pessimistic case resulted in late containment failure due to hydrogen burns. In the central walkthrough, containment failure was about evenly split between early and late modes.

Direct heating and steam explosions had minimal effect on the containment failure modes, even in the pessimistic case. However, direct heating and ex-vessel steam explosions did increase the probability of early suppression pool bypass for the central and pessimistic walkthroughs (from 0 to ~1% and from ~2 to ~11%, respectively).

Although direct heating and ex-vessel steam explosions were the largest contributors to early drywell structural failures, hydrogen burns or detonations were the largest contributors to both early and late penetration failures in the drywell wall. These failures were due to either vacuum breakers sticking open following a burn or to structural damage due to detonations. Since the vacuum breaker lines each contain a motor-operated valve that will not open without ac power, drywell leakage due to stuck-open vacuum breakers was only allowed in cases with ac power initially available or with ac power restored late.

3.4.4 Sequence TB

Table 3.16 summarizes the optimistic, central, and pessimistic containment failure mode and suppression pool

bypass probabilities for TB core-melt accidents. The results for TB are similar to the TQUV results. However, there are more containment failures due to hydrogen burns in the central and pessimistic cases because ac power is not initially available for any of the TB sequences. Thus, it is most likely that there will be fewer burns in TB than in TQUV, but the magnitude of the pressure rise will be larger. In contrast, TB had fewer failures due to late hydrogen burns than TQUV for the optimistic analysis. This resulted from the optimistic estimate that the RCIC failure due to battery depletion would not usually occur until after the containment was steam inert.

The probability of suppression pool bypass in TB is similar to TQUV. However, the failure of ac power in TB results in fewer penetration failures early due to stuck-open vacuum breakers, yet more penetration failures late due to ac power restoration and subsequent opening of the vacuum breaker block valves.

3.4.5 Sequence S_2E

Table 3.17 summarizes the optimistic, central, and pessimistic containment failure mode and suppression pool bypass fractions for S_2E core-melt accidents. These reults are similar to the TQUV results. However, since ac power is available for a larger fraction of the S_2E cases than the TQUV cases (approximately half), there is a smaller percentage of containment failures due to early hydrogen burns in S_2E than in TQUV. Conversely, the chance of containment failure at vessel breach is larger in S_2E than in TQUV for the central and pessimistic cases. These failures are predicted for scenarios in which a significant amount of hydrogen is trapped in the drywell before vessel breach (the drywell is steam inert, so the hydrogen does not burn there), then pushed through the suppression pool into the wetwell at vessel breach. Since the steam is condensed in the suppression pool, the hydrogen entering the wetwell will no longer be steam inert. Thus, a relatively large amount of hydrogen can be burned in the wetwell over a relatively short time.

Since ac power is available in a larger fraction of the S_2E sequence than in the TQUV sequence, the motor-operated valves in the vacuum breaker lines will be open for a greater percentage of the outcomes. Thus, there is a greater probability of induced penetration failures in S_2E than in TQUV.

3.5 Results for Sequoyah

3.5.1 Sequence S_2H

Table 3.18 summarizes the optimistic, central, and pessimistic containment failure modes for sequence S_2H at Sequoyah. The modes of containment failure range anywhere from no containment failure in the optimistic case to almost assured failure at the time of vessel breach due to steam spike/RCS depressurization/hydrogen combustion in the pessimistic case. In this sequence there is a high likelihood that the ice will be bypassed or completely melted at the time of vessel breach due to the initial success of ECC in the injection mode.

The differences in modes of containment failure between the three walkthroughs result from different estimates regarding hydrogen production and combustion. In the optimistic analysis, the hydrogen production is minimized and continuous localized burning near the igniters occurs. This results in very low pressure rises due to hydrogen combustion, and the containment does not fail early. Also in this walkthrough a coolable debris bed forms. This coupled with the availability of containment heat removal prevents late containment failure. Thus the majority of the outcomes result in no containment failure.

In the central walkthrough nearly half of the zirconium is predicted to oxidize before vessel breach. The sudden release of hydrogen from the RCS at vessel breach results in some burns of a global nature which threaten containment integrity. There is also some chance that the sprays fail after vessel breach due to debris in the recirculation sump, thus failing long term containment heat removal. In addition, a coolable debris bed may not form, leading to core-concrete attack and the associated release of noncondensible gases. Both of these factors result in a small chance (~9%) of late containment failure due to overpressurization, either by steam or noncondensible gases.

Pessimistically, all of the zirconium oxidizes before vessel breach. This results in a very large release of hydrogen at the time of the breach. A large hydrogen burn then occurs which is nearly certain to fail containment early. (It is worth noting that the combustion model matters very little here because the amount of hydrogen released from the RCS determines the magnitude of the burn.) Also, a small chance of containment failure is predicted due to vessel depressurization and steam spike alone.

We should further note that direct heating has no effect on the results for this sequence, because of the

massive amounts of water present in the reactor cavity and lower containment (See Section 2.4). In addition pessimistically, we estimate that an in-vessel steam explosion fails containment a very small fraction of the time (~1%).

3.5.2 Sequence S_2HF

Table 3.19 summarizes the reults of the three walk-throughs for sequence S_2HF at Sequoyah. The optimistic results are almost totally late overpressure failures. The central results are split evenly between late overpressure and early failure at the time of vessel breach, and the pessimistic results predict nearly assured failure at the time of vessel breach.

In the optimistic case, minimal amounts of zirconium are oxidized, and localized burning occurs in the vicinity of the igniters. This results in small containment pressure loadings and averts any early containment failure except that due to small preexisting leaks. The initiating sequence involves failure of core cooling and the containment sprays in the recirculation mode. Therefore, no long term containment cooling can be provided, and the containment is certain to fail due to the buildup of steam in the containment atmosphere. Because Sequoyah has a free standing steel containment and the Containment Performance Working Group provided us with information dismissing the possibility of leakage which would arrest the pressure buildup, we concluded that containment failure would always be a gross rupture.

In the central walkthrough more hydrogen was produced, and combustion was considered to be on a global level within any given compartment. These considerations lead to a high conditional probability (~50%) that containment will fail due to hydrogen combustion at vessel breach. If not, the containment most likely ruptures from late over-pressurization. The pessimistic results for S_2HF are identical to those for S_2H, discussed in Section 3.5.1.

3.5.3 Sequence S_3D

The optimistic, central, and pessimistic results for sequence S_3D at Sequoyah are summarized in Table 3.20. The most probable outcome is no containment failure in the optimistic and central cases. In the pessimistic walkthrough, the dominant result is containment failure due to a coincident steam spike and hydrogen burn at the time of vessel breach.

The differences in results among the three walkthroughs are primarily due to differences in estimates regarding hydrogen production and combustion. In the optimistic

case, minimal zirconium oxidation occurs along with localized burning of the hydrogen in the vicinity of the igniters. This combination results in very low pressure rises due to hydrogen combustion. Some probability exists that the RCS will undergo an induced failure which yields low pressure in the RCS at the time of vessel breach. This means a coolable debris bed may not be formed since a high pressure ejection is not available to disperse the debris. Without a coolable debris bed, an eventual basemat melt-through ensues.

In the central case, more substantial hydrogen burns are seen at the time of vessel breach, but the ice condenser acts to mitigate the pressure rise associated with them so that containment is most probably not threatened. However, coolable debris beds are considered less likely, and there is some chance of debris in the recirculation sump failing the sprays after vessel breach. Thus, the chance of late containment failure due to overpressure increases over that in the optimistic results. We also judge that, as for the Surry and Zion S_3D sequences, there is a small probability of inducing a small steam generator tube leak while the RCS remains at high pressure.

With two exceptions, the pessimistic results for S_3D are the same as for the S_2H and S_2HF sequences discussed previously. The first is that we estimate a small probability (~10%) that a small steam generator tube leak will be induced. Second, a remote possibility exists that the containment will fail by late overpressurization produced by the generation of noncondensibles.

3.5.4 Sequence TML

The results for all three walkthroughs are summarized in Table 3.21 for sequence TML at Sequoyah. As can be seen by comparing Table 3.21 with Table 3.20, the probabilities for all containment failure modes for all three walk-throughs are identical to those of S_3D.

3.5.5 Sequence TMLB'

Table 3.22 summarizes the results for the optimistic, central, and pessimistic walkthroughs for sequence TMLB' at Sequoyah. The highest probabilities are associated with no containment failure in the optimistic case and hydrogen combustion at the time of vessel breach in the central and pessimistic cases. Our analyses for this sequence did include the probability of restoration of power late in the accident (after vessel breach).

In the optimistic case minimal zirconium is oxidized which yields minimal hydrogen production. In addition, localized burning occurs in the vicinity of the igniters or of the debris in the reactor cavity (after vessel breach). The combination of these events results in small pressure rises due to hydrogen combustion. The recovery of ac power late in the accident reestablishes containment heat removal which prevents containment failure for the cases where coolable debris beds have formed. Those scenarios in which the debris is not coolable and ac power has been restored result in basemat meltthrough, whereas those in which ac power is not restored result in late containment failure due to overpressurization.

In the central walkthrough, a moderate amount (50%) of zirconium is oxidized and a corresponding amount of hydrogen is produced. Nearly all of this hydrogen is released to the lower compartment at the time of vessel breach. At that time, the core debris along with the accumulator water is ejected into the reactor cavity. The quenching of the core debris produces a steam spike which purges the hydrogen-rich atmosphere in the lower compartment into the upper regions of containment. All this happens very rapidly and results in a large hydrogen burn which is likely to fail containment. If containment does not fail from this source, direct heating provides another challenge. However, direct heating occurs in the lower compartment and its effect is largely quenched by the ice condenser. The key aspect of this sequence is that the reactor cavity is dry at the time of vessel breach. In the other sequences, the lower compartment contains such copious amounts of subcooled water that a large steam spike or direct heating is precluded from occurring.

The pessimistic case is similar to the central case, except for larger contributions from direct heating and late hydrogen burns.

3.6 Sensitivities

As mentioned in Section 2.3, the numerical values assigned to verbal descriptors such as "unlikely" or "remotely possible" are somewhat arbitrary, and the results could be sensitive to these choices. Accordingly, we performed a sensitivity study for an example sequence similar to Surry S_2D using the four alternative numerical sets in Table 2.7. The results, depicted in Table 3.23, indicate that the variation of conditional probability within each walkthrough (optimistic, central, pessimistic) is small compared to the difference in results between walkthroughs.

Although the results are not very sensitive to the choice of numerical values, they are sensitive to the choice of verbal descriptors, i.e., whether phenomena are considered "likely" or "unlikely" to occur. This sensitivity has been covered by the choice of optimistic, central, and pessimitic walkthroughs.

4. SUMMARY AND CONCLUSIONS

To interpret our results in the most illuminating way, it is necessary to clarify the meanings of the words "optimistic", "central", and "pessimistic". For this purpose, let us offer the following observations. Our central estimates are intended to provide a glimpse of the median of the reactor safety community. That is to say, we have a reasonable expectation that if the community were polled on the subjects treated in this document, a substantial fraction would respond that the reality of the situation is better than that depicted by our central estimates, and a comparably substantial fraction would respond that it is worse. Among those who think that reality is better (a group that we could refer to as "optimists"), a substantial fraction would claim that it is or could be as benign as our optimistic estimates imply. Similarly, many of the opposing group (the "pessimists") would claim that reality is or could be as unfavorable as our pessimistic estimates imply.

Viewed in this context, the results of our containment event analyses indicate quite clearly that there are large uncertainties in the reactor safety community's understanding of containment loading and response. The differences between the optimistic and pessimistic results attest that for many accident sequences, one cannot state the probabilities of the containment failure modes within narrow limits. However, one can identify the factors that drive the results and understand the reasons for the differences.

Fortunately, there are a limited number of factors that are governing, and they can be readily identified from our analysis. In the paragraphs below, we shall summarize, for each plant, the principal features of our results and the factors that drive them. In the process, we shall attempt to identify the sources of uncertainty which are principally responsible for the differences between the views of the optimist and the pessimist. Of course, the discussion will be limited only to those sequences which we have analyzed.

Also, in this summary, we shall briefly discuss the degree of coverage afforded by the BMI-2104 source term calculations and offer some suggestions for additional calculations. In general, we have found that the BMI calculations do treat some of the principal containment pathways for the sequences analyzed, but that there are also other sequences and other containment pathways which have significance to risk. A fairly comprehensive itemization of sequence-pathway combinations of potential importance is given in Tables 3.1 through 3.22. These tables also include specific reference to the existence or

lack of corresponding BMI source term calculations. In this section, we shall merely suggest a few additional calculations that would be helpful.

4.1 Surry

In the central walkthrough, the likelihood of early containment failure (before or at the time of vessel breach) has been estimated to be 0.0 to 0.05 for sequences in which containment sprays operate, and 0.0 to 0.005 for sequences in which they do not. The difference occurs because the atmosphere is flammable when the sprays operate but steam-inert when they do not. For all sequences, there is a reasonable likelihood that containment will not fail at all.

In the pessimistic walkthrough, the likelihoods of early containment failure are much higher. There are two causes, either of which is sufficient to produce the result. The first is the pessimistic assessment of direct heating; the second is the pessimistic view regarding the concurrent occurrence of a steam spike and hydrogen burn at the time of vessel breach. Factors which could reduce the importance of these phenomena in the pessimistic walkthrough are (1) less hydrogen generation, (2) a less coherent meltdown, (3) occurrence of temperature-induced failures in the reactor coolant system, and (4) a significantly higher containment failure pressure. Induced steam generator tube ruptures that could lead to a partial bypass of containment are potentially important for sequences where the reactor coolant system pressure remains high (e.g., TMLB'). In-vessel steam explosions could also be important in the pessimistic walkthrough if test data and analyses were to show that their influence is greater than what we attributed (see earlier text).

The source term calculations performed by BMI for Surry cover many of the important containment pathways, although the releases attributable to in-vessel steam explosion and direct heating scenarios were not treated. Also, there was no explicit calculation for S_3D; rather, the calculations were for S_2D. Additional analyses that could help complete the picture include: (1) TMLB' with an induced steam generator tube rupture, and (2) various pathways for the sequence S_3D.

4.2 Zion

The results for Zion are qualitatively similar to those for Surry, although the early containment failure probabilities are generally somewhat less. In the central walkthrough, these probabilities were evaluated to be 0.0 to 0.02 with the sprays or fan coolers operating, and 0.0

to 0.002 with the sprays and fan coolers failed. The factors driving the much higher probabilities in the pessimistic walkthrough are similar to Surry.

The BMI source term calculations for Zion do not include the possibility of containment failure, either early or late. Calculations for S_2D and TMLB' with containment failure occurring at vessel breach are particularly needed. Other S_2D and TMLB' pathways of interest include those involving preexisting containment leakage, and those involving some mode of late containment failure.

4.3 Peach Bottom

There are two fundamentally different kinds of accident sequences for Peach Bottom - those involving a failure of core cooling (AE, TB, TQUV) and those involving a failure or mismatch in containment cooling (TW, TC). For sequences involving failure of core cooling, our central walkthrough resulted in low to moderate probabilities of early containment failure (0.0 to 0.1). The pessimistic walkthrough resulted in somewhat higher probabilities (0.0 to 0.5). For sequences involving a failure or mismatch of containment cooling, pre-core-melt overpressurization and open containment venting became important modes of containment failure or release. (Open containment venting occurs when the operator opens a containment penetration according to the emergency procedure guidelines and does not reclose it when the core degrades.)

Containment overpressurization at Peach Bottom was generally assessed to result in failures of the drywell rather than the wetwell; thus the suppression pool would be bypassed following containment failure. Open containment venting, on the other hand, was assessed to be more likely from the wetwell than the drywell; thus the suppression pool would be utilized as a fission product scrubber. However, the pool would be saturated for sequences in which containment venting was initiated.

When early containment failure did occur during sequences for which there was no core cooling, it was because of phenomena occurring at the time of reactor vessel breach (i.e., vessel depressurization, ex-vessel steam spike or steam explosion, and/or direct heating). These are phenomena for which there is a notable lack of experimental data or analysis for BWRs. Our three walkthroughs varied widely for estimates made regarding the coherency of the meltdown, the magnitude of the thermal or chemical energy transfer, and the likelihood that the suppression pool vents would clear rapidly enough to relieve the pressures that occur during the interaction.

Reduction of the uncertainty in any of these issues could have a large effect on our results.

For sequences involving failure or mismatch of containment cooling, the primary sources of uncertainty related to the feasibility and effectiveness of the new emergency procedure guidelines. At issue are the operator's ability to vent containment by opening one or more containment penetrations after an isolation signal has occurred, his ability to maintain water delivery to the core indefinitely thereafter, and whether or not he recloses the vent if core degradation occurs.

BMI's source term calculations for Peach Bottom do not cover two important sequences: TB and TQUV. Source term calculations are needed particularly for TB (station blackout with delayed HPCI failure), with (1) containment failure occurring at vessel breach, and (2) containment failure occurring later as a result of the buildup of noncondensibles. Also, there is a need for a reevaluation of TC with core melt occurring (1) in an intact containment, and (2) in a containment that is intentionally vented from the wetwell.

4.4 Grand Gulf

As for Peach Bottom, some of the sequences in Grand Gulf involve a failure of core cooling (S_2E, TB, TQUV)and others involve a failure or mismatch in containment cooling (TPI, TC). For sequences involving failure of core cooling, the results of our central walkthrough indicated high probabilities of early containment failure (0.6-0.7). For sequences involving failure or mismatch in containment cooling, pre-core-melt overpressurization or open containment venting were the dominant failure or release modes. (Because of the containment design, the two were considered indistinguishable from the viewpoint of fission product releases.)

Suppression pool bypass either before or after containment failure was assessed to be rare in the central walkthrough but moderately likely in the pessimistic walkthrough. The likelihood of early bypass (i.e., bypass developing before or at the time of vessel breach) was estimated to be around 0.0 to 0.04 for the central walkthrough and 0.1 to 0.2 for the pessimistic walkthrough. Comparable likelihoods were estimated for late bypass.

For sequences involving failure of core cooling, the high probability of early containment failure was associated with two scenarios. The first was the combustion of hydrogen above the suppression pool before

the time of vessel breach. This failure mode occurred when the igniters were not operating because of ac power unavailability and the hydrogen was allowed to accumulate. The second was the concurrence of a steam spike in the drywell and a hydrogen burn in containment at the time of vessel breach. This scenario proceeded as follows: (1) early hydrogfn burns above the suppression pool forced water over the weir wall into the drywell; (2) core debris ejection into the water caused a steam spike; (3) the steam spike forced hydrogen rapidly through the suppression pool into the containment; and (4) the hydrogen burned and overpressurized containment. Availability of igniters did not mitigite this scenario. THe principal factor which could significantly reduce the likelihood of early containment failure from either of the two scenarios is a reduction in the amount of hydrogen produced both in-vessel and ex-vessel.

It is important to note that because of the nature of the containment, source terms at Grand Gulf are more affected by suppression pool bypass than by containment failure. Causes of induced suppression pool bypass include: (1) leakage around the personnel air locks or other penetrations caused by high drywell temperatures and pressures; (2) leakage through vacuum breakers caused by hydrogen combustion events above the suppression pool; (3) leakage through failed piping or drywell failure caused by local or global detonations above the suppression pool; (4) drywell failure caused by an ex-vessel steam explosion ordirect heating during which there is insufficient time for the suppression pool vents to clear; and (5) drywell failure caused by core debris attack on the reactor vessel pedestal. There are very few available experiments or analyses relating to these issues.

The BMI source term calculations for Grand Gulf did not treat the possibility of suppression pool bypass except for the S_2E sequence. Such calculations are needed for other sequences. Furthermore, there have been no calculations for the sequencf TB, a statin blackout with delayed failure of the reactor core isolation cooling (RCIC) system. Calculations for both TB and TQUV are needed with containment failing before or at vessel breach as the result of hydrogen combustion above the suppression pool.

4.5 Sequoyah

The results for different accident sequences in Sequoyah are principally shaped by whether significant amounts of water have been injected into containment and whether the ice has melted or bypass paths have developed before the time of vessel breach. Secondarily, the operation of the air return fans is also important. Thus it is convenient to divide the Sequoyah sequences into

three groups: (1) sequences S_2H and S_2HF, for which there is a large amount of water and a high likelihood of ice melting or bypass before vessel breach; (2) sequence TMLB; for which there is little water, a low probability of ice melting or bypass, and no air return fans; and (3) sequences TML and S_3D, which are similar to Group 1 except for a low likelihood of ice melting or bypass.

In the central walkthrough, the probabilities of early containment failure were estimated to be about 0.5 for Group (1), 0.9 for Group (2), and 0.01 for Group (3). The remaining likelihood was principally attributed to late overpressurization or no containment failure, depending upon whether the containment sprays were operating. In the pessimistic walkthrough, the likelihood of early containment failure was very high for all sequences.

For sequences in Group (1), early containment failure was principally caused by hydrogen burning at the time of vessel breach. Even though the igniters were operating, a large burn was predicted as a result of the rapid release of hydrogen into containment at vessel breach. Because the ice had previously melted, the burn itself was sufficient to fail containment with or without a steam spike. Direct heating was not a factor because of the copious amounts of water in the reactor cavity and on the containment floor. Factors which could reduce the likelihood of early failure are (1) less hydrogen generation overall and (2) a lower hydrogen release from the reactor coolant system when the vessel is breached.

For the sequence in Group (2), the occurrence of early containment failure was associated with two possible occurrences: (1) a concurrent steam spike and hydrogen burn at vessel breach and (2) direct heating. The igniters were not operating at the time because of the unavailability of ac power. The likelihood of early containment failure could be reduced by any of the following factors: (1) less hydrogen generation, (2) less coherency in the meltdown process, (3) occurrence of temperature-induced reactor coolant system failure before vessel breach, and (4) a significantly higher containment failure pressure.

For sequences in Group (3), early containment failure was caused by the concurrence of a steam spike and hydrogen burn, even though the igniters were operating. Factors which could reduce the likelihood of early failure are those identified for Groups (1) and (2).

As mentioned earlier, induced steam generator tube ruptures leading to a partial bypass of containment are potentially important. In-vessel steam explosions could also be important, particularly if test data and analyses

were to show that their influence is greater than what we attributed.

BMI's source term calculations for Sequoyah miss two sequences that are probabilistically very significant: S_2H and S_3D. A source term calculation is needed for S_2H, particularly, with containment failing at vessel breach.

REFERENCES

1. (CLWG). Report of the Containment Loads Working Group, U. S. Nuclear Regulatory Commission, to be published.

2. (CLWG). Concensus Summaries for Standard Problems 1 through 6. Letter Reports Addressed to J. Telford, U. S. Nuclear Regulatory Commission, May-June, 1984.

3. (CLWG). F. E. Haskin, et al., Combustion-Induced Loads in Large Dry PWR Containments, Proceedings of the 2nd Containment Integrity Workshop, NUREG/CP-0056, SAND84-1514, August 1984. Also, letter to M. Silberburg, U. S. Nuclear Regulatory Commission, March 9, 1984.

4. (CPWG). Containment Performance Working Group, Containment Leak Rate Estimates, NUREG-1037, U. S. Nuclear Regulatory Commission, to be published.

5. (BMI). J. A. Giesecke, et al., "Radionuclide Release Under Specific Accident Conditions," BMI-2104, Battelle Columbus Laboratories, Volumes II through VI, July, 1984.

6. (QUEST). R. J. Lipinski, et al., "Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions," SAND84-0410, Sandia National Laboratories, February, 1985.

7. (IDCOR). IDCOR Program Reports, IDCOR Tasks 21.1 (Risk Reduction Potential) and 23.1 (Integrated Containment Analysis), Technology for Energy Corporation, November, 1984.

8. (SASA). F. E. Haskin, et al., "Analysis of Hypothetical Severe Core Damage Accidents for the Zion Pressurized Water Reactor," NUREG/CR-1989, SAND81-0504, Sandia National Laboratories, October, 1982.

9. (SASA). A. L. Camp, et al., "MARCH-HECTR Analysis of Selected Accidents in an Ice-Condenser Containment," NUREG/CR-3912, SAND83-0501, Sandia National Laboratories, December, 1984.

10. (SASA). D. H. Cook, et al., "Loss of Decay Heat Removal Sequences at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-2973, ORNL/TM-8532, Oak Ridge National Laboratory, May, 1983.

11. (SASA). R. M. Harrington, et al., "ATWS at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-3470, ORNL/TM-8902, Oak Ridge National Laboratory, July, 1984.

12. (SASA). S. Z. Bruske, et al., "Severe Accident Sequence Analysis (SASA) Program Sequence Event Tree: Boiling Water Reactor Anticipated Transient Without Scram," NUREG/CR-4165, EGG-2379 (Draft), Idaho National Engineering Laboratory, February 1985.

13. (SAUNA). J. B. Rivard, et al., "Identification of Severe Accident Uncertainties," NUREG/CR-3440, SAND83-1689, Sandia National Laboratories, September 1984.

14. (ASEP). A. M. Kolaczkowski, et al., "Interim Report on Accident Sequence Likelihood Reassessment," Draft Report, Sandia National Laboratories, August 1983.

15. (ASEP). F. T. Harper, et al., "Accident Sequence Likelihood Information," Appendix to NUREG-0956, to be published.

16. (PATF). A. S. Benjamin, et al., "SARRP - Risk Rebaselining and Risk Reduction Analysis," <u>11th Water Reactor Safety Research Information Meeting</u>, NUREG/CP-0048, Vol. 3, January 1984.

17. (RSS). "Reactor Safety Study," WASH-1400, NUREG-75/014, U. S. Nuclear Regulatory Commission, October 1975.

18. (RSSMAP). "Reactor Safety Study, Methodology Applications Program," NUREG/CR-1659, SAND80-1897, Volumes 1-4, Sandia National Laboratories, 1981-1982.

19. (ZPSS). "Zion Probabilistic Safety Study," Commonwealth Edison Co., 1981.

20. (SPSS). "Sizewell B Probabilistic Safety Study," WCAP-9991, Westinghouse Electric Co., 1982.

21. (SPSA). "Seabrook Plant Probabilistic Safety Analysis," Yankee Atomic Electric Co., December 1983.

22. (GESSAR). "GESSAR-II, BWR/6 Standard Plant Probabilistic Risk Assessment," General Electric Co., 1982.

23. (FSAR). Final Safety Analysis Reports for Surry Unit 1, Zion Units 1,2, Peach Bottom Unit 2, Grand Gulf Unit 1, and Sequoyah Unit 1.

24. (S&W). A. Drozd, et al., "Parametric Study of Aerosol Behavior Following AB and TMLB Accidents," <u>ANS 1984 Annual Meeting</u>, TP84-54, Stone and Webster Engineering Corp., 1984.

25. (AMES). L. Greimann, et al., "Final Report, Containment Analysis Techniques, A State-of-the-Art Summary," NUREG/CR-3653, SAND83-7463, Ames Laboratory and Sandia National Laboratories, March 1984.

26. (FVCS). F. T. Harper, et al., "The Effect of Filtered-Vented Containment Systems on Severe Accident Frequencies and Consequences for a Mark III Boiling Water Reactor," NUREG/CR-4066, SAND83-2441, Sandia National Laboratories to be published.

27. (FVCS). A. S. Benjamin, et al., "Value Impact Assessment of Filtered-Vented Containment Systems and Other Safety Options for a BWR Mark I Containment," NUREG/CR-4065, SAND83-1861, Sandia National Laboratories, to be published.

28. (NRR). M. B. Weinstein, "Primary Containment Leakage Integrity: Availability and Review of Failure Experience," Nuclear Safety, 21, 1980.

29. (BNL). W. T. Pratt, et al., "Containment Response During Degraded Core Accidents Initiated by Transients and Small Break LOCA in the Zion/Indian Point Reactor Plants," NUREG/CR-2228, BNL-NUREG-51415, Brookhaven National Laboratories, July 1981.

30. (A-43). A. W. Serkiz, "Containment Emergency Sump Performance, Technical Findings Related to Unresolved Safety Issue A-43," NUREG-0897, U. S. Nuclear Regulatory Commission, April 1983.

31. (NRR). W. G. Lyons, Presentation at NRC/IDCOR Meeting on Containment Loads and Fission Product Behavior, U. S. Nuclear Regulatory Commission, May 15-17, 1984.

32. (ZIP). W. B. Murfin, et al., "Report of the Zion/Indian Point Study," NUREG/CR-1410, SAND80-0617, Sandia National Laboratories, August 1980.

33. (HYDR). A. L. Camp, et al., "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, SAND82-1137, Sandia National Laboratories, August 1983.

34. (IVSE). M. Berman, et al., "An Uncertainty Study of PWR Steam Explosions," NUREG/CR-3369, SAND83-1438, Sandia National Laboratoriess, May 1984.

35. (LPS). S. J. Niemczyk, et al., "The Consequences from Liquid Pathways After a Meltdown Accident," NUREG/CR-1596, SAND80-1669, Sandia National Laboratories, June 1981.

36. (EPRI). "Loss of Offsite Power at Nuclear Power Plants: Data and Analysis," EPRI NP-2301, Interim Report, Electric Power Research Institute, March 1982.

TABLE 2.1. ISSUES TO BE ADDRESSED AND RELEVANCE OF RECENT INFORMATION SOURCES*

	BASES FOR BINNING SOURCE TERMS: ZION	SIZEWELL	SEABROOK	INFORMATION SOURCES: CLWG	CPWG	BMI-2104	QUEST	IDCOR
1. Size of Preexisting Containment Leakage.	X	X	X	--	X	--	--	--
2. Size and Location of the Primary System Break During the Melt Release (e.g., Hot Leg, Cold Leg, PORV, Steam Generator Tube, LPIS Check Valve).	--	--	--	--	--	--	--	--
3. Timing of Accumulator Discharge Relative to Timing of Reactor Vessel Breach.	--	--	X	--	--	X	--	X
4. Occurrence of In-Vessel Steam Explosion Large Enough to Fail the Reactor Vessel.	--	--	--	--	--	--	--	--
5. Occurrence of In-Vessel Steam Explosion Large Enough to Fail the Containment.	X	X	X	--	--	X	--	X
6. Timing and Magnitude of Early Hydrogen Burns.	X	X	X	X	--	X	--	X
7. Magnitude of the Ex-Vessel Steam Spike.	--	--	X	X	--	X	X	X
8. Extent of Direct Heating of the Atmosphere Following Vessel Breach.	--	--	--	X	--	--	--	--
9. Containment Structural Failure Pressure	X	X	X	--	--	--	--	X
10. Size of Containment Leakage Induced by Temperature or Pressure.	--	--	X	--	X	--	--	--
11. Survivability of Containment Sprays and Fan Coolers at Various Times during the Accident.	--	--	--	--	--	--	--	--
12. Survivability of Suppression Pools and Ice Condensers at Various Times During the Accident.	--	--	--	--	--	--	--	--
13. Extent of Core-Concrete Interaction.	--	X	--	X	--	X	--	X
14. Timing and Magnitude of Late Hydrogen Burns.	--	--	X	X	--	X	--	X
15. Potential for Direct Core Debris Attack on Containment Structures.	--	--	--	X	--	--	--	--
16. Potential for Core Debris to Melt Through the Basemat.	X	X	X	--	--	X	--	X
17. Potential for Basemat Meltthrough to Cause a Depressurization of Containment.	--	--	--	--	--	X	--	--
18. Potential for Effluent to Pass Through Adjacent Structures, such as the Auxiliary Building.	--	--	--	--	--	X	--	X

*Based on information made available prior to February 1985. X marks where issue was used for a basis or where information source is applicable.

TABLE 2.2. EVENT DESCRIPTIONS FOR PWR SUBATMOSPHERIC AND LARGE-DRY CONTAINMENTS

QUESTION ASKED BY EVENT TREE	PRIOR QUESTION DEPENDENCIES
1. Is ac power available after the initiating event?	None
2. Does core water makeup fail before containment overpressurization?	None
3. What is the level of preexisting containment leakage or isolation failure?	1
4. Where is the initial reactor coolant system break?	None
5. What is the size of the initial reactor coolant system break?	None
6. Is the containment initially bypassed?	4
7. Are the steam generators wet or dry?	None
8. Do the fan coolers fail to actuate before core degradation?	1
9. Do the containment sprays fail to actuate in the injection mode before core degradation?	1,3,5,6
10. Do the containment sprays fail to actuate in the recirculation mode before core degradation?	9
11. To what degree, if any, is the auxiliary building initially bypassed?	3,4,7
12. Where, if at all, is there a temperature-induced failure of the reactor coolant system during the period of core degradation?	5
13. What is the size of the temperature-induced reactor coolant system failure?	12
14. What is the primary system pressure during core degradation? Also, what would be the containment pressure increment from a primary system blowdown?	5,12,13
15. At what level, if any, is containment bypassed during core degradation?	4,5,12,13
16. What is the rate of blowdown to containment during core degradation?	5,6,13,15
17. Do the containment sprays fail to actuate during the period of core degradation?	1,3,10,16
18. Is there containment heat removal during core degradation?	8,17
19. At what level, if any, does containment fail due to steam pressurization before vessel breach?	2,3,15,18
20. What is the containment pressure before vessel breach?	3,16,18,19
21. Is there a hydrogen burn before vessel breach? Also, what is the pressure increment from such a burn?	3,16,18
22. Does containment fail because of a hydrogen burn before vessel breach? Also, what is the associated failure pressure and standard deviation?	20,21
23. To what degree, if any, is the auxiliary building bypassed before vessel breach?	7,11,12,19,22
24. Do the fan coolers fail after the early hydrogen burn?	8,21
25. Do the containment sprays fail after the early hydrogen burn?	1,17,21,22
26. Is there containment heat removal after the early hydrogen burn?	24,25
27. What is the mode of reactor vessel breach?	14
28. Does direct heating of the containment atmosphere occur just after vessel breach? Also, what is the pressure increment from a steam spike alone and and from a steam spike plus direct heating?	14,17,27
29. Is there a hydrogen burn just after vessel breach? Also, what is the pressure increment from such a burn? Also, what is the associated failure pressure and standard deviation?	3,16,21,26
30. Does containment fail due to a steam spike, direct heating, and/or hydrogen burn just after vessel breach?	14,20,28,29
31. What is the mode of containment failure, if any, just after vessel breach?	3,19,22,27,28,29,30
32. To what degree, if any, is the auxiliary building bypassed just after vessel breach?	23,31

(Continued)

TABLE 2.2. EVENT DESCRIPTIONS FOR PWR SUBATMOSPHERIC AND LARGE-DRY CONTAINMENTS (CONT'D)

QUESTION ASKED BY EVENT TREE	PRIOR QUESTIONS DEPENDENCIES
33. Do the fan coolers fail after vessel breach?	24,28,29
34. Do the containment sprays fail after vessel breach?	1,25,31
35. Is there an oxidation release?	27,28
36. Is ac power restored after vessel breach?	1
37. Do the fan coolers fail late in the accident?	33,36
38. Do the containment sprays fail late in the accident?	34,36
39. Is there containment heat removal late in the accident?	37,38
40. Has the contents of the refueling water storage tank been injected into containment?	9,17,25,34,38
41. What is the amount of water injected into containment?	2,14,16,40
42. Do significant core-concrete interactions occur after vessel breach?	14,18,27,34,41
43. What is the containment pressure late in the accident (18 hours)?	26,39,42
44. In what way, if at all, does containment fail due to gradual pressurization from the boiloff of water in containment (leakage or gross rupture)?	3,15,19,22,31, 41,43
45. Has a large leak or gross failure occurred late in the accident?	3,15,19,22,31, 44
46. What pressure rise would occur if combustible gases were to burn late in the accident?	26,28,39,42
47. Would containment fail if such a burn were to occur? Also, what is the associated failure pressure and standard deviation?	21,29,43,46
48. Does a late hydrogen burn actually occur?	18,21,26,29,37, 39,42,45
49. Does containment fail due to a late hydrogen burn?	47,48
50. To what degree, if any, is the auxiliary building bypassed late in the accident?	32,44,49
51. Do the fan coolers fail after the late hydrogen burn?	37,48
52. Do the containment sprays fail after the late hydrogen burn?	38,48,49
53. Is there containment heat removal very late in the accident?	51,52
54. Is there a vaporization release?	17,25,34,42
55. To what degree, if any, does containment leakage occur due to high temperatures very late in the accident?	41,45,49,53
56. Does basemat meltthrough occur, given no prior containment failure?	26,38,39,45, 49,53,54,55
57. Does containment depressurize after basemat meltthrough?	26,39,53,56
58. What is the ultimate containment failure mode, if any, resulting from core-concrete interactions?	42,45,49,53, 55,56,57
59. To what degree, if any, is the auxiliary building ultimately bypassed very late in the accident?	50,55,57

TABLE 2.3. EVENT DESCRIPTIONS AND LIKELIHOODS FOR SURRY S_2D AND S_3D

EVENT	PRIOR EVENTS	BRANCHING OPTIONS *	LIKELIHOOD OPTIMISTIC	LIKELIHOOD CENTRAL	LIKELIHOOD PESSIMISTIC
3. Preexisting containment leakage or isolation failure		(i) Tech. specs. or less	0.987	0.970	0.947
		(ii) Greater than tech. specs. but insufficient to preclude gradual overpressurization later in accident	0.013	0.028	0.051
		(iii) Greater than (ii) but insufficient to depressurize containment in about 2 hours	0.0	0.002	0.002
		(iv) Greater than (iii)	0.0	0.0	0.0
4. Initial reactor coolant system break location	(a) S_2D Sequence	(i) Hot Leg	Unlikely	Unlikely	Unlikely
		(ii) Cold Leg	Likely	Likely	Likely
		(iii) Other	0.0	0.0	0.0
	(b) S_3D Sequence	(i) Hot Leg	0.0	0.0	0.0
		(ii) Cold Leg	1.0	1.0	1.0
		(iii) Other	0.0	0.0	0.0
9. Containment Spray Failure to Actuate Before Core Degradation	(a) No Large Containment leakage or bypass (level iii or greater)		0.0	0.0	0.0
	(b) Large containment leakage or bypass (level iii or greater)		1.0	1.0	1.0
11. Degree of Auxiliary Building Breakthrough or Bypass, Given Preexisting Containment Leakage or Isolation Failure		(i) No Bypass	Likely	Likely	Likely
		(ii) Partial Bypass Caused by Building Damage or Release at Upper Levels of Building	Unlikely	Unlikely	Unlikely
		(iii) Total Bypass	0.0	0.0	0.0
12. Reactor Coolant System Failure Before Vessel Breach	(a) S_2D Sequence		0.0	0.0	0.0
	(b) S_3D Sequence	(i) Hot Leg	Unlikely	Moderately Possible	Impossible
		(ii) Cold Leg	Likely	Moderately Possible	Unlikely
		(iii) Steam Generator Tube Rupture	Impossible	Remotely Possible	Unlikely
		(iv) None	Impossible	Indeterminate	Likely
13. Size of Induced RCS Failure	(a) Induced Failure in Hot Leg	(i) Large LOCA (A)	1.0	1.0	1.0
		(ii) Small LOCA (S_2)	0.0	0.0	0.0
		(iii) Other	0.0	0.0	0.0
	(b) Induced Failure in Cold Leg or Steam Generator Tube	(i) Large LOCA (A)	0.0	0.0	0.0
		(ii) Small LOCA (S_2)	1.0	1.0	1.0
		(iii) Other	0.0	0.0	0.0

* Lack of an entry in the column entitled "Branching Options" denotes that the positive path is assumed (i.e., the path corresponding to a "Yes" answer). The appropriate likelihood values for the negative path are the complement of the values given for the positive path.

(Continued)

TABLE 2.3. EVENT DESCRIPTIONS AND LIKELIHOODS FOR SURRY S_2D AND S_3D CONT'D

EVENT	PRIOR EVENTS	BRANCHING OPTIONS	LIKELIHOOD OPTIMISTIC	CENTRAL	PESSIMISTIC
17. Containment Spray Failure to Actuate After Induced LOCA	(a) Large Induced LOCA and No Gross Preexisting Leakage (Level iv)		0.0	0.0	0.0
	(b) Small Induced LOCA and No Large Preexisting Leakage (Level iii or Greater)		0.0	0.0	0.0
	(c) Otherwise		1.0	1.0	1.0
21. Hydrogen Burn Before Vessel Breach	(a) Induced Large LOCA and No Large Preexisting Leakage (Level iii or Greater)		Impossible	Impossible	Unlikely
	(b) Otherwise		0.0	0.0	0.0
22. Containment Failure from Hydrogen Burn Before Vessel Breach			Calculated from Containment Pressure Loading and Capacity, See Tables 2.4 and 2.5.		
23. Auxiliary Building Breakthrough or Bypass Given Containment Failure from Hydrogen Burn Before Vessel Breach		(i) No Bypass	0.0	0.0	0.0
		(ii) Partial Bypass Due to Building Damage or Release at Upper Levels of Building	Unlikely	Remotely Possible	Impossible
		(iii) Total Bypass	Likely	Almost Certain	Certain
25. Containment Spray Failure, Given Hydrogen Burn Before Vessel Breach	(a) Containment Not Failed		Remotely Possible	Remotely Possible	Remotely Possible
	(b) Containment Failed	-	Unlikely	Unlikely	Indeterminate
27. Mode of Vessel Breach	(a) No Induced Large LOCA	(i) Steam Explosion-Upper Head Failure (α mode)	0.0	0.0001	0.01*
		(ii) Steam Explosion-Bottom Head Failure	0.0	0.21	0.47*
		(iii) High Pressure Ejection	1.0	0.79	0.52
		(iv) Meltthrough	0.0	0.0	0.0
	(b) Induced Large LOCA	(i) Steam Explosion-Upper Head Failure (α mode)	0.0	0.0001	0.01*
		(ii) Steam Explosion-Bottom Head Failure	0.0	0.21	0.47*
		(iii) High Pressure Ejection	0.0	0.0	0.0
		(iv) Meltthrough	1.0	0.79	0.52
28. Occurrence of Direct Heating at Vessel Breach	(a) Induced Large LOCA and no Steam Explosion-Upper Head Failure		0.0	0.0	0.0
	(b) Otherwise		0.0	1.0	1.0
29. Occurrence of Hydrogen Burn at Vessel Breach	(a) No Preexisting Large Containment Leakage (Level iii or Greater) and No Earlier Hydrogen Burn		Likely	Likely	Likely
	(b) Large Preexisting Containment Leakage and No Induced Large LOCA and No Earlier Hydrogen Burn		Impossible	Likely	Likely
	(c) Otherwise		0.0	0.0	0.0

*Recent test data have not been factored into these estimates, see text.

(Continued)

TABLE 2.3. EVENT DESCRIPTIONS AND LIKELIHOODS FOR SURRY S_2D AND S_3D
CONT'D

EVENT	PRIOR EVENTS	BRANCHING OPTIONS	LIKELIHOOD OPTIMISTIC	CENTRAL	PESSIMISTIC
30. Containment Failure from Steam Spike, Direct Heating, and/or Hydrogen Burn at Vessel Breach			Calculated from Containment Pressure Loading and Capacity, See Tables 2.4 and 2.5.		
32. Auxiliary Building Breakthrough or Bypass Given Containment Failure at Vessel Breach	(a) Steam Explosion α-mode Failure	(i) No Bypass	0.0	0.0	0.0
		(ii) Partial Bypass	0.0	0.0	0.0
		(iii) Total Bypass	1.0	1.0	1.0
	(b) Other Failure Modes	(i) No Bypass	0.0	0.0	0.0
		(ii) Partial Bypass	Unlikely	Remotely Possible	Impossible
		(iii) Total Bypass	Likely	Almost Certain	Certain
34. Containment Spray Failure after Vessel Breach	(a) Containment not Failed		Remotely Possible	Unlikely	Indeterminate
	(b) Containment Failed, but not α-mode		Unlikely	Unlikely	Indeterminate
	(c) α-mode Failure		1.0	1.0	1.0
35. Oxidation Release	(a) Direct Heating or In-Vessel Steam Explosion Causing Vessel Failure		1.0	1.0	1.0
	(b) Otherwise		0.0	0.0	0.0
42. Significant Core-Concrete Interactions after Vessel Breach	(a) Vessel Failure from Steam Explosion	-	Remotely Possible	Unlikely	Indeterminate
	(b) Ejection at High Pressure (S_3D)		Remotely Possible	Unlikely	Indeterminate
	(c) Ejection at Intermediate pressure (S_2D, or S_3D with Induced Small LOCA or Steam Generator Tube Rupture)		Unlikely	Likely	Almost Certain
	(d) Ejection at Low Pressure (S_3D with Induced Large LOCA)		Indeterminate	Likely	Almost Certain
44. Containment Failure from Pressurization Due to Gradual Boiloff of Water in Reactor Cavity	(a) No Prior Containment Failure, Bypass, or Leakage, No Late Sprays, and Debris Bed is Coolable	(i) No Containment Failure	0.0	0.0	0.0
		(ii) Containment Leakage Sufficient to Preclude Further Pressurization	Likely	Remotely Possible	Impossible
		(iii) Containment Structural Failure	Unlikely	Almost Certain	Certain
	(b) Same as (a), but Debris Bed is not Coolable	(i) No Containment Failure	1.0	0.999	0.84
		(ii) Containment Leakage	0.0	0.001	0.0
		(iii) Containment Structural Failure	0.0	0.0	0.16
	(c) Otherwise	(i) No Containment Failure	1.0	1.0	1.0
		(ii) Containment Leakage	0.0	0.0	0.0
		(iii) Containment Structural Failure	0.0	0.0	0.0

(Continued)

TABLE 2.3. EVENT DESCRIPTIONS AND LIKELIHOODS FOR SURRY S_2D AND S_3D
CONT'D

EVENT	PRIOR EVENTS	BRANCHING OPTIONS	LIKELIHOOD OPTIMISTIC	CENTRAL	PESSIMISTI
48. Occurrence of Late Hydrogen Burn	(a) No Prior Burn, No Prior Containment Failure, and Containment Sprays Operable Late		1.0	1.0	1.0
	(b) No Prior Containment Failure, Debris Bed is Coolable, and no Late Containment Sprays		Indeterminate	Indeterminate	Indeterminate
	(c) Otherwise		0.0	0.0	0.0
49. Containment Failure from Late Hydrogen Burn		(i) No Containment Failure	Complement of (ii) and (iii)		
		(ii) Containment Leakage	0.0	0.0	0.0
		(iii) Gross Structural Failure	Calculated from Containment Pressure Loading and Capacity, See Tables 2.4 and 2.5.		
50. Auxiliary Building Breakthrough or Bypass, Given Late Containment Failure	(a) Containment Leakage	(i) No Bypass	Likely	Indeterminate	Unlikely
		(ii) Partial Bypass	Unlikely	Indeterminate	Likely
		(iii) Total Bypass	0.0	0.0	0.0
	(b) Gross Containment Structural Failure	(i) No Bypass	0.0	0.0	0.0
		(ii) Partial Bypass	Unlikely	Remotely Possible	Impossible
		(iii) Total Bypass	Likely	Almost Certain	Certain
52. Late Containment Spray Failure, Given No Earlier Failure	(a) No Containment Failure		Remotely Possible	Unlikely	Indeterminate
	(b) Containment Failure from Late Hydrogen Burn		Unlikely	Unlikely	Indeterminate
54. Vaporization Release	(a) No Containment Sprays and Debris Bed Not Coolable		1.0	1.0	1.0
	(b) Otherwise		0.0	0.0	0.0
55. Late Containment Leakage Due to High Temperature			0.0	0.0	0.0
56. Basemat Meltthrough Before Containment Overpressurization	(a) Containment Spray Injection Operable at Some Time During Accident		Unlikely	Unlikely	Remotely Possible
	(b) No Containment Spray Injection		Indeterminate	Indeterminate	Unlikely
57. Containment Depressurization, Given Basemat Meltthrough	(a) Containment Spray Injection		Likely	Unlikely	Remotely Possible
	(b) No Containment Spray Injection		Likely	Indeterminate	Unlikely

(Continued)

TABLE 2.3. EVENT DESCRIPTIONS AND LIKELIHOODS FOR SURRY S_2D AND S_3D
CONT'D

EVENT	PRIOR EVENTS	BRANCHING OPTIONS	LIKELIHOOD OPTIMISTIC	LIKELIHOOD CENTRAL	LIKELIHOOD PESSIMISTIC
58. Mode of Very Late Containment Failure Given no Prior Containment Failure	(a) Containment Sprays Operable Late but Debris Bed not Coolable	(i) No Containment Failure	Likely	Indeterminate	Unlikely
		(ii) Basemat Meltthrough Only	Unlikely	Indeterminate	Likely
		(iii) Containment Leakage due to Pressurization during Core-Concrete Interaction	0.0	0.0	0.0
		(iv) Gross Containment Structural Failure Due to Pressurization During Core-Concrete Interaction	0.0	0.0	0.0
	(b) Containment Sprays not Operable Late but no Basemat Meltthrough	(i) No Failure	Unlikely	Impossible	Impossible
		(ii) Basemat Meltthrough Only	0.0	0.0	0.0
		(iii) Leakage Due to Pressurization during Core-Concrete	Likely	Unlikely	Impossible
		(iv) Structural Failure due to Pressurization during Core-Concrete	Unlikely	Likely	Certain
	(c) Containment Sprays not Operable Late, Basemat Meltthrough Occurs, but no Depressurization	(i) No Failure	0.0	0.0	0.0
		(ii) Basemat Meltthrough Only	Unlikely	Impossible	Impossible
		(iii) Leakage due to Pressurization during Core-Concrete	Likely	Unlikely	Impossible
		(iv) Structural Failure due to Pressurization during Core-Concrete	Unlikely	Likely	Certain
	(d) Same as (c), but Containment Depressurizes Following Basemat Meltthrough	(i), (iii), (iv)	0.0	0.0	0.0
		(ii) Basemat Meltthrough Only	1.0	1.0	1.0
59. Auxiliary Building Breakthrough or Bypass, Given Very Late Containment Failure	(a) Containment Leakage	(i) No Bypass	Likely	Indeterminate	Unlikely
		(ii) Partial Bypass	Unlikely	Indeterminate	Likely
		(iii) Total Bypass	0.0	0.0	0.0
	(b) Gross Containment Structural Failure	(i) No Bypass	0.0	0.0	0.0
		(ii) Partial Bypass	Unlikely	Remotely Possible	Impossible
		(iii) Total Bypass	Likely	Almost Certain	Certain

TABLE 2.4. CONTAINMENT LOADINGS FOR SURRY S_2D AND S_3D

SOURCE OF PRESSURE	PRIOR EVENTS	PRESSURES (PSIG) OPTIMISTIC	CENTRAL	PESSIMISTIC
(1) Containment pressure before vessel breach	(a) No large preexisting containment leakage and no sprays	25	25	25
	(b) No large preexisting leakage and sprays operable	5	5	5
	(c) Large preexisting leakage (level iii) and no sprays	5	5	5
	(d) Large preexisting leakage and sprays operable	0	0	0
(2) Containment pressure increment from a hydrogen burn before vessel breach, given containment sprays operating	(a) No large preexisting containment leakage and low RCS pressure (S_3D with induced hot-leg LOCA)	0	25	50
	(b) Large preexisting containment leakage and low RCS pressure	0	0	35
	(c) Otherwise	0	0	0
(3) Total containment pressure following early hydrogen burn		Sum of pressures in (1) and (2)		
(4) Containment pressure increment from reactor coolant system blowdown at vessel breach	(a) High RCS pressure (S_3D with no induced RCS failure)	15	15	15
	(b) Intermediate RCS pressure (S_2D, or S_3D with induced cold-leg LOCA or steam generator tube rupture)	7	7	7
	(c) Low RCS pressure (S_3D with induced hot-leg LOCA)	0	0	0
(5) Containment pressure increment from steam spike at vessel breach	(a) High RCS pressure (accumulator discharge at vessel breach)	0	15	26
	(b) Intermediate or low RCS pressure (early accumulator discharge) and sprays operating	0	15	26
	(c) Intermediate or low RCS pressure and sprays never on	0	2	4
(6) Containment pressure increment from combination of steam spike and direct heating at vessel breach	(a) High RCS pressure or steam explosion failing upper head	11	44	104
	(b) Intermediate RCS pressure, no steam explosion failing upper head, and containment sprays operating	11	32	65
	(c) Same as (b), but containment sprays never on	0	8	24
	(d) Low RCS pressure and no steam explosion failing upper head	Same pressures as in (5b), (5c)		
(7) Containment pressure increment from hydrogen burn at vessel breach, given containment sprays operating	(a) No large preexisting containment leakage and no prior burn	25	35	70
	(b) Large preexisting leakage (level iii) and no prior burn	0	25	55
	(c) Otherwise	0	0	0
(8) Total containment pressure following vessel breach	(a) Steam spike only	Sum of pressures in (1), (4), and (5)		
	(b) Steam spike and direct heating	Sum of pressures in (1), (4), and (6)		
	(c) Steam spike and hydrogen burn	Sum of pressures in (1), (4), (5), and (7)		
	(d) Steam spike, direct heating, and hydrogen burn	Sum of pressures in (1), (4), (6), and (7)		

(Continued)

TABLE 2.4. CONTAINMENT LOADINGS FOR SURRY S_2D AND S_3D
(CONT'D)

SOURCE OF PRESSURE	PRIOR EVENTS	PRESSURES (PSIG) OPTIMISTIC	CENTRAL	PESSIMISTIC
(9) Total containment pressure late in accident (18 hrs.)	(a) Sprays never operate and debris bed is not coolable	70	70	70
	(b) Sprays never operate and debris bed is coolable	145	145	145
	(c) Sprays operate early or late but not both	5	15	25
	(d) Sprays operate early and late, debris bed is not coolable	10	10	10
	(e) Sprays operate early and late, debris bed is coolable	5	5	5
(10) Containment pressure increment from a late hydrogen burn, given containment sprays operating at some time	(a) Debris bed is not coolable or is coolable as a result of dispersal associated with direct heating	55	110	110
	(b) Debris is coolable in reactor cavity	55	70	70
(11) Total containment pressure after late hydrogen burn		Sum of pressures in (9) and (10)		

TABLE 2.5. SURRY CONTAINMENT CAPACITY ESTIMATES

(a) Leakage Area Versus Pressure

PRESSURE (psig)	LEAK AREA (in^2) OPTIMISTIC	CENTRAL	PESSIMISTIC	
0	.004	.004	.004	
45	.004	.004	.004	(linear
119	.004	.004	.36	variation)

(b) Structural Failure Pressure

	PRESSURE (psig) OPTIMISTIC	CENTRAL	PESSIMISTIC
Mean Structural Failure Pressure	119	119	85
Standard Deviation	2.5	15	15

Table 2.6. Basis for Pressure Estimates for Concurrent Direct Heating/Steam Spike

		Optimistic	Central	Pessimistic
(1)	Percent of Core Melt Ejected at the Time of Vessel Breach	50%	75%	100%
(2)	Percent of Amount in (1) Quenched Thermally in water	80%	75%	50%
(3)	Percent of Amount in (2) which Reacts with Steam	0%	30%	30%
(4)	Percent of Amount in (1) Quenched Thermally in Atmosphere	2%	28%	50%
(5)	Percent of Amount in (4) which Reacts with Steam	0%	50%	0%
(6)	Percent of Amount in (4) which Reacts with Oxygen	0%	0%	50%
(7)	Pressure Increment	11 psi	44 psi	104 psi

TABLE 2.7. ALTERNATIVE ASSIGNMENT OF VALUES TO VERBAL DESCRIPTORS

VERBAL DESCRIPTOR	LIKELIHOOD ALT. 1 (BASE CASE)	ALT. 2	ALT. 3	ALT. 4
Certain or Almost Certain	1.0	1.0	1.0	1.0
Likely	0.9	0.9	0.9	0.9
Indeterminate	0.5	0.5	0.5	0.5
Unlikely	0.1	0.01	0.01	0.1
Remotely Possible	0.001	0.001	0.0001	0.01
Impossible	0	0	0	0

Table 3.1

RESULTS FOR SURRY S_2D

(Small Pipe Break LOCA with Failure of Emergency Core Cooling Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$1x10^{-5}$	
Potential Value with Recent Plant Changes		$1x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.99	.49	.48	.01	.001
Basemat Meltthrough Only	.01	.28	.28	.01	--
Late Overpressurization (Pressure Relief)	.001	.02	.02	--	--
Late Overpressurization (Rupture)	--	.16	.15	.05	--
Late Hydrogen Burn	--	.05	.06	.09	.05
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres. + Hydrogen Burn	--	--	--	.84	--
Early Steam Spike/Depres. + Direct Heating*	--	--	.01	--	.94
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.002	.002	.002	.002
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.01	.03	.03	.05	.05
(b) Induced Small Steam Generator Tube Leak**	--	--	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial break is likely to occur in a cold leg. Additional RCS failures comparable in size to the initial break are not induced prior to vessel breach. Principal mode of vessel breach is high pressure ejection. An in-vessel steam explosion causing failure of the bottom head but not containment failure occurs with moderate frequency in the central and pessimistic walkthroughs. A coolable ex-vessel debris bed generally does not occur in the central and pessimistic walkthroughs; however, vaporization releases are scrubbed by an overlying water layer. Oxidation releases occur in the central and pessimistic walkthroughs.

	BMI-2104 CALCULATION
(1) No Containment Failure. Containment sprays survive throughout the accident.	Yes, without steam explosion.
(2) Basemat Meltthrough. Containment sprays survive throughout the accident.	No, but source term is similar to (1).
(3) Late Overpressurization (Rupture). Containment sprays fail either after vessel breach or after a late hydrogen burn that does not fail containment. Auxiliary building is generally bypassed.	No.
(4) Late Hydrogen Burn. Containment sprays survive after vessel breach but may fail after containment failure. Auxiliary building is generally bypassed.	No.
(5) Early Steam Spike/Depressurization/Hydrogen Burn/Direct Heating. Moderate probability of containment spray failure after containment failure. Auxiliary building is generally bypassed.	Yes, without steam explosion or direct heating.
(6) Large Isolation Failure or Preexisting Leak. Moderate probability that containment sprays do not actuate because containment pressure setpoint is not reached (pessimistic walkthrough). Significant deposition in auxiliary building is likely.	No.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.2

RESULTS FOR SURRY S_3D
(Small Reactor Coolant Pump Seal LOCA with Failure of Emergency Core Cooling Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		9×10^{-5}	
Potential Value with Recent Plant Changes		9×10^{-6}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
No Containment Failure	.98	.63	.58	.01	--
Basemat Meltthrough Only	.01	.16	.16	.002	--
Late Overpressurization (Pressure Relief)	.001	.02	.02	--	--
Late Overpressurization (Rupture)	--	.16	.15	.03	.06
Late Hydrogen Burn	--	.03	.05	.16	--
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres. + Hydrogen Burn	--	--	--	.79	--
Early Steam Spike/Depres. + Direct Heating*	--	--	.04	--	.93
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.002	.002	.002	.002
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.01	.03	.03	.05	.05
(b) Induced Small Steam Generator Tube Leak**	--	.001	.001	.10	.10

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial RCS failure is a cold leg pump seal LOCA. There is comparable frequency of hot leg or cold leg induced LOCAS in the central and optimistic walkthroughs. Worsened seal LOCAs or small SGTRs can occur in the pessimistic walkthrough. The modes of vessel breach are high-pressure ejection, an in-vessel steam explosion that fails the bottom head (but does not threaten containment), or meltthrough accompanying induced hot leg rupture. A permanently coolable debris bed is usually formed in the optimistic walkthrough. A permanently or temporarily coolable debris bed is formed in a sizeable fraction of the central walkthrough, and rarely in the pessimistic walkthrough. An oxidation release usually occurs in the central and pessimistic walkthroughs. A vaporization release either does not occur, or is mitigated by overlying water.

	BMI-2104 CALCULATION
(1) No Containment Failure: Sprays survive throughout the accident. A permanently coolable debris bed generally occurs. (2) Basemat Meltthrough: Sprays usually survive, but a permanently coolable debris bed is not usually formed. (3) Late Overpressurization: Sprays fail after vessel breach or after a late hydrogen burn (which does not threaten containment). A coolable debris bed often forms, but dries out after spray failure. No vaporization release occurs. The auxiliary building is usually bypassed. (4) Late Hydrogen Burn: Sprays survive until containment failure or fail after vessel breach with comparable frequency; sprays seldom survive after containment failure. No vaporization release occurs. The auxiliary building is usually bypassed. (5) Steam Spike/Depressurization/Hydrogen Burn/Direct Heating: Sprays fail shortly after vessel breach or fail later comparable frequency; long term spray survival is rare. Vaporization release rarely occurs. The auxiliary building is usually bypassed. (6) Isolation Failure: No gross containment failure occurs (central), or isolation failure is later followed by direct heating failure. There is a small probability that sprays never operate because the pressure set-point is not reached. The leak passes through the auxiliary building, but later gross failures bypass the auxiliary building.	No. BMI-2104 calculations were made for S_3D, but consequences should be similar to corresponding sequences in S_2D (no steam explosions, induced RCS failures, or direct heating).

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.3

RESULTS FOR SURRY TMLB'
(Station Blackout, Including Loss of Auxiliary Feedwater)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$2x10^{-5}$	
Potential Value with Recent Plant Changes		$3x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
No Containment Failure	.74	.38	.18	.02	--
Basemat Meltthrough Only	.05	.09	.09	.002	--
Late Overpressurization (Pressure Relief)	.10	.03	.02	--	--
Late Overpressurization (Rupture)	.01	.29	.20	.58	.04
Late Hydrogen Burn	.10	.20	.51	.30	.03
Early Steam Spike and/or Vessel Depressurization	--	--	--	.10	--
Early Steam Spike/Depres. + Hydrogen Burn	--	--	--	--	--
Early Steam Spike/Depres. + Direct Heating*	--	--	.005	--	.92
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.002	.002	.002	.002
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.01	.03	.03	.05	.05
(b) Induced Small Steam Generator Tube Leak**	--	.001	.001	.10	.10

III. PRINCIPAL CONTAINMENT PATHWAYS

General: An induced failure of the hot leg or induced seal LOCA usually occurs. In the pessimistic walkthrough, small SGTRs are possible. Vessel failure modes are pressure ejection or meltthrough accompanying induced hot leg ruptures, or in-vessel steam explosions failing the bottom head in the central and pessimistic walkthroughs. Vaporization release almost always occurs and oxidation release almost always occurs in the central and pessimistic walkthroughs.

		BMI-2104 CALCULATION
(1)	No Containment Failure: Sprays recovered late; a permanently coolable debris bed often forms.	No, but consequences similar to below.
(2)	Basemat Meltthrough: Sprays recovered in approximately half of the cases; a permanently coolable debris bed is not formed.	Yes, (no steam explosions, induced RCS failures, or direct heating).
(3)	Late Overpressurization: Sprays either do not recover or are failed by a later hydrogen burn that does not fail containment. A debris bed may be temporarily cooled but dries out after spray failure. The auxiliary building is bypassed.	No.
(4)	Late Hydrogen Burn: Sprays are recovered before the burn, but may fail again after containment rupture. Auxiliary building bypassed.	No.
(5)	Steam Spike/Depressurization/Direct Heating: Sprays never operate prior to containment failure. Moderate probability of later recovery. The auxiliary building is bypassed.	Yes, (no steam explosions, direct heating, or induced RCS failures).
(6)	Isolation Failure: Either no gross containment failure occurs or there is a direct heating failure at vessel breach. Sprays may or may not recover. The isolation failure is usually mitigated by the auxiliary building, but later gross containment failures bypass the auxiliary building.	No.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.4

RESULTS FOR SURRY AB
(Large Pipe Break LOCA with Failure of All AC Power)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$< 1\times10^{-9}$	
Potential Value with Recent Plant Changes		$< 1\times10^{-10}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.69	.20	.20	.01	.01
Basemat Meltthrough Only	.19	.16	.16	.004	.004
Late Overpressurization (Pressure Relief)	.11	.03	.03	--	--
Late Overpressurization (Rupture)	.01	.25	.25	.63	.62
Late Hydrogen Burn	--	.36	.36	.35	.35
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres. + Hydrogen Burn	--	--	--	--	--
Early Steam Spike/Depres. + Direct Heating*	--	--	--	--	--
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.002	.002	.002	.002
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.01	.03	.03	.03	.05
(b) Induced Small Steam Generator Tube Leak**	--	--	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial break occurs with comparable frequency in the hot legs or cold legs. Additional RCS failures are not induced. The principal mode of vessel failure is a low pressure meltthrough; in-vessel steam explosions fail the bottom head (but do not threaten containment) with moderate frequency in central and pessimistic walkthroughs. Permanently coolable debris beds are not generally formed. Vaporization releases occur in all walkthroughs and oxidation releases occur with moderate frequency in central and pessimistic walkthroughs.

	BMI-2104 CALCULATION
(1) No Containment Failure: Containment sprays are recovered late in the accident.	No, but consequences similar to basemat meltthrough.
(2) Basemat Meltthrough: Containment sprays are not generally recovered.	Yes, (no steam explosions).
(3) Late Overpressurization: Containment sprays are inoperative throughout the accident. The auxiliary building is generally bypassed.	No, but consequences should be similar to late hydrogen burn.
(4) Late Hydrogen Burn: Containment sprays are recovered prior to the hydrogen burn, and may fail after containment failure. The auxiliary building is generally bypassed.	Yes, without in-vessel steam explosions.
(5) Steam Spike/Depressurization: The sprays never operate prior to containment failure. There is moderate probability of later spray recovery. The auxiliary building is bypassed.	No.
(6) Isolation Failure: No gross containment failure occurs. There is moderate probability of later recovery of containment sprays.	Yes, for isolation failure with no gross containment failure, and no in-vessel steam explosions.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.5

RESULTS FOR ZION S_2D
(Catastrophic Reactor Coolant Pump Seal LOCA with Failure of Emergency Core Cooling Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$2x10^{-4}$	
Potential Value with Recent Plant Changes		$2x10^{-5}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.99	.54	.54	.07	.03
Basemat Meltthrough Only	.01	.33	.33	.17	.07
Late Overpressurization (Pressure Relief)	--	.01	.01	--	--
Late Overpressurization (Rupture)	--	.10	.10	.58	.25
Late Hydrogen Burn	--	.01	.02	.02	.04
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres. + Hydrogen Burn	--	.001	--	.16	--
Early Steam Spike/Depres. + Direct Heating*	--	--	.003	--	.60
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.004	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	--	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial RCS failure is likely to occur in a cold leg. Induced RCS failures do not occur. The reactor vessel usually fails by pressure ejection, although in-vessel steam explosions fail the bottom head with moderate frequency in the central and pessimistic walkthroughs. Containment sprays are usually activated early (central and optimistic) or at vessel breach (pessimistic). Neither vaporization nor oxidation releases occur in the optimistic walkthrough; both may occur in central and pessimistic walkthroughs.

	BMI-2104 CALCULATION
(1) No Containment Failure: Containment sprays are activated early and survive throughout the accident. A permanently coolable debris bed occurs.	Yes, (no steam explosions or direct heating).
(2) Basemat Meltthrough: Containment sprays generally survive; no permanently coolable debris beds form.	No, but consequences similar to above.
(3) Late Overpressurization: Sprays fail at vessel breach or at the time of a late hydrogen burn that does not threaten containment. A coolable debris bed does not form. The auxiliary building is generally bypassed.	No.
(4) Late Hydrogen Burn: Containment sprays (if activated) fail at vessel breach or at the time of the burn. Debris beds are only temporarily coolable. The auxiliary building is bypassed.	No.
(5) Steam Spike/Hydrogen Burn/Direct Heating: Sprays, if activated, fail shortly after vessel breach. Debris beds not cooled or only temporarily cooled. Auxiliary building bypassed.	No.
(6) Isolation Failure: Sprays may or may not be activated early, may or may not survive. Debris beds may or may not be coolable. The leakage is transmitted through the auxiliary building.	No.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.6

RESULTS FOR ZION S_3D
(Small Reactor Coolant Pump Seal LOCA with Failure of Emergency Core Cooling Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		Not Evaluated	
Potential Value with Recent Plant Changes		Not Evaluated	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
No Containment Failure	.99	.68	.66	.11	.02
Basemat Meltthrough Only	.01	.19	.19	.12	.03
Late Overpressurization (Pressure Relief)	--	.01	.01	--	--
Late Overpressurization (Rupture)	--	.11	.11	.51	.10
Late Hydrogen Burn	--	.006	.01	.03	.07
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres. + Hydrogen Burn	--	.001	--	.22	--
Early Steam Spike/Depres. + Direct Heating*	--	--	.02	--	.77
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.004	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	.001	.001	.10	.10

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial RCS failure is a cold leg seal LOCA. Induced RCS failures in hot or cold legs (central and optimistic) or worsened seal LOCA or small SGTRs (pessimistic) can occur. The reactor vessel usually fails by high-pressure ejection, with moderate probability of in-vessel steam explosions failing the bottom head (central and pessimistic) or meltthrough following induced hot leg rupture. A permanently coolable debris bed forms usually (optimistic), often (central), or rarely (pessimistic).

(1) No Containment Failure: Sprays survive throughout the accident. A permanently coolable debris bed is formed.

(2) Basemat Meltthrough: Sprays survive. A coolable debris bed is formed.

(3) Late Overpressurization: Sprays fail after vessel breach or after a late hydrogen burn that does not fail containment. There is moderate probability that sprays do not operate before vessel breach. The auxiliary building is generally bypassed.

(4) Late Hydrogen Burn: Sprays operate prior to the burn (central) or fail early (pessimistic). The auxiliary building is generally bypassed.

(5) Steam Spike/Hydrogen Burn/Direct Heating: Sprays are usually not actuated before vessel breach, and usually fail at vessel breach. Permanently coolable debris beds are not usually formed. The auxiliary building is generally bypassed.

(6) Isolation Failure: Either no gross containment failure occurs or there is a failure at vessel breach (pessimistic). Sprays survive. There may be a permanently coolable debris bed. Leakage is through the auxiliary building; however, a later gross failure bypasses the auxiliary building.

BMI-2104 CALCULATION

No BMI-2104 calculations were made for S_3D, but consequences should be similar to corresponding sequences in S_2D (no steam explosions, induced RCS failures, or direct heating).

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 5.7

RESULTS FOR ZION TMLB'
(Station Blackout, Including Loss of Auxiliary Feedwater)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		6×10^{-6}	
Potential Value with Recent Plant Changes		6×10^{-6}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
No Containment Failure	.75	.51	.43	.10	.03
Basemat Meltthrough Only	.12	.19	.18	.07	.04
Late Overpressurization (Pressure Relief)	.11	.02	.03	--	--
Late Overpressurization (Rupture)	.01	.22	.23	.51	.37
Late Hydrogen Burn	--	.06	.13	.21	.12
Early Steam Spike and/or Vessel Depressurization	--	--	--	.003	--
Early Steam Spike/Depres. + Hydrogen Burn	--	--	--	--	--
Early Steam Spike/Depres. + Direct Heating*	--	--	.002	--	.43
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.004	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	.001	.001	.10	.10

III. PRINCIPAL CONTAINMENT PATHWAYS

General: An induced rupture of the hot leg or induced seal LOCA usually occurs. Small SGTRs are possible (pessimistic). The principal modes of vessel breach are high-pressure ejection, an in-vessel steam explosion that fails the bottom head, or meltthrough following induced hot leg rupture. Containment sprays are recovered with moderate probability late in the accident, but may fail again after recovery. A permanently coolable debris bed is formed often (optimistic), sometime (central), or rarely (pessimistic). An oxidation release occurs in the central and pessimistic walkthroughs. A vaporization release occurs with moderate probability in the pessimistic walkthrough. Sprays never operate prior to vessel breach.

	BMI-2104 CALCULATION
(1) No Containment Failure: Sprays are recovered and continue to operate. A permanently coolable debris bed is formed.	Yes, (no steam explosions induced RCS failures, or direct heating).
(2) Basemat Meltthrough: Sprays may or may not be recovered. Debris beds, if formed, are only temporarily coolable.	No, but consequences similar to above.
(3) Late Overpressurization: Sprays are not recovered. The auxiliary building is generally bypassed.	No.
(4) Late Hydrogen Burn: Sprays are recovered, but may fail at the time of containment failure. Permanently coolable debris beds are not usually formed. The auxiliary building is usually bypassed.	No.
(5) Steam Spike/Direct Heating: Sprays do not operate. There is moderate probability of a vaporization release. The auxiliary building is bypassed.	No.
(6) Isolation Failure: Either no containment failure occurs or there is a failure at vessel breach (pessimistic). Sprays may or may not operate. Permanently coolable debris beds are rarely formed. Leakage is through the auxiliary building, but later containment failure will bypass the auxiliary building.	No.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.8

RESULTS FOR PEACH BOTTOM AE
(Large LOCA with Failure of Emergency Core Coolant Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		2×10^{-8}	
Potential Value with Recent Plant Modifications		2×10^{-8}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	--	--	--	--	--
Basemat Meltthrough Only	--	--	--	--	--
Late Pressure-Ind. Leak in Drywell (Relief)	1.00	.41	.41	--	--
Late Temperature-Ind. Leak in Drywell (Relief)	--	.05	.05	--	--
Late Temperature-Ind. Failure of Drywell	--	.04	.04	.05	.04
Late Overpressurization of Drywell (Rupture)	--	.40	.40	.44	.39
Late Overpressurization of Wetwell (Rupture)	--	--	--	--	--
Drywell Breach from Melt-Structure Attack	--	--	--	--	--
DW Overpressurization at Vessel Breach (Rupture)**	--	--	--	--	.06
WW Overpressurization at Vessel Breach (Rupture)**	--	--	--	--	--
DW Breach from In-Vessel Steam Explosion (Missile)	--	--	--	--	.005
Pre-Vessel-Breach Pressure-Ind. Leak in DW (Relief)	--	.05	.05	--	--
Pre-Vessel-Breach Overpressurization of DW (Rupture)	--	.05	.05	.50	.50
Pre-Vessel-Breach Overpressurization of WW (Rupture)	--	--	--	--	--
Large Isolation Failure or Preexisting Leak In DW	--	--	--	.005	.004
Large Isolation Failure or Preexisting Leak In WW	--	--	--	.003	.002
Containment Venting (Not Reclosed)	--	--	--	--	--
Pre-Core-Melt Pressure-Induced Leak in DW (Relief)	--	--	--	--	--
Pre-Core-Melt Overpressurization of DW (Rupture)	--	--	--	--	--
Pre-Core-Melt Overpressurization of WW (Rupture)	--	--	--	--	--
Induced LOCA Outside Containment	--	--	--	--	--
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Leak in WW***	.05	.11	.11	.20	.20
(b) Preexisting Small Leak in DW***	.006	.01	.01	.02	.02
(c) Unmitigated Small Leak through MSIVs***	--	.001	.001	.10	.10

III. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

*Case 2 for the central and pessimistic walkthroughs includes in-vessel and ex-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

**Events causing containment pressurization at vessel breach include vessel depressurization, ex-vessel steam spike and/or steam explosion, and ex-vessel direct heating.

***Small containment leaks include those producing leakage greater than containment design, but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Unmitigated small MSIV leaks have little effect on reactor pressure, but condenser vacuum is lost and there is no leakage control.

Table 3.8 (Continued)

RESULTS FOR PEACH BOTTOM AE

III. PRINCIPAL CONTAINMENT PATHWAYS

General: AC power is available. The reactor vessel almost always fails by low pressure meltthrough, with only a slight chance of upper head failure due to an in-vessel steam explosion in the pessimistic case. Low pressure systems including suppression pool cooling are not available so that sustained debris coolability is precluded and core-concrete interactions take place. Drywell leakage or rapid depressurization into the reactor building occurs in all principal pathways. This leads to blowout panel relief, fire damper closure, standby gas treatment system isolation, and actuation of reactor building fire sprays.

	BMI-2104 CALCULATIONS
(1) Late Pressure-Induced Leak in Drywell (Pressure Relief): Pressure-induced leakage from the drywell is sufficient to arrest the pressure buildup due to accumulation of noncondensible gases from core-concrete interactions.	No.
(2) Late Temperature-Induced Leak in Drywell (Pressure Relief): Similar to previous case but leakage is temperature-induced.	No.
(3) Late Temperature-Induced Failure of Drywell: Hot gases from core-concrete interactions heat the steel drywell to the point where structural integrity can no longer be maintained.	No.
(4) Late Overpressurization of Drywell (Rupture): Similar to previous mode but drywell failure is pressure-induced.	No.
(5) Overpressurization of Drywell at Vessel Breach (Rupture): Overpressure is due to an ex-vessel steam explosion which occurs when the bottom head fails and molten core debris falls into water on the drywell floor.	No.
(6) Pre-Vessel-Breach Pressure-Induced Leak in Drywell (Pressure Relief, H2): Sufficient hydrogen is produced due to in-vessel Zr oxidation to pressurize the containment to the point where a drywell leak is induced.	No.
(7) Pre-Vessel-Breach Overpressurization of Drywell (Rupture, H2): Similar to the previous mode but the containment is postulated to rupture instead of leak.	Yes.

Table 3.9

RESULTS FOR PEACH BOTTOM TB
(Station Blackout, Delayed Failure of High Pressure Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$8x10^{-6}$	
Potential Value with Recent Plant Modifications		$1x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.27	.04	.04	.002	.001
Basemat Meltthrough Only	--	--	--	--	--
Late Pressure-Ind. Leak in Drywell (Relief)	.73	.39	.38	--	--
Late Temperature-Ind. Leak in Drywell (Relief)	--	.05	.04	--	--
Late Temperature-Ind. Failure of Drywell	--	.04	.04	.03	.03
Late Overpressurization of Drywell (Rupture)	--	.38	.37	.33	.28
Late Overpressurization of Wetwell (Rupture)	--	--	--	--	--
Drywell Breach from Melt-Structure Attack	--	.008	.004	.13	.06
DW Overpressurization at Vessel Breach (Rupture)**	--	.09	.13	.50	.62
WW Overpressurization at Vessel Breach (Rupture)**	--	--	--	--	--
DW Breach from In-Vessel Steam Explosion (Missile)	--	--	--	--	.005
Pre-Vessel-Breach Pressure-Ind. Leak in DW (Relief)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of DW (Rupture)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of WW (Rupture)	--	--	--	--	--
Large Isolation Failure or Preexisting Leak In DW	--	--	--	.004	.004
Large Isolation Failure or Preexisting Leak In WW	--	--	--	.002	.002
Containment Venting (Not Reclosed)	--	--	--	--	--
Pre-Core-Melt Pressure-Induced Leak in DW (Relief)	--	--		--	--
Pre-Core-Melt Overpressurization of DW (Rupture)	--	--	--	--	--
Pre-Core-Melt Overpressurization of WW (Rupture)	--	--	--	--	--
Induced LOCA Outside Containment	--	--	--	--	--
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Leak in DW***	.05	.11	.11	.20	.20
(b) Preexisting Small Leak in WW***	.006	.01	.01	.02	.02
(c) Unmitigated Small Leak through MSIVs***	1.00	1.00	1.00	1.00	1.00

III. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

*Case 2 for the central and pessimistic walkthroughs includes in-vessel and ex-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

**Events causing containment pressurization at vessel breach include vessel depressurization, ex-vessel steam spike and/or steam explosion, and ex-vessel direct heating.

***Small containment leaks include those producing leakage greater than containment design, but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Unmitigated small MSIV leaks have little effect on reactor pressure, but condenser vacuum is lost and there is no leakage control.

Table 3.9 (Continued)

RESULTS FOR PEACH BOTTOM TB

III. PRINCIPAL CONTAINMENT PATHWAYS

General: AC power is not available; however, there is a 60% chance of restoring AC power following vessel breach. The reactor vessel is at high pressure during the meltdown. ADS is not available after battery depletion. The reactor vessel usually fails by a high pressure meltthrough, but the quantity of debris ejected is limited to that which has accumulated in the vessel bottom head. The principal containment pathways involve drywell failure or leakage to the reactor building. The reactor building blowout panels relieve, the standby gas treatment system is isolated when the fire dampers [illegible] and the reactor building fire sprays are actuated.

		BMI-2104 CALCULATIONS
(1)	No Containment Failure: After vessel breach, late in the accident, AC power is restored, containment heat removal systems are operating, and a coolable debris bed is established and maintained.	No.
(2)	Late Pressure-induced Leak in Drywell (Pressure Relief): Pressure-induced leakage from the drywell is sufficient to arrest the pressure buildup due to accumulation of noncondensible gases (when the core debris is not coolable) and steam (when there is no late containment heat removal).	No.
(3)	Late Temperature-Induced Leak in Drywell (Pressure Relief): Similar to 2 but leakage is temperature-induced.	No.
(4)	Late Temperature-Induced Failure of Drywell: Hot gases from core-concrete interactions heat the steel drywell to the point where structural integrity can no longer be maintained.	No.
(5)	Late Overpressurization of Drywell (Rupture): Similar to containment failure mode 2, but drywell leakage is not induced or is not sufficient to prevent drywell rupture.	No.
(6)	Drywell Breach from Melt-Structure Attack: Molten core debris released from the reactor vessel flows across the drywell floor, contacts the steel shell of the drywell and melts through.	No.
(7)	Drywell Overpressurization at Vessel Breach (Rupture): Overpressurization is due to the combined loads associated with vessel depressurization, any ex-vessel melt coolant interactions which occur when the bottom head is melted through and molten core contacts water on the drywell floor, and any direct heating of the drywell atmosphere by melt ejected from the vessel under high pressure.	No.

Table 3.10

RESULTS FOR PEACH BOTTOM TC
(Transient with Failure to Achieve Subcriticality)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		7×10^{-6}	
Potential Value with Recent Plant Modifications	(6×10^{-7})	6×10^{-7}	(2×10^{-6})

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	--	--	--	--	--
Basemat Meltthrough Only	--	--	--	--	--
Late Pressure-Ind. Leak in Drywell (Relief)	.65	.21	.21	--	--
Late Temperature-Ind. Leak in Drywell (Relief)	--	.02	.02	--	--
Late Temperature-Ind. Failure of Drywell	--	.02	.02	.005	.005
Late Overpressurization of Drywell (Rupture)	--	.21	.21	.03	.05
Late Overpressurization of Wetwell (Rupture)	--	--	--	--	--
Drywell Breach from Melt-Structure Attack	--	.003	.003	.04	.04
DW Overpressurization at Vessel Breach (Rupture)**	--	.03	.04	.06	.06
WW Overpressurization at Vessel Breach (Rupture)**	--	--	--	--	--
DW Breach from In-Vessel Steam Explosion (Missile)	--	--	--	--	.005
Pre-Vessel-Breach Pressure-Ind. Leak in DW (Relief)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of DW (Rupture)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of WW (Rupture)	--	--	--	--	--
Large Isolation Failure or Preexisting Leak In DW	--	--	--	--	--
Large Isolation Failure or Preexisting Leak In WW	--	--	--	--	--
Containment Venting (Not Reclosed)	.25	.25	.25	.04	.04
Pre-Core-Melt Pressure-Induced Leak in DW (Relief)	--	--	--	--	--
Pre-Core-Melt Overpressurization of DW (Rupture)	.09	.25	.25	.72	.72
Pre-Core-Melt Overpressurization of WW (Rupture)	.01	--	--	--	--
Induced LOCA Outside Containment	--	--	--	.10	.10
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Leak in DW***	.05	.11	.11	.20	.20
(b) Preexisting Small Leak in WW***	.006	.01	.01	.02	.02
(c) Unmitigated Small Leak through MSIVs***	.03	.03	.03	.13	.13

III. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

*Case 2 for the central and pessimistic walkthroughs includes in-vessel and ex-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

**Events causing containment pressurization at vessel breach include vessel depressurization, ex-vessel steam spike and/or steam explosion, and ex-vessel direct heating.

***Small containment leaks include those producing leakage greater than containment design, but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Unmitigated small MSIV leaks have little effect on reactor pressure, but condenser vacuum is lost and there is no leakage control.

Table 3.10 (Continued)

RESULTS FOR PEACH BOTTOM TC

III. PRINCIPAL CONTAINMENT PATHWAYS

General: AC power is available, and in only 3% of the cases is offsite power lost. The estimated frequency of the TC sequence depends on the assumptions regarding the likelihood and effects of venting the containment. In the optimistic and central cases, it is assumed that venting will usually prevent core melt. The corresponding sequence frequency is the lowest of the values given in Block I. If the operator fails to vent containment, it is considered likely (optimistic) or moderately possible (central) that high pressure coolant injection will fail due to overheating of the lube oil and ADS will fail due to high containment pressure. In this event, the core melts before containment overpressurization. In the pessimistic case, it is assumed that the venting procedures are usually not effective for ATWS sequences and that either high pressure coolant injection or ADS remain effective until containment failure. Thus, containment overpressurization precedes (and causes) core meltdown. Containment venting or failure causes the reactor building blowout panels to relieve, the fire dampers to close isolating the standby gas treatment system, and the fire sprays to actuate.

	BMI-2104 CALCULATIONS
(1) Late Pressure-Induced Leak in Drywell (Pressure Relief): The operators may vent the wetwell to prevent containment failure, but injections system still fail -- most likely due to adverse pump room environments following venting. The wetwell vent is reclosed following failure of injection systems; however, late in the accident, noncondensibles and steam from core concrete interactions increase the drywell pressure until sufficient leakage develops to arrest the pressure buildup. Alternatively, the operators may fail to vent the wetwell and the scenario proceeds as described under "General".	No.
(2) Late Overpressurization of Drywell (Rupture): Similar chronology to mode 1, but drywell leakage is not induced or is not sufficient to prevent drywell rupture.	No.
(3) Drywell Breach from Melt-Structure Attack: Pathway proceeds as for modes 1 & 2; however, molten core debris released from the reactor vessel flows across the drywell floor, contacts the steel shell of the drywell, and melts it through.	No.
(4) Drywell Overpressurization at Vessel Breach (Rupture): Overpressurization is due to the combined loads associated with vessel depressurization, any ex-vessel melt-coolant interactions which occur when the bottom head is melted through and molten core contacts water on the drywell floor, and any direct heating of the drywell atmosphere by melt ejected from the vessel under high pressure.	No.
(5) Containment Venting (Not Reclosed): The operators vent the wetwell to prevent containment failure, but fail to reclose the vent following failure of injection systems.	No.
(6) Pre-Core-Melt Overpressurization of Drywell (Rupture): The wetwell is not vented and the drywell fails due to steam buildup from the saturated suppression pool. Containment failure leads to failure of the injection systems and thus to core meltdown.	Yes, except BMI-2104 calculation assumes primary system remains pressurized.
(7) Pre-Core-Melt Overpressurization of Wetwell (Rupture): Same chronology as for mode 6, but the containment is assumed to fail in the wetwell. Suppression pool drainage as a result of the wetwell failure is considered unlikely in the optimistic walkthrough, indeterminant in the central walkthrough, and likely in the pessimistic walkthrough.	No.
(8) Induced LOCA Outside Containment (Assume Worse Than Pathway 3): Repressurization of the reactor vessel due to a positive reactivity insertion when cold low pressure water is injected causes failure of a check valve isolating the low pressure systems from the high pressure reactor vessel. A LOCA is induced outside containment.	No.

Table 3.10 (Continued)

RESULTS FOR PEACH BOTTOM TC

III. PRINCIPAL CONTAINMENT PATHWAYS

General: AC power is available, and in only 3% of the cases is offsite power lost. The estimated frequency of the TC sequence depends on the assumptions regarding the likelihood and effects of venting the containment. In the optimistic and central cases, it is assumed that venting will usually prevent core melt. The corresponding sequence frequency is the lowest of the values given in Block I. If the operator fails to vent containment, it is considered likely (optimistic) or moderately possible (central) that high pressure coolant injection will fail due to overheating of the lube oil and ADS will fail due to high containment pressure. In this event, the core melts before containment overpressurization. In the pessimistic case, it is assumed that the venting procedures are usually not effective for ATWS sequences and that either high pressure coolant injection or ADS remain effective until containment failure. Thus, containment overpressurization precedes (and causes) core meltdown. Containment venting or failure causes the reactor building blowout panels to relieve, the fire dampers to close isolating the standby gas treatment system, and the fire sprays to actuate.

		BMI-2104 CALCULATIONS
(1)	Late Pressure-Induced Leak in Drywell (Pressure Relief): The operators may vent the wetwell to prevent containment failure, but injections system still fail -- most likely due to adverse pump room environments following venting. The wetwell vent is reclosed following failure of injection systems; however, late in the accident, noncondensibles and steam from core concrete interactions increase the drywell pressure until sufficient leakage develops to arrest the pressure buildup. Alternatively, the operators may fail to vent the wetwell and the scenario proceeds as described under "General".	No.
(2)	Late Overpressurization of Drywell (Rupture): Similar chronology to mode 1, but drywell leakage is not induced or is not sufficient to prevent drywell rupture.	No.
(3)	Drywell Breach from Melt-Structure Attack: Pathway proceeds as for modes 1 & 2; however, molten core debris released from the reactor vessel flows across the drywell floor, contacts the steel shell of the drywell, and melts it through.	No.
(4)	Drywell Overpressurization at Vessel Breach (Rupture): Overpressurization is due to the combined loads associated with vessel depressurization, any ex-vessel melt-coolant interactions which occur when the bottom head is melted through and molten core contacts water on the drywell floor, and any direct heating of the drywell atmosphere by melt ejected from the vessel under high pressure.	No.
(5)	Containment Venting (Not Reclosed): The operators vent the wetwell to prevent containment failure, but fail to reclose the vent following failure of injection systems.	No.
(6)	Pre-Core-Melt Overpressurization of Drywell (Rupture): The wetwell is not vented and the drywell fails due to steam buildup from the saturated suppression pool. Containment failure leads to failure of the injection systems and thus to core meltdown.	Yes, except BMI-2104 calculation assumes primary system remains pressurized.
(7)	Pre-Core-Melt Overpressurization of Wetwell (Rupture): Same chronology as for mode 6, but the containment is assumed to fail in the wetwell. Suppression pool drainage as a result of the wetwell failure is considered unlikely in the optimistic walkthrough, indeterminant in the central walkthrough, and likely in the pessimistic walkthrough.	No.
(8)	Induced LOCA Outside Containment (Assume Worse Than Pathway 3): Repressurization of the reactor vessel due to a positive reactivity insertion when cold low pressure water is injected causes failure of a check valve isolating the low pressure systems from the high pressure reactor vessel. A LOCA is induced outside containment.	No.

Table 3.11

RESULTS FOR PEACH BOTTOM TQUV
(Transient with Failure of Feedwater and ECC Injection Capability)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		2×10^{-7}	
Potential Value with Recent Plant Modifications		2×10^{-8}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	--	--	--	--	--
Basemat Meltthrough Only	--	--	--	--	--
Late Pressure-Ind. Leak in Drywell (Relief)	1.00	.45	.44	--	--
Late Temperature-Ind. Leak in Drywell (Relief)	--	.05	.05	--	--
Late Temperature-Ind. Failure of Drywell	--	.05	.05	.07	.06
Late Overpressurization of Drywell (Rupture)	--	.44	.43	.67	.56
Late Overpressurization of Wetwell (Rupture)	--	--	--	--	--
Drywell Breach from Melt-Structure Attack	--	.01	.008	.25	.14
DW Overpressurization at Vessel Breach (Rupture)**	--	--	.02	.008	.23
WW Overpressurization at Vessel Breach (Rupture)**	--	--	--	--	--
DW Breach from In-Vessel Steam Explosion (Missile)	--	--	--	--	.005
Pre-Vessel-Breach Pressure-Ind. Leak in DW (Relief)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of DW (Rupture)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of WW (Rupture)	--	--	--	--	--
Large Isolation Failure or Preexisting Leak In DW	--	--	--	.004	.003
Large Isolation Failure or Preexisting Leak In WW	--	--	--	.002	.002
Containment Venting (Not Reclosed)	--	--	--	--	--
Pre-Core-Melt Pressure-Induced Leak in DW (Relief)	--	--	--	--	--
Pre-Core-Melt Overpressurization of DW (Rupture)	--	--	--	--	--
Pre-Core-Melt Overpressurization of WW (Rupture)	--	--	--	--	--
Induced LOCA Outside Containment	--	--	--	--	--
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Leak in DW***	.05	.11	.11	.20	.20
(b) Preexisting Small Leak in WW***	.006	.01	.01	.02	.02
(c) Unmitigated Small Leak through MSIVs***	.29	.29	.29	.36	.36

III. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

*Case 2 for the central and pessimistic walkthroughs includes in-vessel and ex-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

**Events causing containment pressurization at vessel breach include vessel depressurization, ex-vessel steam spike and/or steam explosion, and ex-vessel direct heating.

***Small containment leaks include those producing leakage greater than containment design, but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Unmitigated small MSIV leaks have little effect on reactor pressure, but condenser vacuum is lost and there is no leakage control.

Table 3.11 (Continued)

RESULTS FOR PEACH BOTTOM TQUV

III. PRINCIPAL CONTAINMENT PATHWAYS

General: AC power is available although in 29% of the cases the initiating event is loss of offsite power. The reactor vessel usually fails by a high pressure meltthrough, but the quantity of debris ejected is limited to the amount which has accumulated in the vessel bottom head. This is a small fraction of the core mass in the optimistic and central walkthroughs. There is also a small probability of failure of the upper head due to an in-vessel steam explosion in the pessimistic walkthrough. Low pressure systems including suppression pool cooling are not available so that sustained debris coolability is precluded and core concrete interactions take place. Drywell leakage or rapid depressurization into the reactor building occurs in all principal pathways. This leads to blowout panel relief, fire damper closure, standby gas treatment system isolation, and actuation of reactor building fire sprays.

	BMI-2104 CALCULATIONS
(1) Late Pressure-Induced Leak in Drywell (Pressure Relief): Pressure-induced leakage in the drywell is sufficient to arrest the pressure buildup due to accumulation of noncondensible gases from core-concrete interactions.	No.
(2) Late Temperature-Induced Leak in Drywell (Pressure Relief): Similar to mode 1 but leakage is temperature-induced.	No.
(3) Late Temperature-Induced Failure of Drywell: Hot gasses from core concrete interactions heat the steel drywell to the point where structural integrity can no longer be maintained.	No.
(4) Late Overpressurization of Drywell (Rupture): Similar to containment failure mode 1, but drywell leakage is not induced or is not sufficient to prevent drywell rupture.	No.
(5) Drywell Breach from Melt-Structure Attack: Molten core debris released from the reactor vessel flows across the drywell floor, contacts the steel shell of the drywell, and melts through it.	No.
(6) Pressure-Induced Failure of Drywell at Vessel Breach (Rupture): Overpressurization is due to the combined loads associated with vessel depressurization, any ex-vessel melt coolant interactions which occur when the bottom head is melted through and molten core contacts water on the drywell floor, and any direct heating of the drywell atmosphere by melt ejected from the vessel under high pressure.	No.

Table 3.12

RESULTS FOR PEACH BOTTOM TW
(Transient with Failure of Containment Heat Removal)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		9×10^{-6}	
Potential Value with Recent Plant Modifications	(2×10^{-7})	5×10^{-7}	(9×10^{-6})

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
No Containment Failure	--	--	--	--	--
Basemat Meltthrough Only	--	--	--	--	--
Late Pressure-Ind. Leak in Drywell (Relief)	.10	.13	.13	--	--
Late Temperature-Ind. Leak in Drywell (Relief)	--	.02	.02	--	--
Late Temperature-Ind. Failure of Drywell	--	.02	.02	.04	.03
Late Overpressurization of Drywell (Rupture)	--	.13	.13	.33	.33
Late Overpressurization of Wetwell (Rupture)	--	--	--	--	--
Drywell Breach from Melt-Structure Attack	--	.003	.003	.12	.12
DW Overpressurization at Vessel Breach (Rupture)**	--	--	--	--	.03
WW Overpressurization at Vessel Breach (Rupture)**	--	--	--	--	--
DW Breach from In-Vessel Steam Explosion (Missile)	--	--	--	--	.005
Pre-Vessel-Breach Pressure-Ind. Leak in DW (Relief)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of DW (Rupture)	--	--	--	--	--
Pre-Vessel-Breach Overpressurization of WW (Rupture)	--	--	--	--	--
Large Isolation Failure or Preexisting Leak In DW	--	--	--	.003	.003
Large Isolation Failure or Preexisting Leak In WW	--	--	--	.002	.002
Containment Venting (Not Reclosed)	--	.25	.25	.49	.46
Pre-Core-Melt Pressure-Induced Leak in DW (Relief)	.90	.23	.23	--	--
Pre-Core-Melt Overpressurization of DW (Rupture)	--	.22	.22	.02	.02
Pre-Core-Melt Overpressurization of WW (Rupture)	--	--	--	--	--
Induced LOCA Outside Containment	--	--	--	--	--
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Leak in DW***	.05	.11	.11	.20	.20
(b) Preexisting Small Leak in WW***	.006	.01	.01	.02	.02
(c) Unmitigated Small Leak through MSIVs***	.1	.1	.1	.19	.19

III. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

*Case 2 for the central and pessimistic walkthroughs includes in-vessel and ex-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

**Events causing containment pressurization at vessel breach include vessel depressurization, ex-vessel steam spike and/or steam explosion, and ex-vessel direct heating.

***Small containment leaks include those producing leakage greater than containment design, but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Unmitigated small MSIV leaks have little effect on reactor pressure, but condenser vacuum is lost and there is no leakage control.

Table 3.12 (Continued)

RESULTS FOR PEACH BOTTOM TW

III. PRINCIPAL CONTAINMENT PATHWAYS

General: The estimated frequency of the TW sequence strongly depends upon the assumptions regarding the likelihood and effects of venting the containment. In the optimistic case, it is assumed that venting will prevent core melt. The corresponding sequence frequency is the lowest of the values given in Block I. If the operator fails to vent containment, it is considered likely that he will continue to provide core water makeup through the various options available despite the increasing pressure in containment. Thus, unvented core melts most of the time result in Pre-Core-Melt Induced leak of the drywell which relieves the pressure and prevents later rupture failure of the containment. The rest result in late overpressurization leaks. The pessimistic and central cases allow for melts due to pump cavitation or pump overheating even though the containment is vented. The corresponding sequence frequencies are in the mid range and upper end, respectively of the values given in Block I. The probability that the vents will be reclosed is assumed to be .5 since it is not clear whether closing the vent will be beneficial and operating procedures do not give guidance beyond core melt. Venting without reclosing prevents suppression pool bypass. Unvented melts result in leakage or rupture of the drywell with blowout panel relief, fire damper closure, standby gas treatment system isolation, and fire spray actuation before core melt. Reclosed vents result in these effects well after vessel breach.

	BMI-2104 CALCULATIONS
(1) Late Pressure-induced Leak in Drywell (Pressure Relief): Pressure induced leakage from the drywell is sufficient to arrest the pressure buildup due to accumulation of noncondensible gasses from core concrete interactions.	No.
(2) Late Temperature-Induced Leak in Drywell (Pressure Relief): Similar to the previous pathway but leakage is temperature induced.	No.
(3) Late Temperature-Induced Drywell Failure: Hot gasses from core concrete interactions heat the drywell to the point where structural integrity can no longer be maintained.	No.
(4) Late Overpressurization of Drywell (Rupture): Similar to previous mode but failure is pressure induced.	No.
(5) Drywell Breach from Melt-Structure Attack: After vessel breach some of the core debris accumulated against the steel wall of the containment directly attacks and melts through the wall.	No.
(6) Overpressurization of Drywell at Vessel Breach (Rupture): Overpressure is due to an ex-vessel steam explosion which occurs when the bottom head fails and molten debris falls into water on the drywell floor.	No.
(7) Containment Venting (Not Reclosed): The wetwell is vented to prevent containment failure, but the EEC pumps fail and the core melts with the vents still open.	No.
(8) Pre-Core-Melt Pressure-Induced Leak in Drywell (Pressure Relief): The lack of suppression pool cooling causes the pressure in the containment to induce a leak in the drywell sufficient to arrest the pressure buildup.	No.
(9) Pre-Core-Melt Overpressurization of Drywell (Rupture): Similar to previous pathway but the containment is postulated to rupture instead of leak.	Yes.

Table 3.13

RESULTS FOR GRAND GULF TC
(Transient with Failure to Achieve Subcriticality)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$5x10^{-6}$	
Potential Value with Recent Plant Modifications		$5x10^{-7}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	--	--	--	--	--
Basemat Meltthrough Only	--	--	--	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	--	--	--	--	--
Late Hydrogen Burn	--	--	--	--	--
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres./H_2 Burn	--	--	--	--	--
Early Steam Spike/Depres./H_2 Burn/Direct Heating*	--	--	--	--	--
Early Hydrogen Combustion (Before Vessel Breach)	--	--	--	--	--
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	--
Large Isolation Failure or Preexisting Leak Only	--	--	--	--	--
Pre-Core-Melt Overpressurization (Pressure Relief)	--	--	--	--	--
Pre-Core-Melt Overpress. (Rupture) or Venting	1.00	1.00	1.00	1.00	1.00
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by Pre-Existing Cont. Leaks:					
(a) Small Leak**	.02	.03	.03	.06	.06
(b) Large Leak or Isolation Failure**	--	.003	.003	.004	.004

III. DRYWELL-TO-WETWELL LEAKAGE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
Design Leakage Only or Small Cracks	1.00	.99	.98	.75	.67
Late Ind. Penetration Failure (e.g., Vacuum Breaker)	.001	.005	.005	.22	.20
Late Ind. Drywell Structural Failure	--	.001	.001	.03	.02
Early Ind. Penetration Failure	--	.002	.002	.004	.01
Early Ind. Drywell Structural Failure	--	--	.008	--	.09
	1.00	1.00	1.00	1.00	1.00

IV. PRINCIPAL CONTAINMENT PATHWAYS

General: Igniters are available, but no burns occur because the containment is steam inert. Containment always fails before core melt. The reactor vessel usually fails by a low pressure meltthrough, with a small probability of pressurized ejection in the central and pessimistic walkthroughs. There is also a small probability of either failure of the bottom head or containment failure due to missile breach following in-vessel steam explosions in the pessimistic walkthrough. Containment sprays may be operating early, but can not prevent containment overpressure. Sprays fail at containment failure.

	BMI-2104 CALCULATION
(1) Pre-Core-Melt Overpressurization: Small probability of induced drywell wall leakage due to steam explosions, direct heating, or late pedestal attack. Small probability of oxidation release, but no vaporization releases because drywell is flooded.	Yes, except assumed RPV would remain pressurized. Also, steam explosions, direct heating, and drywell flooding were not considered.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Large leaks preclude gradual overpressurization but may not prevent early containment failure from vessel depressurization/steam spikes/hydrogen burns/direct heating.

Table 3.14

RESULTS FOR GRAND GULF TPI
(Transient with a Stuck Open Relief Valve and Failure of Containment Heat Removal)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		5×10^{-7}	
Potential Value with Recent Plant Modifications		2×10^{-8}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	--	--	--	--	--
Basemat Failure Only	--	--	--	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	--	--	--	--	--
Late Hydrogen Burn	--	--	--	--	--
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres./H_2 Burn	--	--	--	--	--
Early Steam Spike/Depres./H_2 Burn/Direct Heating	--	--	--	--	--
Early Hydrogen Combustion (Before Vessel Breach)	--	--	--	--	--
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	--
Large Isolation Failure or Pre-existing Leak Only	--	.002	.002	.004	.004
Pre-Core-Melt Overpressurization (Pressure Relief)	--	--	--	--	--
Pre-Core-Melt Overpress. (Rupture) or Venting	1.00	1.00	1.00	1.00	1.00
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by Pre-existing Cont. Leak:					
(a) Small Leak**	.02	.03	.03	.06	.06
(b) Large Leak or Isolation Failure**	--	.003	.003	.004	.004

III. DRYWELL-TO-WETWELL LEAKAGE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
Design Leakage Only or Small Cracks	1.00	.99	.98	.74	.67
Late Ind. Penetration Failure (e.g., Vacuum Breaker)	.001	.005	.005	.23	.20
Late Ind. Drywell Structural Failure	--	.001	.001	.03	.02
Early Ind. Penetration Failure	--	.002	.002	.004	.01
Early Ind. Drywell Structural Failure	--	--	.008	--	.09
	1.00	1.00	1.00	1.00	1.00

IV. PRINCIPAL CONTAINMENT PATHWAYS

General: Igniters are usually available, but no burns occur because the containment is steam inert. Containment usually fails before core melt, but there is a small probability of a large preexisting leak. The reactor vessel usually fails by a low pressure meltthrough, with a small probability of failure of the upper or bottom heads due to in-vessel steam explosions in the pessimistic walkthrough. There is also a very small probability of containment failure due to missile breach resulting from in-vessel steam explosions in the pessimistic walkthrough. Containment sprays may be operating early, but can not prevent containment overpressure because the heat exchangers are not operating. Sprays fail at containment failure if AC is available early.

	BMI-2104 CALCULATION
(1) Pre-Core-Melt Overpressurization: Small probability of induced drywell wall leakage due to steam explosions, direct heating, or late pedestal attack. Small probability of oxidation release, but no vaporization releases because drywell is flooded.	Yes, except steam explosions and drywell flooding were not considered.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Large leaks preclude gradual overpressurization but may not prevent early containment failure from vessel depressurization/steam spikes/hydrogen burns/direct heating.

Table 3.15

RESULTS FOR GRAND GULF TQUV
(Transient with Failure of Feedwater and Emergency Core Coolant Injection Capability)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$2x10^{-6}$	
Potential Value with Recent Plant Modifications		$1x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
No Containment Failure	.44	.05	.05	--	--
Basemat Meltthrough Only	--	--	--	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	.33	.21	.21	.001	--
Late Hydrogen Burn	.23	.18	.18	.18	.17
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	.004
Early Steam Spike/Depres./H_2 Burn	--	.26	.26	.32	.29
Early Steam Spike/Depres./H_2 Burn/Direct Heating*	--	--	.007	--	.03
Early Hydrogen Combustion (Before Vessel Breach)	--	.30	.30	.50	.50
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	--
Large Isolation Failure or Preexisting Leak	--	.001	.001	--	--
Pre-Core-Melt Overpressurization (Pressure Relief)	--	--	--	--	--
Pre-Core-Melt Overpress. (Rupture) or Venting	--	--	--	--	--
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by Pre-Existing Cont.Leak:					
(a) Small Leak**	.02	.03	.03	.06	.06
(b) Large Leak or Isolation Failure**	--	.003	.003	.004	.004

III. DRYWELL-TO-WETWELL LEAKAGE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
Design Leakage Only or Small Cracks & Leaks	.99	.97	.96	.49	.44
Late Ind. Penetration Failure (e.g., Vac. Breaker)	.006	.02	.02	.33	.30
Late Induced DW Structural Failure	--	.001	.001	.03	.02
Early Induced Penetration Failure	--	.007	.007	.14	.13
Early Induced DW Structural Failure	--	--	.009	.02	.11
	1.00	1.00	1.00	1.00	1.00

IV. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Large leaks preclude gradual overpressurization but may not prevent early containment failure from vessel depressurization/steam spikes/hydrogen burns/direct heating.

Table 3.15 (Continued)

RESULTS FOR GRAND GULF TQUV

IV. PRINCIPAL CONTAINMENT PATHWAYS

General: AC failure prevents igniter operation in 73% of the cases. The reactor vessel usually fails by a low pressure meltthrough, with a small probability of pressurized ejection in the central and pessimistic walkthroughs. There is also a small probability of failure of the upper or bottom heads due to in-vessel steam explosions in the pessimistic walkthrough. Containment sprays will not be operating early, but may be initiated by burns in cases where ECC has failed due to operator failure to ADS or may be initiated after AC power recovery.

	BMI-2104 CALCULATION
(1) No Containment Failure: Containment heat removal systems are operating late. The debris bed is coolable.	No, but releases would be minimal.
(2) Late Overpressurization: Overpressure by either noncondensibles (when debris bed not coolable) or steam (when there is no late containment heat removal). A very small percentage of oxidation releases occur, but there is no drywell wall leakage, so the releases are scrubbed in the pool. Most vaporization releases are scrubbed either by water that has flooded the drywell or by the suppression pool. A very small percentage are not scrubbed late due to late induced drywell leakage from burns or pedestal failure.	Yes, except no drywell wall leakage or drywell flooding.
(3) Late Hydrogen Burn: Small probability of induced drywell penetration failure, but any subsequent vaporization releases will be scrubbed before entering the DW due to DW flooding. Small probability of oxidation or vaporization release due to early drywell wall leakage.	No. Calculations assumed igniterswere always available.
(4) Early Steam Spike/Depres./Hydrogen Burn Small probability of induced drywell penetration failure, but vaporization releases will be scrubbed before entering the DW due to DW flooding. Small probability of oxidation release due to early drywell leakage.	No. Calculations did not consider drywell flooding.
(5) Early Steam Spike/Depres./Direct Heating: Small probability that there will be insufficient time for vents to clear, resulting in drywell structural failure. Most failures are due to burning the hydrogen generated during direct heating. Small probability of unscrubbed oxidation release due to drywell wall leakage. Drywell will be flooded after containment failure, providing scrubbing for vaporization releases.	No. Calculations did not consider direct heatir or drywell flooding.
(6) Early Hydrogen Combustion: Small probability of induced drywell structural failure or penetration failure. Drywell flooding prevents vaporization releases, but small probability of oxidation release.	No. Calculations assumed igniters were always available.

Table 3.16

RESULTS FOR GRAND GULF TB
(Station Blackout, Delayed Failure of High Presure Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		2×10^{-6}	
Potential Value with Recent Plant Modifications		4×10^{-7}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.61	.06	.06	--	--
Basemat Failure Only	--	--	--	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	.39	.03	.03	--	--
Late Hydrogen Burn	--	.25	.25	.24	.24
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	.006
Early Steam Spike/Depres. and H_2 Burn	--	.25	.25	.24	.23
Early Steam Spike/Depres/H_2 Burn/Direct Heating	--	--	--	--	.009
Early Hydrogen Combustion (Before Vessel Breach)	--	.41	.41	.51	.51
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	--
Large Isolation Failure or Pre-existing Leak Only	--	--	--	--	--
Pre-Core-Melt Overpressurization (Pressure Relief)	--	--	--	--	--
Pre-Core-Melt Overpress. (Rupture) or Venting	--	--	--	--	--
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by Pre-existing Cont. Leak:					
(a) Small Leak**	.02	.03	.03	.06	.06
(b) Large Leak or Isolation Failure**	--	.003	.003	.004	.004

III. DRYWELL-TO-WETWELL LEAKAGE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
Design Leakage Only or Small Cracks	1.00	.96	.96	.50	.44
Late Ind. Penetration Failure (e.g., Vacuum Breaker)	.004	.03	.03	.40	.36
Late Ind. Drywell Structural Failure	--	.001	.001	.03	.02
Early Ind. Penetration Failure	--	.002	.002	.07	.07
Early Ind. Drywell Structural Failure	--	--	.008	.005	.09
	1.00	1.00	1.00	1.00	1.00

IV. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Large leaks preclude gradual overpressurization but may not prevent early containment failure from vessel depressurization/steam spikes/hydrogen burns/direct heating.

Table 3.16 (Continued)

RESULTS FOR GRAND GULF TB

IV. PRINCIPAL CONTAINMENT PATHWAYS

General: Igniters are inoperable early due to loss of all AC. The reactor vessel usually fails by a low pressure meltthrough, with a small probability of pressurized ejection in the central and pessimistic walkthroughs. There is also a small probability of failure of the upper or bottom heads due to in-vessel steam explosions in the pessimistic walkthrough. Containment sprays will not be operating early, due to AC power loss, but may be initiated late, after power recovery.

	BMI-2104 CALCULATION
(1) No Containment Failure: Containment heat removal systems are operating late (after AC recovery). The debris bed is coolable.	No. BMI-2104 did not calculate this sequence.
(2) Late Overpressurization: Overpressure by either noncondensibles (when debris bed not coolable) or steam (when there is no late containment heat removal). A very small percentage of oxidation releases occur, but there is no drywell wall leakage, so the releases are scrubbed in the pool. Most vaporization releases are scrubbed either by water that has flooded the drywell or by the suppression pool. A very small percentage are not scrubbed late due to late induced drywell leakage from burns or pedestal failure.	
(3) Late Hydrogen Burn: Small probability of induced drywell penetration failure, but any subsequent vaporization releases will be scrubbed before entering the DW due to DW flooding. Small probability of oxidation or vaporization release due to early drywell wall leakage.	
(4) Early Steam Spike/Depres./Hydrogen Burn: Small probability of induced drywell penetration failure, but vaporization releases will be scrubbed before entering the DW due to DW flooding. Small probability of oxidation release due to early drywell leakage.	
(5) Early Steam Spike/Depres./Direct Heating: Small probability that there will be insufficient time for vents to clear, resulting in drywell structural failure. Most failures due to burning the hydrogen generated during direct heating. Small probability of unscrubbed oxidation release due to drywell wall leakage. Drywell will be flooded after containment failure, providing scrubbing for vaporization releases.	
(6) Early Local Hydrogen Detonation or Deflagration: Small probability of induced drywell structural failure or penetration failure. Drywell flooding prevents vaporization releases, but small probability of oxidation release.	

Table 3.17

RESULTS FOR GRAND GULF S_2E
(Small Pipe Break LOCA with Failure of Emergency Core Coolant Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		6×10^{-8}	
Potential Value with Recent Plant Modifications		6×10^{-9}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.30	.01	.01	--	--
Basemat Failure Only	--	--	--	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	.70	.25	.25	.002	.002
Late Hydrogen Burn	--	.14	.14	.12	.12
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	.003
Early Steam Spike/Depres./H_2 Burn	--	.39	.39	.58	.56
Early Steam Spike/Depres./H_2 Burn/Direct Heating	--	--	--	--	.02
Early Hydrogen Combustion (Before Vessel Breach)	--	.20	.20	.29	.29
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.002
Large Isolation Failure or Pre-existing Leak Only	--	.001	.001	.001	--
Pre-Core-Melt Overpressurization (Pressure Relief)	--	--	--	--	--
Pre-Core-Melt Overpress. (Rupture) or Venting	--	--	--	--	--
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by Pre-existing Cont. Leaks:					
(a) Small Leak**	.02	.03	.03	.06	.06
(b) Large Leak or Isolation Failure**	--	.003	.003	.004	.004

III. DRYWELL-TO-WETWELL LEAKAGE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
Design Leakage Only or Small Cracks	.99	.93	.93	.54	.46
Late Ind. Penetration Failure (e.g., Vacuum Breaker)	.007	.03	.03	.30	.29
Late Ind. Drywell Structural Failure	--	.001	.001	.03	.03
Early Ind. Penetration Failure	--	.03	.03	.13	.12
Early Ind. Drywell Structural Failure	--	--	.008	.002	.09
	1.00	1.00	1.00	1.00	1.00

IV. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Large leaks preclude gradual overpressurization but may not prevent early containment failure from vessel depressurization/steam spikes/hydrogen burns/direct heating.

Table 3.17 (Continued)

RESULTS FOR GRAND GULF S_2E

IV. PRINCIPAL CONTAINMENT PATHWAYS

General: AC failure prevents igniter operation in 50% of the cases. The reactor vessel usually fails by a low pressure meltthrough, with a very small probability of failure of the upper or bottom heads or containment failure by missile breach (if upper pool is dumped) due to in-vessel steam explosions in the pessimistic walkthrough. Containment sprays will not be operating early, but may be initiated after AC recovery.

		BMI-2104 CALCULATION
(1)	No Containment Failure: Containment heat removal systems are operating late. The debris bed is coolable.	No, but releases will be minimal.
(2)	Late Overpressurization: Overpressure by either noncondensibles (when debris bed not coolable) or steam (when there is no late containment heat removal). Practically all oxidation releases releases are scrubbed. Most vaporization releases are scrubbed either by water that has flooded the drywell or by the suppression pool. A very small percentage are not scrubbed late due to late induced drywell leakage from burns or pedestal failure.	No. Calculation showed early containment failure due to hydrogen burn.
(3)	Late Hydrogen Burn: Small probability of induced drywell penetration failure, but any subsequent vaporization releases will be scrubbed before entering the DW due to DW flooding. Small probability of oxidation or vaporization release due to early drywell wall leakage.	No. Calculations assumed igniters were always operating.
(4)	Early Steam Spike/Depres./Hydrogen Burn: Small probability of induced drywell penetration failure, but vaporization releases will be scrubbed before entering the DW due to DW flooding. Small probability of oxidation release due to early drywell leakage.	No. Calculations did not consider drywell flooding.
(5)	Early Steam Spike/Depres./Direct Heating: Small probability that there will be insufficient time for vents to clear, resulting in drywell structural failure. Most failures due to burning the hydrogen generated during direct heating. Small probability of unscrubbed oxidation release due to drywell wall leakage. Drywell will be flooded after containment failure, providing scrubbing for vaporization releases.	No. Calculations did not consider direct heating or drywell flooding.
(6)	Early Local Hydrogen Detonation or Deflagration: Small probability of induced drywell structural failure or penetration failure. Drywell flooding prevents vaporization releases, but small probability of oxidation release.	Yes, but no drywell wall leakage or drywell flooding.

Table 3.18

RESULTS FOR SEQUOYAH S_2H
(Small Pipe Break LOCA with Failure of Emergency Core Cooling Recirculation)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$2x10^{-5}$	
Potential Value with Recent Plant Modifications		Not Evaluated	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	1.0	.40	.40	--	--
Basemat Meltthrough Only	.001	.004	.004	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	.001	.09	.09	--	--
Late Hydrogen Burn	--	.002	.002	--	--
Early Steam Spike and/or Vessel Depressurization	--	--	--	.003	.003
Early Steam Spike/Depres./H_2 Burn	--	.50	.50	.99	.98
Early Steam Spike/Depres./H_2Burn/Direct Heating*	--	--	--	--	--
Hydrogen Combustion Before Vessel Breach	--	--	--	--	--
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.006	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	--	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial RCS failure occurs in hot or cold leg with comparable frequency. Induced RCS failures do not occur. The vessel usually fails by pressure ejection into a water filled cavity. Ice Condenser bypass paths never (O), sometimes (C), and usually (P) develop before core degradation. The auxiliary building is nearly always bypassed after containment failure. Oxidation and vaporization releases rarely occur.

	BMI-2104 CALCULATION
(1) No Containment Failure: Containment sprays survive and a coolable debris bed is formed. (2) Late Overpressurization: Either the sprays fail after vessel breach and containment failure occurs due to steam buildup or the debris is not coolable and containment failure occurs due to noncondensible gas accumulation. (3) Early Steam Spike/Hydrogen Burn: Ice melting or bypass paths develop prior to or during the time of vessel breach. Containment fails at the time of vessel breach due to hydrogen combustion and the sprays usually (C) or seldom (P) survive.	No BMI-2104 calculations were made for S_2H. Releases from the RCS should be similar to those for S_2HF. Reductions in source term due to containment should be similar to those achieved in TML which was calculated.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.19

RESULTS FOR SEQUOYAH S_2HF
(Small Pipe Break LOCA with Failure of Emergency Core Cooling Recirculation and Containment Spray Recirculation

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$5x10^{-6}$	
Potential Value with Recent Plant Modifications		$5x10^{-7}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	--	--	--	--	--
Basemat Meltthrough Only	--	--	--	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	1.00	.47	.47	--	--
Late Hydrogen Burn	--	.02	.02	--	--
Early Steam Spike and/or Vessel Depressurization	--	--	--	.003	.003
Early Steam Spike/Depres./H_2 Burn	--	.51	.51	.99	.98
Early Steam Spike/Depres./H_2 Burn/Direct Heating.	--	--	--	--	--
Hydrogen Combustion Before Vessel Breach	--	--	--	--	--
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.004	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	--	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial RCS failure occurs in hot or cold leg with comparable frequency. Induced RCS failures do not occur. The vessel usually fails by pressure ejection into a water filled cavity. Ice Condenser bypass paths never (O), sometimes (C), and usually (P) develop before core degradation. Sprays are defined to be failed in initiating sequence, no credit is given for recovery. A coolable debris bed usually forms. The auxiliary building is usally bypassed. Oxidation and vaporization releases rarely occur.

	BMI-2104 CALCULATION
(1) Late Overpressurization: No hydrogen burns occur which challenge containment. A coolable debris bed forms but the sprays are inoperable and the containment fails due to steam buildup.	No, but there was a calculation for late containment failure due to a hydrogen burn which may be applicable here.
(2) Early Steam Spike/Hydrogen Burn: Ice melting or bypass paths develop prior to or during the time of vessel breach. A large hydrogen burn occurs at vessel breach failing containment. The sprays have been inoperable since the beginning of the accident. The debris is sometimes (C) or usually (P) not coolable.	No

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.20

RESULTS FOR SEQUOYAH S_3D
(Small Reactor Coolant Pump Seal LOCA with Failure of Emergency Core Cooling Injection)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		4×10^{-5}	
Potential Value with Recent Plant Modifications		2×10^{-7}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL CASE 1*	CENTRAL CASE 2*	PESS. CASE 1*	PESS. CASE 2*
No Containment Failure	.90	.66	.66	--	--
Basemat Meltthrough	.10	.02	.02	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	.001	.31	.31	.001	.001
Late Hydrogen Burn	--	--	--	.003	.003
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres./H_2Burn	--	.009	.009	.99	.98
Early Steam Spike/Depres./H_2Burn/Direct Heating*	--	--	--	--	--
Early Hydrogen Burn	--	--	--	--	--
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.004	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	.001	.001	.10	.10

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial RCS failure is a pump seal LOCA in the cold leg. Induced RCS failure may occur in hot (C) and cold (O&C) legs, or a worsened seal LOCA or small SGTR (P) may occur. Vessel failure usually occurs with the RCS at high or intermediate pressure. The ice condenser is very seldom bypassed or void of ice until late. A coolable debris bed forms usually (O), sometimes (C), or very seldom (P). The auxiliary building is nearly always bypassed after containment failure. Oxidation and vaporization releases nearly never occur.

	BMI-2104 CALCULATION
(1) No Containment Failure: Sprays survive throughout the accident and maintain the debris coolable.	No BMI-2104 calculations were made for S3D, but source terms for TML may apply.
(2) Basemat Meltthrough: Sprays survive but the debris is not coolable, concrete attack occurs and produces meltthrough.	
(3) Late Overpressurization: Sprays usally survive, the debris is not coolable, concrete attack proceeds, and containment fails due to noncondensible gas accumulation.	
(4) Early Steam Spike/Hydrogen Burn: Sprays fail at the time of vessel breach, the debris is not coolable.	

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.21

RESULTS FOR SEQUOYAH TML
(Transient with Failure of all Feedwater and No Feed and Bleed Capability)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		$3x10^{-6}$	
Potential Value with Recent Plant Modifications		$3x10^{-7}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.90	.66	.66	--	--
Basemat Meltthrough Only	.10	.02	.02	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	.001	.31	.31	.001	.001
Late Hydrogen Burn	--	--	--	.003	.003
Early Steam Spike and/or Vessel Depressurization	--	--	--	--	--
Early Steam Spike/Depres./H_2Burn	--	.009	.009	.99	.98
Early Steam Spike/Depres./H_2Burn/Direct Heating*	--	--	--	--	--
Hydrogen Combustion Before Vessel Breach	--	--	--	--	--
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.004	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	.001	.001	.1	.10

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial loss of coolant is via the relief valves for the RCS. Induced RCS failure in hot (C) and cold (O&C) legs or small SGTR (P) may occur. Vessel failure usually occurs with the RCS at high or intermediate pressure. The ice condenser is very seldom bypassed or void of ice until late. A coolable debris bed forms usually (O), sometimes (C), or very seldom (P). The auxiliary building is nearly always bypassed after containment failure. Oxidation and vaporization releases nearly never occur.

		BMI-2104 CALCULATION
(1)	No Containment Failure: Sprays survive throughout the accident and maintain the debris coolable.	Yes, BMI-2104 TML delta calculation is misnamed. Containment never failed.
(2)	Basemat Meltthrough: Sprays survive but the debris is not coolable, concrete attack occurs and produces meltthrough.	No, but calculation for any basemat meltthrough scenario should suffice.
(3)	Late Overpressurization: Sprays usally survive, the debris is not coolable, concrete attack proceeds, and containment fails due to noncondensible gas accumulation.	No, BMI-2104 TML delta calculation is misnamed. Containment never failed.
(4)	Early Steam Spike/Hydrogen Burn: Sprays fail at the time of vessel breach, the debris is not coolable.	Yes

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.22

RESULTS FOR SEQUOYAH TMLB'
(Station Blackout, Including Loss of Auxiliary Feedwater)

I. SEQUENCE FREQUENCY

	OPT.	CENTRAL	PESS.
ASEP Baseline Value		1×10^{-6}	
Potential Value with Recent Plant Modifications		2×10^{-7}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL		PESS.	
		CASE 1*	CASE 2*	CASE 1*	CASE 2*
No Containment Failure	.67	.007	.006	--	--
Basemat Meltthrough Only	.08	.006	.006	--	--
Late Overpressurization (Pressure Relief)	--	--	--	--	--
Late Overpressurization (Rupture)	.25	.08	.08	.002	.002
Late Hydrogen Burn	--	.006	.005	.06	.06
Early Steam Spike and/or Vessel Depressurization	--	--	--	.01	--
Early Steam Spike/Depres./H_2Burn	--	.90	.84	.82	.64
Early Steam Spike/Depres./H_2Burn/Direct Heating*	--	--	.06	--	.18
Hydrogen Combustion Before Vessel Breach	--	--	--	.10	.10
In-Vessel Steam Explosion (Missile Breach)	--	--	--	--	.01
Large Isolation Failure or Preexisting Leak	--	.003	.003	.004	.004
	1.00	1.00	1.00	1.00	1.00
Fraction of Above Affected by:					
(a) Preexisting Small Containment Leak**	.02	.03	.03	.06	.06
(b) Induced Small Steam Generator Tube Leak**	--	.001	.001	.1	.1

III. PRINCIPAL CONTAINMENT PATHWAYS

Continued on next page.

* Case 2 for central and pessimistic walkthroughs includes the effects of in-vessel steam explosions and ex-vessel direct heating. Case 1 does not include these effects. Direct heating includes hydrogen burning if the atmosphere is flammable. See text for a discussion of limitations associated with the results for steam explosions and direct heating.

** Small containment leaks include those producing leakage greater than containment design but not large enough to preclude gradual overpressurization from steam and/or noncondensibles. Likewise, small steam generator tube leaks include those producing primary-to-secondary leakage greater than design but not large enough to depressurize the primary system below the accumulator setpoint before vessel breach.

Table 3.22 (Continued)

RESULTS FOR SEQUOYAH TMLB'

III. PRINCIPAL CONTAINMENT PATHWAYS

General: Initial loss of coolant is through the PORV. Induced RCS failure in hot (C) and cold (O&C) legs or small SGTR (P) may occur. Vessel failure usually occurs with the RCS at high or intermediate pressure. The ice condenser is very seldom bypassed or void of ice until very late. A coolable debris bed forms usually (O), sometimes (C), or very seldom (P). The auxiliary building is nearly always bypassed after containment failure. Oxidation and vaporization releases rarely (O), sometimes (C), or usually (P) occur.

	BMI-2104 CALCULATION
(1) No Containment Failure: A coolable debris bed is formed. The sprays are operable after late power restoration.	No, but releases would be minimal.
(2) Basemat Meltthrough: A coolable debris bed is not formed, the sprays are operable after late power restoration preventing overpressure.	No, but should be similar to any basemat meltthrough case.
(3) Late Overpressurization: Sprays are inoperable (O) or operable (C) after late power restoration. Containment failure occurs due to steam and/or noncondensible gas buildup.	Calculation corresponds to central case with no sprays. Other calculations are available which address effects of sprays late in accident.
(4) Late Hydrogen Burn: Containment fails due to a late hydrogen burn. The sprays do not operate.	No, but should be similar to late overpressurization case with no sprays.
(5) Early Steam Spike/Hydrogen Burn: The sprays do (C) or do not (P) operate. The debris is not coolable. The auxiliary building is usually bypassed.	Yes for case with no sprays late. Other calculations available which address effects of late sprays.
(6) Early Steam Spike/Hydrogen Burn/Direct Heating: The sprays do (C) or do not (P) operate. The debris is not coolable. The auxiliary building is usally bypassed.	No, calculations available involving direct heating.
(7) Early Hydrogen Burn: The sprays do not operate. The debris is not coolable. The auxiliary building is usually bypassed.	No, but enough information may be available from case involving failure at vessel breach due to a hydrogen burn. Some additional information would be required for cases involving direct heating at vessel breach.

TABLE 3.23 SENSITIVITY OF SAMPLE CASE RESULTS
TO ALTERNATIVE ASSIGNMENTS OF NUMERICAL VALUES

	OPTIMISTIC				CENTRAL				PESSIMISTIC			
Unlikely	.1	.01	.01	.1	.1	.01	.01	.1	.1	.01	.01	.1
Remote Poss.	.001	.001	.0001	.01	.001	.001	.0001	.01	.001	.001	.0001	.01
No Failure	.95	.99	.99	.93	.46	.54	.54	.45	.006	.007	.007	.006
Meltthrough	.05	.006	.005	.06	.45	.45	.45	.45	.06	.06	.06	.06
Late Overpressure	.001	.001		.01	.09	.01	.01	.09	.03	.02	.02	.03
Late Leak									.01	.01	.01	.01
Late H_2 Burn					.001			.001	.06	.05	.05	.06
Early Leak									.03	.03	.03	.03
Steam Spike												
Early H_2 Burn									.04	.04	.04	.04
SS + H_2 Burn									.77	.77	.77	.77
Isol. Failure					.002	.002	.002	.002	.002	.002	.002	.002