CONTAINMENT EVENT ANALYSIS AND ESTIMATION OF SOURCE TERM FREQUENCIES

APPENDIX TO NUREG-0956

by

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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 basemat meltthrough, at which time the containment partly depressurized.

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 or basemat meltthrough; 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, loss of feedwater transients (TMLU) or anticipated transients without scram (TKMU) for either of the two PWR dry containments, Surry and Zion, whereas the Accident Sequence Evaluation Program (ASEP) identified these accidents as being potentially important. 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 objective 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 six reference plants addressed in BMI-2104.

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.

(By agreement with NRC, this version of the document, submitted in July 1984, is somewhat narrower than the version to be submitted in November 1984. This version treats only 4 plants - Surry, Zion, Peach Bottom, and Grand Gulf - and for those four plants, only the accident sequences treated in BMI-2104 are considered, a total of 12. At this time, we are not including phenomenology that is outside the capability of the BMI-2104 code methodology - most notably steam explosions and direct heating of the containment atmosphere. Accident sequence frequencies account for some but not all of the plant modifications that have occurred as a result of the TMI accident. The November version will address each of these items and remove many of the limitations.)

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 6 references plants. That analysis is scheduled to be completed and documented by summer 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 and upper/lower bounds.

Our primary focus is upon the containment event trees. We have developed a containment event tree for each accident sequence analyzed in this study. Because many of them are similar, we intend to combine them into a single containment event tree for each type of plant.

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

(1) Reactor coolant system failure modes. What is the size and location of the reactor coolant system breach and the pressure in the system at the time of breach?

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

(3) Containment failure modes. 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, 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 the issues that must be addressed in order for the event trees to be quantified. Also shown are the subsets of these 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 Estimate 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 28-30.

We encountered several problems 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 for only the 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 specify a single best estimate but rather a range of possible values. In particular, the CLWG and CPWG generally declined to provide best estimates of containment loading and performance and instead provided low, medium, and high estimates. In the CLWG, concensus was generally reached more often on the low and high estimates than on the medium estimates, whereas the CPWG stated that all three of their estimates were highly conjectural and subject to change. Neither group felt that it was appropriate to specify weighting factors or probability distributions for their results.

When we quantified our containment event trees, therefore, we also 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. Like CLWG and CPWG, 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.

The general method is depicted schematically in Figure 2.1. It is worth noting again that we have calculated the frequencies of accident pathways that can lead to distinct source terms. We have not calculated the magnitudes of the source terms. However, one can often qualitatively judge that certain pathways are similar enough in character to permit them to be binned to the same source term.

2.2 Specific Example Surry S_2D

We will illustrate the application of the method by providing the details for a particular accident sequence - Surry S_2D. This is a sequence initiated by a small-break LOCA with subsequent failure of the emergency core cooling system in the injection mode. Containment sprays are operative as the accident develops toward a core meltdown.

The various questions for S_2D are listed in Table 2.2 together with the answers we assigned. It should first 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.

The remainder of this subsection provides the rationale for our assignment of values in Table 2.2:

Question 1: Likelihood of RCS Break Size in the Larger Range

In previous PRAs, S_2 type LOCAs were taken to represent break areas of 1/2 to 2 inches (RSS, Ref. 17). More recently, the S_2 LOCAs have been subdivided into an S_2 category (~1 to 2 inches) and an S_3 category (~1/2 to 1 inch), with the letter representing a class of LOCAs initiated by reactor coolant pump seal failures. ASEP (Ref. 14) estimates the probability of S_3 to be approximately 10 times that of S_2. Hence, the likelihood of the break size being in the larger range is 0.1.

Question 2. Likelihood of the Break Being in the Hot Leg, Given the RCS Break Size is in the Larger Range

If the initiating event is a pipe break rather than a reactor coolant pump seal failure, the likelihood that the break occurs in the hot leg versus the cold leg is taken to be governed by the respective lengths of piping. From the FSAR, the total length of cold leg and hot leg piping are comparable. Hence the likelihood of the break being in the hot leg was taken to be 0.5.

Question 3. Likelihood of Preexisting Containment Leakage or Isolation Failure Sufficient to Preclude Containment Overpressurization, Given Delayed Failure of Containment Sprays

The containment leakage area required to preclude overpressurization in the event of a loss of containment cooling is generally figured to be at least 4 square inches (BMI, Ref. 5). The likelihood of such an opening in subatmospheric containments is extremely small because the leak would be almost immediately detected. This observation is borne out by precursor data (NRR, Ref. 28), which shows that even very small preexisting leaks are rare for subatmospheric containments. Generally, preexisting leakage is limited by the capacity of the vacuum pumps, corresponding to a leak area of 0.07 in^2 (CPWG, Ref. 4).

The RSS estimate of containment isolation failure for Surry was 2×10^{-3}, which we used for all cases (optimistic/central/pessimistic).

Question 4. Likelihood that the RCS Pressure Falls Below the Accumulator Discharge Pressure

We evaluated two subcases for this question: (a) RCS break size in the larger range (1-2 inch-diameter), and (b) RCS break size in the smaller range (0.5-1 inch-diameter).

First consider the larger size range. BMI-2104 calculations for Surry S_2D assuming a 2-inch diameter break size indicated that the primary system pressure declines to about 130 psia at the time of core slump. This value is well below the accumulator setpoint of 600 psia. IDCOR results (Ref. 7) corroborate the finding that the accumulators discharge before core slump. Other calculations performed with the MARCH and RELAP codes indicate that the accumulator setpoint will be reached prior to vessel breach for break diameters at least as small as 1-inch, but with a decreasing margin for the smaller break size (SASA, Ref 8; BNL, Ref. 29). Thus, we took the answer for this question to range from "likely" to "almost certain", as shown in Table 2.2.

Now consider the smaller size range. Based on an extrapolation of the results mentioned above, we assessed that it was unlikely for the accumulator setpoint to be reached for a 0.5-inch-diameter LOCA but likely for a 1.0-inch-diameter LOCA. We assigned answers to this question accordingly.

Question 5. Likelihood of a Steam Spike Just Following Reactor Vessel Meltthrough

The CLWG and IDCOR program both agreed that a quenching of some of the core debris is likely if water is available. In this sequence, 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 up to the time of vessel breach. (If this were not enough, the portion of accumulator water not discharged prior to vessel breach would discharge into the reactor cavity after vessel breach.) Hence, we took a steam spike to be "likely."

The primary issue between CLWG and IDCOR regards the size of the steam spike rather than its likelihood of occurrence. IDCOR calculated a steam spike of 1.0 bar (15 psi) for the S_2D sequence, assuming that 50% of the core debris quenched. CLWG (Ref. 2) reported that the spike from core debris quenching could be nil or as high as 1.9 bar (27 psi) corresponding to quenching of 100% of the core debris. To this must be added the pressure increment from primary system blowdown, about 0.5 bar (7 psi) according to BMI-2104. 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 0.5 bar (optimistic), 1.5 bar (central), and 2.4 bar (pessimistic).

Question 6. Likelihood of a Global Hydrogen Burn Prior to or Just Following Reactor Vessel Meltthrough

For S_2D, calculations in BMI-2104 indicate that the hydrogen concentration in containment is sufficient for a global burn to occur any time after core slumping into the lower plenum of the reactor vessel. If a burn does not occur prior to reactor vessel meltthrough, many experts (CLWG, BMI-2104, SASA) consider a global hydrogen burn to be a likely occurrence 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. We therefore took the likelihood of a global hydrogen burn prior to or just following the vessel breach to be "unlikely" for the optimistic case and "likely" for the pessimistic case. For the central estimate, we took the likelihood to be "indeterminate".

The size of the burn is also an issue. For a global burn occurring during the time frame of interest, the amount of hydrogen participating in the burn is limited by that which can be produced in-vessel (i.e., during the core heatup and slumping portions of the accident). 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). A separate CLWG submittal (Ref. 3) used a lower bound of 25% and upper bound of 100% for large, dry 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. IDCOR calculated about 25% Zircaloy oxidation in-vessel for Surry S_2D (Ref. 7). 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.

According to the aforementioned CLWG submittal (Ref. 3), the containment pressure increments corresponding to these amounts of Zircaloy oxidation would be 20 psi (optimistic), 38 psi (central), and 71 psi (pessimistic). 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 7. Likelihood of Containment Structural Failure Just Following Reactor Vessel Meltthrough

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 reinforcement. 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 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.

To obtain the failure likelihoods in Table 2.2, we evaluated the containment pressure just following reactor vessel meltthrough by adding the pressure increments for the steam spike and the hydrogen burn discussed above under Questions 6 and 7 to the pressure existing prior to vessel breach. Based on BMI-2104 calculations, we took the containment pressure prior to vessel breach to be 5 psig if containment was not leaking, or 0 psig if it was. Table 2.3 summarizes the pressure estimates leading to the likelihoods in Table 2.2.

<u>Question 8. Likelihood of Large Induc[illegible] Containment Leakage Just Following Reactor Vessel Meltthrough, Given No Structural Failure Occurs</u>

The Containment Performance Working Group (Ref. 4) developed two types of models to evaluate containment leakage before failure. One was based on the degradation of penetration seals caused by exposure to high temperatures for a sustained period of time. The other was based on pressure-induced yielding of the penetration stiffeners, valves, or seals.

The model for Surry was temperature-based, with leakage occurring only for seal temperatures exceeding 350°F. To test the impact of the model on containment response, the CPWG performed calculations for sequence TMLB' (station blackout) with the MARCH code, modified to include the leakage model. The results indicated that containment atmospheric temperatures would rise to about 450°F, that significant leakage would occur, and that the leakage would preclude containment overpressurization.

These results for Surry are now believed to be extremely pessimistic for a variety of reasons. Two of the principal reasons are: (1) the presence of the outboard penetration valves was ignored, and (2) the Mod 1.1 version of MARCH greatly overestimates the containment temperature. Regarding the second point, BMI-2104 calculations for the same TMLB' sequence predicted maximum atmospheric temperatures of only 280°F, far below the threshold required to initiate seal degradation. The temperatures were even lower for the S_2D sequence due to spray operation. A likely reason for the difference is that MARCH 1.1 uses the subroutine INTER to calculate the core-concrete interaction, whereas the BMI-2104 calculations utilized CORCON, an improved core-concrete analysis code.

For these reasons, we concluded that there was no basis to asssume that the Surry containment would leak due to temperature-induced failures of the penetration seals. We did feel, however, that there was a potential for presssure-induced leakage to occur. To investigate this point, we looked at the CPWG leakage model for Zion, which was based on pressure loadings, and assumed that it applied to Surry.

Figure 2.2 shows the pressure dependent leakage model developed by the CPWG. The area of leak which would preclude overpressure failure is very large (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 6 in^2 is considered necessary to preclude overpressure failure, the CPWG medium leak model would never preclude failure; the high leak model would always preclude failure for pressures exceeding 90 psig, and would never preclude failure for pressure less than 90 psig.

There are numerous uncertainties as to the pressures at which leaks develop, and in fact we understand that a present consensus is that the model in Figure 2.2 probably overstates the expected amount of leakage. For this reason, we used the data in Figure 2.2 semi-qualitatively. For the optimistic case, we interpreted Figure 2.2 to imply that no leakage could occur given there was no structural failure. For the central case, our interpretation was that no leak could develop that was large enough to preclude later overpressurization, but a lower-capacity leak (<1 inch^2) was 50% probable. For the pessimistic case, we took the pressure at which large leaks can develop to be 90 psig, from Figure 2.2, with a standard deviation of 15 psi from the RSS. We calculated leakage likelihoods for each of the cases considered in Table 2.3, and then reduced the calculated likelihoods by 50% to reflect the downside uncertainty. We also took Figure 2.2 to imply that a lower-capacity leak (<1 inch^2) was 100% probable for the pessimistic case.

Question 9. Likelihood of Containment Spray Failure within 30 Minutes After Vessel Breach

This question asks whether the containment sprays operate long enough to remove most of the airborne fission products released from the fuel during the melt phase. The RSS considered containment spray failure to be inevitable after containment rupture due to pump cavitation. No other cause of spray failure was considered. However, it is believed that sprays might also fail because of debris in the sump clogging the screens and causing cavitation 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 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.

It is very difficult to obtain specific information about the operability of pumps under conditions that exceed their design basis. The manufacturers, of course, do not guarantee the success or failure of their pumps if operated beyond specifications. The prevailing opinion regarding centrifugal pumps, however, is that they are capable of operating in cavitating environments for hours before failing due to mechanical damage. This opinion is supported by tests conducted at Sandia under Task Action Plan A-43 (Ref. 30).

Based on these observations, the likelihood of containment spray failure is considered somewhere between "unlikely" and "remotely possible" if there is no steam spike and no containment failure. If containment fails, the optimistic judgment is that spray failure is "unlikely" (consistent with an IDCOR observation, Ref. 7), whereas the pessimistic judgment is that spray failure is "likely" (consistent with the RSS). The likelihood descriptors in Table 2.2 reflect these judgments.

Question 10. Likelihood of Containment Spray Failure, Given Survival within the First 30 Minutes After Vessel Breach.

This question asks whether the containment sprays operate long enough to remove most of the fission products released from the fuel during the vaporization (core-concrete) phase. If the sprays do not fail in the first 30 minutes following vessel breach, it is considered possible that earlier pump damage could cause failure during continued operation. The descriptors in Table 2.2 represent our judgments consistent with the observations made under Question 9.

Question 11. Likelihood of Core-Concrete Interaction Producing a Vaporization Release.

The vaporization release may be precluded by either of two occurrences. First, if sprays continue to operate throughout the accident, there is a possibility that a permanently coolable debris bed will form in the reactor cavity. Second, if there is a strong fuel-water interaction (steam spike), the core debris may be scattered so sparsely through containment that significant core-concrete interaction is precluded. Conversely, if there are no sprays and no steam spike, 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 the case where the sprays continue to operate, the IDCOR analysis (Ref. 7) takes the outcome of the accident to be a permanently coolable debris bed. The BMI-2104 analysis, to the contrary, allows the core-concrete interaction to occur with the water in the reactor cavity boiling off faster than it can reinfiltrate. Thus, we took th occurrence of the vaporization release to range from "unlikely" (consistent with the IDCOR analysis) to "likely" (consistent with the BMI-2104 analysis) if sprays are operating.

Question 12. Likelihood of a Global Hydrogen Burn During or Just Following the Vaporization Release.

Because ignition sources are available during this sequence and the atmosphere is not steam-inerted, we took a late hydrogen burn to be a "likely" occurrence provided there is a sufficient amount of hydrogen to support the propagation of the burn.

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 core-concrete interaction is included. However, the oxygen available could only burn about 150% zirconium equivalent. The amount available depends on whether core-concrete interaction releases flammables and whether a prior burn has occurred.

We estimated that a late hydrogen burn could involve as much as 800 lb of hydrogen (optimistic), 1200 lb (central), or 1600 lb (pessimistic) if core-concrete interactions occurred. These figures correspond to hydrogen ignition thresholds of about 6%, 9%, and 12%, respectively, if sprays are operating. These are consistent with the range of ignition thresholds considered in previous analyses (CLWG, Ref. 2; SASA, Ref. 9).

Question 13. Likelihood of Containment Structural Failure from a Global Hydrogen Burn During or Just Following the Vaporization Release.

We calculated the likelihood of containment structural failure from a late hydrogen burn as described under Question 7. Table 2.4 summarizes the pressure estimates leading to the likelihoods in Table 2.2. We obtained containment pressures just prior to the burn from BMI-2104 assuming that the burn occurred just after the peak of the vaporization release.

Question 14. Likelihood of Late Containment Spray Failure, Given Containment Failure Occurs

As described under Question 9, we based our optimistic and pessimistic descriptors for the likelihood of spray failure given containment structural failure on information from IDCOR and the RSS.

Question 15. Likelihood of Containment Leakage from a Global Hydrogen Burn During or Just Following the Vaporization Release, Given No Structural Failure Occurs

We calculated the likelihood of containment leakage from a late hydrogen burn as described under Question 8, using containment pressure loadings from Table 2.4.

Question 16. Likelihood of Basemat Meltthrough, Given the Occurrence of a Core-Concrete Interaction

If the core-concrete interaction occurs (viz., Question 11), 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 valid 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. Based on these observations, we took the occurrence of meltthrough to be "likely" (optimistic) if the sprays have failed. If the sprays continue to operate, we took the occurrence of meltthrough to be less assured, as shown in Table 2.2.

Question 17. Likelihood of Late Containment Overpressurization

Late overpressure failure is considered to be "impossible" if containment sprays continue to operate, consistent with all analyses performed to date (BMI-2104, SASA, IDCOR). It is also considered to be "impossible" if there has been an isolation failure or a pressure-induced leak exceeding about 4 inch^2. Late overpressure failure is estimated to be "certain" if there is no core-concrete interaction and no sprays, because all the core decay energy is transmitted to the containment atmosphere. If there is a core-concrete interaction but no meltthrough, and sprays have failed, some of the energy could be transmitted through the basemat to the substrate underlying containment. Overpressure failure is therefore estimated to be "likely" (optimistic), "almost certain" (central), or "certain" (pessimistic) for this case.

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. We therefore took the occurrence of late containment overpressurization to range from "unlikely", consistent with the above discussion, to "likely", consistent with the BMI-2104 analysis, if basemat meltthrough were to occur.

Question 18. Likelihood that the Release Bypasses the Auxiliary Building, Given Containment Leakage or Structural Failure

Most containment penetrations lead into the auxiliary building. An exception is the equipment hatch; however, CPWG results seem to indicate that pressure-induced leakage through the equipment hatch is not the dominant pathway for leakages that are large enough to preclude containment overpressurization. We therefore estimated that it was "unlikely" that most of the leakage would bypass the auxiliary building.

On the other hand, the auxiliary building subtends only a small portion of the containment structural surface area, and in particular, does not subtend the upper springline. We therefore estimated that it was "likely" that a structural failure of containment would result in bypass of the auxiliary building. For the pessimistic case, we took bypass to be "almost certain".

2.3 Treatment of Verbal Descriptors

Interpretation of words such as "likely", "indeterminate", "unlikely", or "almost impossible" 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 2.5 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 3.5.

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.

Sequences evaluated for Surry in addition to S_2D 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).

TMLB' and AB are somewhat easier to analyze than S_2D because the containment sprays do not operate (hence the question of delayed spray failure is moot). For this first iteration, we assumed that power was not recovered prior to containment failure; the second iteration will include the possibility of power recovery after core melting and before containment failure.

For TMLB', a key question is whether the primary system fails from high temperatures before the core melts through the reactor vessel, and if so, where the failure occurs. Possible locations are the reactor coolant pump seals (cold leg), the steam generation tubes, or the reactor vessel nozzle welds (hot leg). Temperature-induced failure could cause the primary system to depressurize prior to vessel breach, allowing the accumulators to discharge while the core is in the vessel, and reducing the size of the steam spike following vessel breach. (It would also preclude the occurrence of direct atmospheric heating resulting from high pressure ejection of core debris from the vessel; however direct heating was not analyzed in this iteration.) We took the temperature-induced LOCA for TMLB' to range from "unlikely" for the optimistic estimate (consistent with previous PRAs and with BMI-2104 analyses) to "likely" for the pessimistic estimate (consistent with the majority opinion of the CLWG). For the central estimate, we took the likelihood of the induced LOCA to be "indeterminate".

Sequences evaluated for Zion are S_2D 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 (for S_2D) as well as survivability of the sprays. The containment is atmospheric rather than subatmospheric; hence the likelihood of preexisting leakage is somewhat higher. The containment failure pressure is higher than at Surry; hence induced leakage becomes relatively more important as a containment failure mode.

The BWR plants evaluated in this study were Peach Bottom (Mark I containment) and Grand Gulf (Mark III containment). The sequences analyzed for Peach Bottom were TW (transient event with loss of containment cooling), TC (transient event with failure to scram), and AE (large LOCA with failure of emergency core cooling). Sequences analyzed for Grand Gulf were TC, TPI (transient event with stuck-open safety/relief valve and loss of suppression pool cooling), and TQUV (transient event with loss of feedwater and emergency core cooling).

Many of the questions we posed for Peach Bottom and Grand Gulf were quite different from those we posed for Surry and Zion. This is to be expected, since the BWR containment 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 containment fail before the core melts? For the TW and TC sequences, it is usually assumed that containment fails before the emergency core cooling system fails (RSS, SASA, IDCOR). The likely causes of ECCS failure, given containment failure, are cavitation of the pumps or deformation of the cooling lines. There is some likelihood, however, that the emergency core cooling pumps will fail before the containment fails. Possible causes of early ECCS failure are insufficient cooling of the lube oil or underventilation of the pump room. For the AE sequence, containment could fail as a result of overpressurization from steam and hydrogen 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. The possibility of early containment failure caused by hydrogen generation during AE is treated in BMI-2104.

(2) Will the primary system still be pressurized at vessel breach? This question applies only to the TW and TC sequences, since primary system depressurization is guaranteed for AE. In most cases, the automatic depressurization system (ADS) would actuate automatically during these sequences to reduce the primary system pressure before the core melts. (For TW, operation of the steam-driven turbine of the high pressure coolant injection system also reduces the primary system pressure.) However, the ADS would fail to operate automatically if the containment rupture were such that the drywell pressure stayed above ~75 psig. In that case, the safety/relief valve pilot air pressure would be insufficient to actuate the ADS, and the primary system would remain pressurized unless the operator took some special actions.

(3) 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.

(4) Will containment failure lead to failure of the suppression pool function? While no one has published an analysis of whether the suppression pool in a Mark I containment would survive an overpressurization failure of the containment, many structural experts feel that pool survival would be very questionable (FVCS, Ref. 27). Because the containment is a free-standing steel shell structure with a high failure pressure, the forces associated with the failure could be violent. Failure of the suppression pool would be a moot question if the containment failed in the drywell.

(5) Will leak paths develop in the drywell that could cause the suppression pool to be bypassed? This question primarily concerns the AE sequence. The containment penetration seals at Peach Bottom are elastomeric, like those at Surry. The potential for overheating of these seals is greater at Peach Bottom than at Surry, however, because the small size of the drywell compartment makes it more susceptible to thermal loading from the core-concrete interaction (CLWG, SASA). If the seals survive the thermal loading, there is a chance that a leak path could develop as a result of a direct core debris attack on the drywell structure, causing failure of the shell at a location where it is not directly backed by concrete (CLWG).

(6) Will secondary containment be bypassed? The secondary containment could be bypassed if the primary containment failed in the wetwell at a location where there is a direct pathway to the outside environment. The RSS considered this possibility.

(7) Will the standby gas treatment system (SGTS) fail to remove fission products from the secondary containment atmosphere? It is likely that the blowers in the SGTS will operate throughout the accident, but it is also likely that the filters will become ineffective due to one or more of the following occurrences: (a) steam overloading, (b) particulate overloading, and (c) overheating (BMI-2104, FVCS).

In this first iteration, we did not address the change in procedures which allows for the venting of containment. Further, for TC, we neglected the possibility that the primary system might be overloaded by the pressure transient occurring just after containment isolation. The implementation of automatic trip of the reactor coolant pumps should cause the probability of this event to be very low.

The questions we posed for Grand Gulf were similar to Peach Bottom, particularly for the sequences TPI and TC. For these sequences, the primary differences were in the quantification. Grand Gulf is not preinerted as is Peach Bottom; hence pre-existing leakage is somewhat more likely to go undetected. The drywell is contained within the wetwell; thus, drywell leakage results only in suppression pool bypass, not release from containment. The penetration seals are steel welded;

hence temperature-induced leakage is relatively less credible (excepting the possibility of diffusion flames, see below). The containment is a fairly low-pressure concrete structure whose likely point of failure is at the upper springline; hence the suppression pool function is much more likely to survive the failure of containment. The reactor vessel pedestal supports the vessel at the nozzles; hence an attack on the pedestal by core debris could lead to destruction of the vessel and bypass of the suppression pool (CLWG).

For the Grand Gulf TQUV sequence, 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 increasing the likelihood of a steam spike. (2) The high temperatures produced by diffusion flames could induce leakage through drywell or containment penetrations (CLWG). (3) 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. Finally, hydrogen burning in the drywell during the core-concrete attack could lead to overheating of the drywell.

For this first iteration, our quantification of the BWR containment event trees was based far more heavily on the use of verbal descriptors than for the PWRs. We pursued this approach partly because the information base for the BWRs was less complete than for the PWRs, and partly because we had insufficient time to do otherwise. We plan to provide a more quantitative assessment of the BWR event likelihoods during the second iteration.

3. RESULTS

3.1 Results for Surry

3.1.1 Sequence AB

Table 3.1 summarizes the optimistic, central, and pessimistic containment failure modes for sequence AB at Surry. In all three quantifications significant fractions of the containment failures are attributed to late overpressurization due to the accumulation of gases from core-concrete interactions. In the pessimistic quantification, leakage in excess of design leakage but insufficient to preclude late overpressurization is postulated to be induced before the late overpressurization. Basemat meltthrough is another containment failure mode which is significant in all three quantifications: 79% of the optimistic containment failures, 49% of the central containment failures, and 5% of the pessimistic containment failures. Induced leakage in excess of design leakage is postulated in half of the central and all of the pessimistic basemat meltthroughs. BMI-2104 analyzed basemat meltthrough for sequence AB at Surry; however, pressure-temperature-induced leakage in excess of design leakage was not considered.

Pessimistically, a significant fraction (38%) of containment failures is attributed to late hydrogen burns. Such burns are precluded early in the accident due to high steam concentrations; however, eventually such steam inerting could be negated due to condensation on passive heat sinks and aerosols thereby permitting the combustion of accumulated hydrogen and carbon monoxide. This possibility is consistent with recent, unpublished MARCH and CONTAIN analyses performed at Sandia National Laboratories. Containment failure due to a late hydrogen burn is one of the containment failure modes analyzed in BMI-2104.

The pessimistic quantification also indicates a significant fraction (15%) of containment failures due to induced leakage sufficient to preclude gradual overpressurization. Such leakages are postulated early enough to result in higher releases than would be obtained from late overpressurization.

3.1.2 Sequences S_2D and S_3D

Table 3.2 summarizes the optimistic, central, and pessimistic containment failure modes for the sequences S_2D and S_3D at Surry.

Optimistically, the most likely outcome (95%) of $S_{2,3}D$ core melt is no containment failure with core-concrete attack being prevented or arrested before meltthrough or gradual overpressurization can occur. Only a 5% chance of basemat meltthrough and a 0.1% chance of late, gradual overpressurization result from the optimistic quantification. Basemat meltthrough is more likely than gradual overpressure because of continued containment heat removal by the containment sprays. Containment failures due to hydrogen burning do not occur in the optimistic quantification because of optimistic assumptions regarding hydrogen production, core-concrete termination, and ignition thresholds.

The principle containment pathways in the central quantification are the same as for the optimistic quantification; however, in the central quantification, the likelihood of arresting core-concrete interactions before basemat meltthrough is deemed "indeterminant" resulting in a nearly equal split between basemat meltthrough and no containment failure. The fraction of containment failures due to gradual overpressurization also increases in the central quantification due to higher probabilities of delayed containment spray failure. Half of these late, gradual overpressures are postulated to be preceeded by induced leakages in the central quantification. The higher probabilities attached to in-vessel hydrogen production and sustained core-concrete interactions result in a small fraction (0.1%) of the central containment failure being attributed to late hydrogen burns.

Pessimistically, containment failure at vessel breach due to a coincident 27 psi steam spike and combustion of hydrogen from 100% in-vessel Zr oxidation accounts for 79% of the containment failures. Early hydrogen burns contribute 4% and late hydrogen burns contribute 6% of containment failures in the pessimistic quantification. The chances of induced leakage, either due to the containment loadings following vessel breach or later also appear in the pessimistic quantification.

3.1.3 Sequence TMLB'

Table 3.3 summarizes the optimistic, central, and pessimistic containment failure modes for the TMLB' sequence at Surry. In all three quantifications, there is a significant fraction of containment failures attributed to basemat meltthrough. Basemat meltthrough was one of the containment failure modes analyzed in BMI-2104.

In the pessimistic quantification, the largest fraction of containment failures is attributed to late hydrogen burns, postulating that inerting by high steam concentrations would be negated due to condensation on passive heat sinks and aerosols after the buildup of significant hydrogen (and possibly carbon monoxide) concentrations. This result is consistent with recent unpublished results of a MARCH sensitivity study being performed at Sandia National Laboratories.

The pessimistic quantification also indicates significant fractions of containment failure due to late, gradual overpressurization and late, pressure-temperature-induced leakage. These two containment failure modes are mutually exclusive in that the extent of the induced leakage must be sufficient to preclude late, gradual overpressurization. However, in all of the pessimistic gradual overpressures (and in 50% of the central) we postulate that high containment loadings would result in leakage in excess of the design leakage postulated in BMI-2104. Since induced leakage would likely occur long before gradual overpressure, the releases for either the late-leakage or late-overpressure containment failure modes would depend on the magnitude of the induced leakage. The central quantification is similar to the pessimistic quantification except the likelihood of late leakage is greatly reduced.

In the optimistic quantification, no containment failures are attributed to late hydrogen burns. This results from optimistic assumptions regarding the extent of combustible gas production, the ignition threshold, and combustion completeness.

3.2 Results for Zion

3.2.1 Sequence S_2D

Table 3.4 summarizes the optimistic, central, and pessimistic containment failure modes for the sequences S_22D and S_3D at Zion.

The optimistic and central estimates for Zion S_2D are very similar to Surry S_2D, Section 3.1.2. The higher probability of preexisting leakage for Zion results from the fact that Zion is an atmospheric containment whereas Surry is subatmospheric.

Pessimistically, containment failure at vessel breach due to a coincident 27 psi steam spike and combustion of hydrogen from 100% in-vessel Zr oxidation accounts for 27% of the containment failures. The threat from hydrogen burning is generally lower at Zion than at Surry because of the higher containment failure pressure. The chances of induced leakage, either due to the containment loadings following vessel breach or later also appear in the pessimistic quantification. Pessimistically, all late, gradual overpressurizations are assumed to be preceeded by leakage in excess of design leakage due to high pressure-temperature loadings.

3.2.2 Sequence TMLB'

Table 3.5 summarizes the optimistic, central, and pessimistic containment failure modes for the TMLB' sequence at Zion. The results for Zion TMLB' are similar to Surry TMLB', Section 3.1.3, except for the estimates for late hydrogen burning. As opposed to Surry, late hydrogen burns were found not to be a threat for the Zion containment because the atmospheric conditions for flammability were not attained. It was judged that late burns large enough to threaten containment could occur only if containment cooling were restored, but as mentioned in Section 3.4, restoration of power after core degradation was not considered as a pathway in this iteration.

3.3 Results for Peach Bottom

3.3.1 Sequence AE

The conditional containment failure mode probabilities for the AE accident sequence are summarized in Table 3.6 for the optimistic, central, and pessimistic cases.

In the pessimistic case, containment failure probabilities are divided approximately equally between early overpressurization and late overpressurization. Early overpressurization occurs either before or at vessel breach as a result of a buildup of steam and hydrogen in containment. As mentioned in Section 2.4, about 70% cledding oxidation must be assumed to attain sufficient amounts of hydrogen to threaten containment. (Of course, the hydrogen does not burn because the containment is inerted.) If early failure does not occur, containment eventually overpressurizes from the noncondensibles produced by concrete ablation in the drywell. In the pessimistic case, structural failure is estimated to be a more likely outcome than leakage because the outboard containment penetrations are assumed to be sufficiently protected from overheating. The failure most often occurs in the drywell, which causes the suppression pool to be bypassed. (Structural failures in the wetwell also lead to pool bypass in the pessimistic case.)

In the optimistic case, the majority of releases are associated with temperature-induced leakage caused by overheating of the penetration seals during the core-concrete interaction. Since the leakages occur in the drywell, the suppression pool is bypassed.

In the central case, most of the containment failures are late but a significant fraction (~10%) are early. The late containment failures are about equally divided between structural failure and temperature-induced leakage.

3.3.2 Sequence TC

Table 3.7 summarizes the optimistic, central and pessimistic containment failure mode fractions for TC sequences leading to core melt. The most likely containment failure scenario for the TC sequence is one in which the reactor stays at elevated power (20 to 30 percent) leading to rapid heatup of the suppression pool, steam break-through and buildup in containment, containment failure followed by suppression pool boiling (and/or draining), loss of reactor coolant makeup, and core melt. Pessimistically, all containment failures are taken to result in this manner (i.e. before core melt) and are assumed to occur in the drywell as indicated by the structural analyses performed at Ames Laboratory (1).

In the central quantification, the possibility of early containment failure in the wetwell is permitted but deemed unlikely, and, if failure occurs in the wetwell, draining of the suppression pool is considered likely, so that only a 1% chance of containment effluent passing through suppression pool water results in the central quantification. Further, fraction of this 1% is assigned to account for the possibility of a small bypass of the suppression pool (for example backleakage through vacuum breakers or temperature induced drywell leakage).

Optimistically, the probability of wetwell failure is taken to be equal to the probability of drywell failure, and the probability of retaining water in a failed suppression pool is taken to be 0.5. This increases the fraction of early containment failures in which effluent would pass through water in the suppression pool to 23%. Of course the high temperature of the suppression pool water would reduce the effectiveness of fission product scrubbing. In the optimistic quantification, we also permitted (as "unlikely") the possibility that ECC injection would fail early, leading to core melt with containment intact (effectively an accelerated TQUV sequence). The possibility of vessel breach resulting from the initial pressure spike was not considered in our quantification although this could conceivably result if the recirculation pumps failed to trip.

The detailed event tree used to quantify containment failure modes for the BWR sequences included secondary containment effects. However, for the TC sequences involving containment failure before core melt, we judged that if the secondary containment was not failed or bypassed, the secondary containment blowout panels would relieve and the standby gas treatment system would not be extremely effective in removing fission products.

3.3.3 <u>Sequence TW</u>

The conditional probabilities for containment failure modes for TW are summarized in Table 3.8 for optimistic, central, and pessimistic sets of assumptions.

The results for TW are similar to TC, Section 3.3.2, in that the most significant pathway is pre-core melt overpressurization with an unscrubbed release. However, the likelihood of the accident degenerating into a TQUV was judged more likely for TW than for TC, as was the likelihood of early induced leakage. These differences reflect the fact that the TW accident develops much more slowly than TC, with high temperatures and pressures persisting for a much longer period of time before containment fails.

3.4 Results for Grand Gulf

3.4.1 Sequence TC

Table 3.9 summarizes the optimistic, central, and pessimistic containment failure mode fractions for TC core-melt accidents at Grand Gulf. The most likely containment failure mode in all three quantifications is failure before core melt due to overpressure following suppression pool overheating. This containment failure mode is the one analyzed in BMI-2104 for the Grand Gulf TC sequence. The suppression pool survives but is at saturation temperature. Hence, subsequent fission product releases are scrubbed, but the efficiency of scrubbing is lower than if the pool were subcooled.

In the optimistic quantification the possibility that ECC is lost early is recognized although considered "unlikely." Early loss of ECC renders the core subcritical and essentially converts the TC sequence into an accelerated TQUV sequence (see Section 3.4.3). The result, in the optimistic quantification, is a significant fraction (9%) for late containment failure and a small fraction (1%) of no containment failure. Pessimistically, the possibility of some bypass of the suppression pool (due primarily to induced leakage) is recognized although deemed "unlikely."

3.4.2 Sequence TPI

Table 3.10 summarizes the conditional containment failure mode probabilities for TPI based on optimistic, central, and pessimistic sets of assumptions.

The results for TPI are similar to TC, Section 3.4.2. However, the likelihood of the accident degenerating into a TQUV-type accident with a depressurized primary system was taken to be higher for TPI, as was the likelihood of early induced leakage. As mentioned in Section 3.3., these differences reflect the much longer period of time leading to the buildup of temperature and pressure in containment.

3.4.3 Sequence TQUV

Table 3.11 summarizes the optimistic, central, and pessimistic containment failure mode fractions for TQUV core-melt accidents. In all three quantifications, there is a significant fraction of containment failures which occur "late" -- more than one-half hour after vessel breach. These late containment failures are due primarily to the accumulation of non-condensible gases from concrete ablation with a smaller contribution from late combustion events. In the optimistic and central quantifications, such late containment failures are most likely. Pessimistically, the fraction of early containment failures due to hydrogen burning (especially hydrogen burning at vessel breach) approaches the fraction of late containment failures.

Pessimistically, drywell leakage which would result in a small bypass of the suppression pool is considered unlikely early but indeterminant late. This results in a relatively significant pessimistic fraction of containment failures which are accompanied by small bypass of the suppression pool.

In the optimistic and central quantifications, the possibility of leakage sufficient to prevent overpressurization of containment is recognized, albeit unlikely; so that, some containment failures occur by early and late leakages.

Finally, in the optimistic case, there is a significant fraction (10%) attributed to no containment failure. The containment would not fail if hydrogen releases were small and the concrete ablation was limited due to spreading of the debris within the drywell.

3.5 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 Surry S_2D using the four alternative numerical sets in Table 2.5. The results, depicted in Table 3.12, indicate that the variation of conditional probability within each class (optimistic, central, pessimistic) is small compared to the difference in results between classes.

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 pessimistic walk-throughs.

4. OVERALL SUMMARY AND RECOMMENDATION

(We have not yet prepared an overall summary of our results nor completed our consideration of recommendations for additional source term calculations. Based on our work to date, however, we are able to make some preliminary observations on the latter subject. The following paragraphs provide these observations.)

4.1 Surry and Zion

The principal containment pathways identified in this appendix for Surry AB, $S_{2,3}D$, and TMLB' are, for the most part, treated in existing BMI source term calculations. In many cases, however, it would be necessary to combine or extrapolate the results of existing BMI calculations to achieve complete coverage. For example, a calculation exists for the Surry AB sequence in which late containment failure occurs due to hydrogen burning, but this pessimistically significant containment failure mode is not specifically addressed in the BMI calculations for the Surry TMLB' sequence. By coupling primary system results from the BMI TMLB' calculation with containment results from the BMI AB-gamma calculation, one could achieve a surrogate for the TMLB' -late gamma scenario. In other cases, existing BMI calculations for one scenario may serve as adequate surrogates for other scenarios. For example, BMI calculations performed for a hot-leg S_2D could be used as a conservative surrogate for a cold-leg S_2D at the same plant.

We have, however, identified two areas in which additional calculations may be warranted. First, in Surry sequences, there is a significant occurrence of late leakages in our pessimistic results. If the final CPWG model for pressure-induced leakage is consistent with the assumptions cited in Section 2 of this report, then additional leakage calculations for Surry would appear warranted. BMI has already performed such leakage calculations for Zion. Second, in TMLB' accidents it has been postulated that the reactor coolant system pressure boundary could fail due to high temperatures relatively early in the accident (near the beginning of the melt release). If such failure were to occur in the hot leg, existing BMI calculations (e.g., AB-hot leg) might well serve as adequate surrogates for the primary system retention. However, if temperature-induced steam generator tube ruptures prove feasible in TMLB' and TMLU accidents, the resulting source term would be unique in that fission products would be relieved directly to the atmosphere through the main steam relief valve.

4.2 Peach Bottom

We have identified two areas in which the existing BMI calculations for Peach Bottom appear insufficient. First, considering the frequency of TQUV sequences and the possibility of ECC pump failure before containment failure in TC and TW sequences, we recommend source term calculations be performed for the TQUV sequence at Peach Bottom. Second, scenarios involving late leaks or late containment failures must currently be conservatively binned with scenarios involving early containment failure. This seems overly conservative, and we recommend late leakage be addressed as one of the possible TQUV containment failure modes.

4.3 Grand Gulf

Both the TC and TW sequences lack a calculation which includes a small bypass of the suppression pool. It would not necessarily take a large fraction of the flow bypassing the pool to significantly change the magnitude of the fission product release. Thus, we recommend that a calculation be performed for either TC or TW with leakage through the drywell wall developing soon after vessel breach. The TC sequence is probably the better choice since it has a higher source term.

A similar situation exists for the TQUV sequence, in that none of the calculations have included suppression pool bypass. The recommended scenario would be an early suppression pool bypass followed by late containment failure. In addition, our pessimistic quantification showed a high likelihood of containment failure at vessel breach. Since the calculation performed in BMI-2104 had containment failing very late, we recommend that an early failure scenario be calculated. From our results, it would be advisable to include a small suppression pool bypass in this calculation.

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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	--	--	--	--	--
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
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

*Based on information made available prior to August 1984. X marks where issue was used for a basis or where information source is applicable.

TABLE 2.2. EVENT DESCRIPTIONS AND LIKELIHOODS FOR SURRY S_2D

EVENT	PRIOR EVENTS	LIKELIHOOD OPTIMISTIC	CENTRAL	PESSIMISTIC
1. RCS Break Size in the Larger Range		.1	.1	.1
2. RCS Break in the Hot Leg	RCS Break Size in the Larger Range	.5	.5	.5
3. Preexisting Containment Leakage or Isolation Failure Sufficient to Preclude Gradual Overpressurization		.002	.002	.002
4. RCS Pressure Falls Below the Accumulator Discharge Pressure	RCS Break Size in the Larger Range	Almost Certain	Likely	Likely
	RCS Break Size in the Smaller Range	Likely	Indeter-minate	Unlikely
5. Steam Spike Following Reactor Vessel Meltthrough	Containment Sprays Operating	Likely	Likely	Likely
6. Global Hydrogen Burn Prior to or Just Following Reactor Vessel Meltthrough	Containment Sprays Operating	Unlikely	Indeter-minate	Likely
7. Containment Structural Failure Just Following Reactor Vessel Meltthrough	Hydrogen Burn with Preexisting Leak	-0-	-0-	.32
	Hydrogen Burn w/o Preexisting Leak	-0-	-0-	.45
	Steam Spike + Hydrogen Burn with Pre-existing Leak	-0-	-0-	.91
	Steam Spike + Hydrogen Burn w/o Pre-existing Leak	-0-	-0-	.95
8. Containment Leakage Sufficient to Preclude Gradual Overpressurization Induced Just Following Reactor Vessel Meltthrough	Hydrogen Burn and No Structural Failure	-0-	-0-	.16
	Steam Spike + Hydrogen Burn and No Structural Failure	-0-	-0-	.45
9. Containment Spray Failure Occurring Within 30 Minutes After Vessel Breach	No Steam Spike and No Containment Failure	Remotely Possible	Remotely Possible	Unlikely
	Steam Spike and No Containment Failure	Remotely Possible	Unlikely	Indeter-minate
	Containment Failure	Unlikely	Unlikely	Likely
10. Containment Spray Failure Occurring Within 90 Minutes After Vessel Breach	Steam Spike and No Containment Failure and No Prior Spray Failure	Remotely Possible	Unlikely	Inter-minate
	Containment Failure and No Prior Spray Failure	Unlikely	Unlikely	Likely
11. Core-Concrete Interaction which Produces a Vaporization Release	Steam Spike and Sprays Inoperative	Indeter-minate	Likely	Almost Certain
	Sprays Operating	Unlikely	Indeter-minate	Likely
12. Global Hydrogen Burn During or Just Following the Vaporization Release	Atmosphere is Flammable	Likely	Likely	Likely
13. Containment Structural Failure from a Global Hydrogen Burn During or Just Following the Vaporization Release	No Prior Burn, No Core-Concrete, Sprays Operating or Leak Exists	-0-	-0-	.27
	No Prior Burn, Core-Concrete Occurs, Sprays Operating or Leak Exists	-0-	-0-	.27
	Prior Burn, Core-Concrete Occurs, Sprays Operating or Leak Exists	-0-	-0-	-0-
	No Prior Burn, No Core-Concrete, No Spray or Leak	-0-	-0-	.77
	No Prior Burn, Core-Concrete Occurs, No Spray or Leak	-0-	.005	.77
	Prior Burn, Core-Concrete Occurs, No Spray or Leak	-0-	.005	-0-
14. Late Containment Spray Failure	Containment Failure Occurs	Unlikely	Unlikely	Likely
15. Late Containment Leakage	Late Hydrogen Burn, Sprays Operating, No Prior Burn, No Prior Leak	-0-	-0-	.09
	Late Hydrogen Burn, Sprays Inoperative, No Prior Burn, No Prior Leak	-0-	-0-	.3
	No Late Hydrogen Burn, Sprays Inoperative	-0-	-0-	.2

(Continued)

TABLE 2.2. EVENT DESCRIPTIONS AND LIKELIHOODS FOR SURRY S_2D
CONT'D

EVENT	PRIOR EVENTS	LIKELIHOOD OPTIMISTIC	LIKELIHOOD CENTRAL	LIKELIHOOD PESSIMISTIC
16. Basemat Meltthrough	Sprays Inoperative, Core-Concrete Occurs	Likely	Almost Certain	Certain
	Sprays Operating, Core-Concrete Occurs	Indeterminate	Likely	Almost Certain
17. Late Containment Overpressurization	No Basemat Meltthrough, No Spray or Leak	Likely	Almost Certain	Certain
	Basemat Meltthrough Occurs, No Spray or Leak	Unlikely	Indeterminate	Likely
18. Release Bypasses the Auxiliary Building	Above Ground Structural Failure	Likely	Likely	Almost Certain
	Above Ground Containment Leak	Unlikely	Unlikely	Unlikely

TABLE 2.3. ESTIMATES OF CONTAINMENT PRESSURE, LOADING AND CAPACITY FOR SURRY S_2D

	Optimistic	Central	Pessimistic
Containment Pressure Just Following Vessel Breach (psig):			
H_2 Burn With Preexisting Leak	27	45	78
H_2 Burn Without Preexisting Leak	32	50	83
Steam Spike + H_2 Burn With Preexisting Leak	27	60	105
Steam Spike + H_2 Burn Without Preexisting Leak	32	65	110
Containment Pressure Causing Structural Failure (± Standard Deviation)(psig)	119 (±2.5)	119 (± 15)	85 (± 15)

Table 2.4. ESTIMATES OF CONTAINMENT PRESSURE CAPACITY AND LOADING FROM A LATE HYDROGEN BURN

	Optimistic	Central	Pessimisti
Containment Pressure Following a Late Hydrogen Burn (psig):			
No prior burn, No core-concrete, Spray or leak exists	5*	43	76
No prior burn, Core-concrete occurs, Spray or leak exists	43	59	76
Prior burn occurs, Core-concrete occurs, Spray or leak exists	43	59	43**
No prior burn, No core-concrete, No spray or leak	25*	63	96
No prior burn, Core-concrete occurs, No spray or leak	63	79	96
Prior burn occurs, Core-concrete occurs, No spray or leak	63	79	63**
Containment Pressure Causing Structural Failure (± Standard Deviation)(psig)	119 (±2.5)	119 (±15)	85 (±15)

*No burning occurs because of insufficient hydrogen
**Oxygen depletion occurs before burn completion.

TABLE 2.5. ALTERNATIVE ASSIGNMENT OF VALUES TO VERBAL DESCRIPTORS

VERBAL DESCRIPTOR	LIKELIHOOD ALT. 1 (BASE CASE)	ALT. 2	ALT. 3	ALT. 4
Certain 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 AB

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
AB (Hot Leg)		$<1\times10^{-9}$	
AB (Cold Leg)		$<1\times10^{-9}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.01	--	--
Basemat Meltthrough	.79	.49	.05
Late Overpressurization	.20	.50	.41
Late Induced Leakage	--	--	.15
Late Hydrogen Burn	--	.002	.38
Early Induced Leakage	--	--	--
Early Steam Spike	--	--	--
Early Hydrogen Burn	--	--	--
Early Steam Spike + H_2 Burn	--	--	--
Isolation Failure or Preexisting Leak	.002	.002	.002

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) Basemat Meltthrough. Vaporization release occurs. A low-capacity leak may develop (central and pessimistic).	Yes, but without leak
(2) Late Overpressurization. Vaporization release may or may not occur (optimistic) or occurs (central and pessimistic). A low-capacity leak may develop (central and pessimistic).	No
(3) Late Induced Leakage. Vaporization release occurs.	No
(4) Late Hydrogen Burn. Vaporization release occurs.	Yes
(5) Isolation Failure or Preexisting Leak. Vaporization release occurs (central and pessimistic) or may occur (optimistic).	Yes

TABLE 3.2
RESULTS FOR SURRY $S_{2,3}D$

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
S_2D Hot Leg		$5x10^{-6}$	
S_2D Cold Leg		$5x10^{-6}$	
S_3D Cold Leg		$9x10^{-5}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.95	.46	.006
Basemat Meltthrough	.05	.45	.06
Late Overpressurization	.001	.09	.03
Late Induced Leakage	--	--	.01
Late Hydrogen Burn	--	.001	.06
Early Induced Leakage	--	--	.03
Early Steam Spike	--	--	--
Early Hydrogen Burn	--	--	.04
Early Steam Spike + H_2 Burn	--	--	.77
Isolation Failure or Preexisting Leak	.002	.002	--

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) No Containment Failure. Sprays survive. Core is coolable in reactor cavity, hence no vaporization release.	No, but source term similar to (2)
(2) Basemat Meltthrough. Sprays survive. Core attacks basemat, and there is a vaporization release.	Yes, for S_2D hot leg
(3) Late Overpressurization. Sprays fail either before or after vaporization release. A low capacity leak may develop in central and pessimistic cases.	No
(4) Late Hydrogen Burn. Sprays fail either before or after vaporization release.	No
(5) Early Steam Spike & H_2 Burn. Sprays fail at containment failure. Vaporization release occurs.	Yes, for S_2D cold leg
(6) Isolation Failure or Preexisting Leak. Sprays survive (central) or fail after vessel breach (pessimistic). Vaporization release occurs (pessimistic) or may occur (central).	No

TABLE 3.3
RESULTS FOR SURRY TMLB'

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
TMLB Short Term		$2x10^{-5}$	
TMLB Long Term		$1x10^{-4}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.006	--	--
Basemat Meltthrough	.48	.47	.04
Late Overpressurization	.52	.52	.38
Late Induced Leakage	--	--	.14
Late Hydrogen Burn	--	.005	.43
Early Induced Leakage	--	--	--
Early Steam Spike	--	--	--
Early Hydrogen Burn	--	--	.001
Early Steam Spike + H_2 Burn	--	--	--
Isolation Failure or Preexisting Leak	.002	.002	.002

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) <u>No Containment Failure</u>. Vaporization Release occurs.	No, but source term is similar to (2)
(2) <u>Basemat Meltthrough</u>. Vaporization release occurs. A low-capacity leak may develop (central and pessimistic).	Yes, but without leak
(3) <u>Late Overpressurization</u>. Vaporization release may or may not occur (optimistic) or does occur (central and pessimistic). A low capacity leak may develop (central and pessimistic).	No
(4) <u>Late Induced Leakage</u>. Vaporization release occurs.	No.
(5) <u>Late Hydrogen Burn</u>. Vaporization release occurs.	No
(6) <u>Early Hydrogen Burn</u>. Vaporization release occurs.	No.
(7) <u>Isolation Failure or Preexisting Leak</u>. Vaporization release occurs (central and pessimistic) or may occur (optimistic).	No

TABLE 3.4
RESULTS FOR ZION S_2D

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
SEFC (Seal LOCA due to loss of component cooling water)		$2x10^{-4}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.95	.49	.02
Basemat Meltthrough	.05	.45	.17
Late Overpressurization	--	.05	.10
Late Induced Leakage	--	--	.13
Late Hydrogen Burn	--	--	--
Early Induced Leakage	--	--	.30
Early Steam Spike	--	--	--
Early Hydrogen Burn	--	--	--
Early Steam Spike + H_2 Burn	--	--	.27
Isolation Failure or Preexisting Leak	.003	.009	.009

III. PRINCIPAL CONTAINMENT PATHWAYS

	BM1-2104 CALC.
(1) <u>No Containment Failure</u>. Sprays survive. Debris is cooled in reactor cavity, and there is no vaporization release.	Yes
(2) <u>Basemat Meltthrough</u>. Sprays survive. A vaporization release occurs.	No, but source term is similar to (1)
(3) <u>Late Overpressurization</u>. Sprays fail either before or after the vaporization release. A low-capacity leak may be induced (central and pessimistic).	No
(4) <u>Late Induced Leakage</u>. Sprays fail after the vaporization release.	No
(5) <u>Early Induced Leakage</u>. A leak is induced shortly after vessel breach by a pressure spike (HB+SS). Sprays may or may not survive. Vaporization release occurs.	No
(6) <u>Early Steam Spike + H_2 Burn</u>. Sprays fail at containment breach. Vaporization release occurs.	No
(7) <u>Isolation Failure or Preexisting Leak</u>. Sprays survive (optimistic and central) or fail after vessel breach (pessimistic). Vaporization release does not occur (optimistic) or does occur (pessimistic).	No

TABLE 3.5
RESULTS FOR ZION TMLB'

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
TE (Seismic Induced; leads to seal LOCA)		6×10^{-6}	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.006	--	--
Basemat Meltthrough	.48	.45	.04
Late Overpressurization	.52	.54	.44
Late Induced Leakage	--	--	.50
Late Hydrogen Burn	--	--	--
Early Induced Leakage	--	--	--
Early Steam Spike	--	--	--
Early Hydrogen Burn	--	--	--
Early Steam Spike + H_2 Burn	--	--	--
Isolation Failure or Preexisting Leak	.003	.009	.009

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) <u>No Containment Failure</u>. Vaporization release occurs.	Yes
(2) <u>Basemat Meltthrough</u>. Vaporization release occurs. A low capacity leak may develop (central and pessimistic).	No
(3) <u>Late Overpressurization</u>. Vaporization release does not occur (optimistic) or does occur (central and pessimistic). A low capacity leak may develop (central and pessimistic).	No No
(4) <u>Late Induced Leakage</u>. Vaporization release occurs.	No
(5) <u>Early Steam Spike + H_2 Burn</u>. Vaporization release occurs.	No
(6) <u>Isolation Failure or Preexisting Leak</u>. Vaporization release may or may not occur (optimistic and central) or does occur (pessimistic).	No

TABLE 3.6
PEACH BOTTOM AE

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
AE		$2x10^{-7}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	--	--	--
Late Induced Leakage			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	.90	.45	.05
Late Overpressurization			
Scrubbed	.04	.02	.005
Partly Scrubbed	--	--	--
Not Scrubbed	.05	.43	.44
Leakage Induced Before or at Vessel Breach			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	.005
Overpres. Before or at Vessel Breach			
Scrubbed	--	.005	.005
Partly Scrubbed	--	.005	.04
Not Scrubbed	--	.09	.45
Pre-Core-Melt Overpressurization			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) Late Induced Leakage/Unscrubbed Release. No early containment leakage or failure. A temperature-induced leak develops in the drywell during the core-concrete interaction, bypassing the suppression pool but preventing gross containment failure. The secondary containment remains intact (blowout panels relieve). The standby gas treatment system does not filter the vaporization release (central & pessimistic)	No
(2) Late Overpressurization/Unscrubbed Release. No containment leakage, no early containment failure. Containment fails either in the drywell, causing bypass of the suppression pool, or in the wetwell (optimistic and central), causing the suppression pool to drain. The standby gas treatment system does not filter the vaporization release.	No No
(3) Overpressurization Before or at Vessel Breach/Unscrubbed Release. Containment failure occurs in the drywell early due to buildup of steam and hydrogen. Suppression pool is bypassed, and the secondary containment and the standby gas treatment system fail such that the release is not filtered.	Yes

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TABLE 3.7
PEACH BOTTOM TC

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
TC		$7x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	--	--	--
Late Induced Leakage			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Late Overpressurization			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Leakage Induced Before or at Vessel Breach			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Overpres. Before or at Vessel Breach			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Pre-Core-Melt Overpressurization			
Scrubbed	.23	.009	.005
Partly Scrubbed	--	.001	.005
Not Scrubbed	.67	.99	.99
Equivalent to TQUV	.10	.001	--

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) Pre-Core Melt Overpressurization. Containment fails in the wetwell before core-melt due to steam overpressurization. The primary system has depressurized before core-melt. The suppression pool remains filled and is not bypassed. A vaporization release occurs and secondary containment is not bypassed.	No
(2) Pre-Core-Melt Overpressurization/Unscrubbed Release. Containment fails either in the drywell, causing bypass of the suppression pool, or in the wetwell, causing the pool to drain. The primary system has depressurized before vessel breach. A vaporization release occurs, and secondary containment is not bypassed.	Yes except BMI calculation assumes primary system remains pressurized
(3) Equivalent to TQUV. The ECCS pumps fail before containment fails. The core melts before containment failure, and the accident progresses as an accelerated TQUV sequence. (TQUV has not yet been analyzed for Peach Bottom).	No

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TABLE 3.8
PEACH BOTTOM TW

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
TW		$8x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	--	--	--
Late Induced Leakage			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Late Overpressurization			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Leakage Induced Before or at Vessel Breach			
Scrubbed	.01	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	.01	--	--
Overpres. Before or at Vessel Breach			
Scrubbed	.006	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	.02	--	--
Pre-Core-Melt Overpressurization			
Scrubbed	.11	.008	.005
Partly Scrubbed	--	.001	.005
Not Scrubbed	.34	.89	.99
Equivalent to TQUV	.50	.10	.001

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) Pre-Core Melt Overpressurization/Scrubbed Release. Containment fails in the wetwell before core-melt due to steam overpressurization. The primary system has depressurized before core-melt. The suppression pool remains filled and is not by-passed. A vaporization release occurs and secondary containment is not bypassed.	No
(2) Pre-Core Melt Overpressurization/Unscrubbed Release. Containment fails in the drywell causing bypass of the suppression pool, or in the wetwell, causing the pool to drain. The primary system has depressurized before vessel breach. A vaporization release occurs, and secondary containment is not bypassed.	Yes
(3) Equivalent to TQUV. The ECCS pumps fail before containment fails. The core melts before containment failure, and the accident progresses as an accelerated TQUV sequence. (TQUV has not yet been analyzed for Peach Bottom).	No

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TABLE 3.9
GRAND GULF TC

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
$T_{23}C$		$5x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.009	--	--
Late Induced Leakage			
Scrubbed	.004	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Late Overpressurization			
Scrubbed	.09	.001	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Leakage Induced Before or at Vessel Breach			
Scrubbed	.001	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Overpres. Before or at Vessel Breach			
Scrubbed	.001	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Pre-Core-Melt Overpressurization			
Scrubbed	.90	1.00	.90
Partly Scrubbed	.001	.002	.10
Not Scrubbed	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) <u>Late Overpressurization/Scrubbed Release</u> The ECCS pumps fail due to high temperature before containment failure. The scenario progresses as an accelerated TQUV scenario. The primary system depressurizes. Containment does not fail or leak early. Core-concrete interactions occur yielding a vaporization release and late containment failure due to accumulation of noncondensible gases.	No
(2) <u>Pre-Core-Melt Overpressurization/Scrubbed Release</u>. Containment fails before core-melt due to steam overpressurization. The primary system has depressurized. There is no bypass of the suppression pool. A vaporization release occurs.	Yes, except BMI calculation assumes primary system remains pressurized.
(3) <u>Pre-Core-Melt Overpressurization/Small Bypass</u> Containment fails before core-melt due to steam overpressurization. The primary system has depressurized. A leak through the drywell wall develops after vessel breach which allows a small bypass of the suppression pool. A vaporization release occurs.	No

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TABLE 3.10
GRAND GULF TPI

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
$T_{23}PQI$		$2x10^{-7}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.05	--	--
Late Induced Leakage			
Scrubbed	.02	.004	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Late Overpressurization			
Scrubbed	.42	.08	--
Partly Scrubbed	--	.009	--
Not Scrubbed	--	--	--
Leakage Induced Before or at Vessel Breach			
Scrubbed	.03	.001	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--
Overpres. Before or at Vessel Breach			
Scrubbed	.03	.007	--
Partly Scrubbed	--	.001	--
Not Scrubbed	--	--	--
Pre-Core-Melt Overpressurization			
Scrubbed	.45	.90	.90
Partly Scrubbed	--	.002	.10
Not Scrubbed	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) <u>No Containment Failure</u>. The ECCS pumps fail due to high temperature before containment failure. The scenario progresses as a retarded TQUV scenario. The primary system depressurizes. Containment does not fail or leak early. Core-concrete interactions are arrested and steady state is achieved before containment failure.	No, but similar to TQUV.
(2) <u>Late Overpressurization/Scrubbed Release</u>. Same as (1) except as follows. Core-concrete interactions occur yielding a vaporization release and late containment failure due to accumulation of noncondensible gases.	No
(3) <u>Pre-Core-Melt Overpressurization/Scrubbed Release</u>. Containment fails before core-melt due to steam overpressurization. The primary system has depressurized. There is no bypass of the suppression pool, and a vaporization release occurs.	Yes
(4) <u>Pre-Core-Melt Overpressurization/Small Bypass</u> Same as (3) except as follows. A leak through the drywell wall develops after vessel breach which allows a small bypass of the suppression pool, and a vaporization release occurs.	No

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TABLE 3.11
GRAND GULF TQUV

I. SEQUENCE FREQUENCIES

	OPT.	CENTRAL	PESS.
T_1QUV		$4x10^{-6}$	

II. CONTAINMENT FAILURE MODE PROBABILITIES

	OPT.	CENTRAL	PESS.
No Containment Failure	.09	.001	--
Late Induced Leakage			
Scrubbed	.04	.04	--
Partly Scrubbed	--	.004	--
Not Scrubbed	--	--	--
Late Overpressurization			
Scrubbed	.85	.79	.23
Partly Scrubbed	.001	.09	.28
Not Scrubbed	--	--	--
Leakage Induced Before or at Vessel Breach			
Scrubbed	.003	.004	.002
Partly Scrubbed	--	.004	.001
Not Scrubbed	--	--	--
Overpres. Before or at Vessel Breach			
Scrubbed	.006	.07	.22
Partly Scrubbed	--	.007	.27
Not Scrubbed	--	--	--
Pre-Core-Melt Overpressurization			
Scrubbed	--	--	--
Partly Scrubbed	--	--	--
Not Scrubbed	--	--	--

III. PRINCIPAL CONTAINMENT PATHWAYS

	BMI-2104 CALC.
(1) No Containment Failure. The primary system is depressurized before vessel breach. Core-concrete interactions are arrested and steady state is achieved.	No
(2) Late Overpressurization/Scrubbed Release The primary system is depressurized before vessel breach. Hydrogen burns in the wetwell may cause water to overflow into the drywell. If so, a steam spike occurs at vessel breach, and diffusion flames in the wetwell may cause a small containment leak to develop. Core-concrete interactions occur, giving rise to a vaporization release and late containment failure due to the buildup of noncondensible gases.	Yes, except no modeling of steam spikes, diffusion flames, or containment leaks.
(3) Late Overpressurization/Small Bypass. Same as (2) except a leak is induced in the drywell wall after vessel breach such that a small bypass of the suppression pool occurs.	No
(4) Overpressurization Before or at Vessel Breach/ Scrubbed Release. The primary system depressurizes before vessel breach. A steam spike occurs forcing hydrogen into the wetwell which burns at about the same time. The containment fails. Core-concrete interactions take place yielding a vaporization release. All releases from the primary system pass through the suppression pool.	No
(5) Overpressurization Before or at Vessel Breach/ Small Bypass. Same as (4) except a leak is induced in the drywell wall after vessel breach allowing a small bypass of the suppression pool.	No

TABLE 3.12 SENSITIVITY OF SURRY S_2D 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				

TO BE PROVIDED

FIGURE 2.1. Illustration of Containment Event Methodology

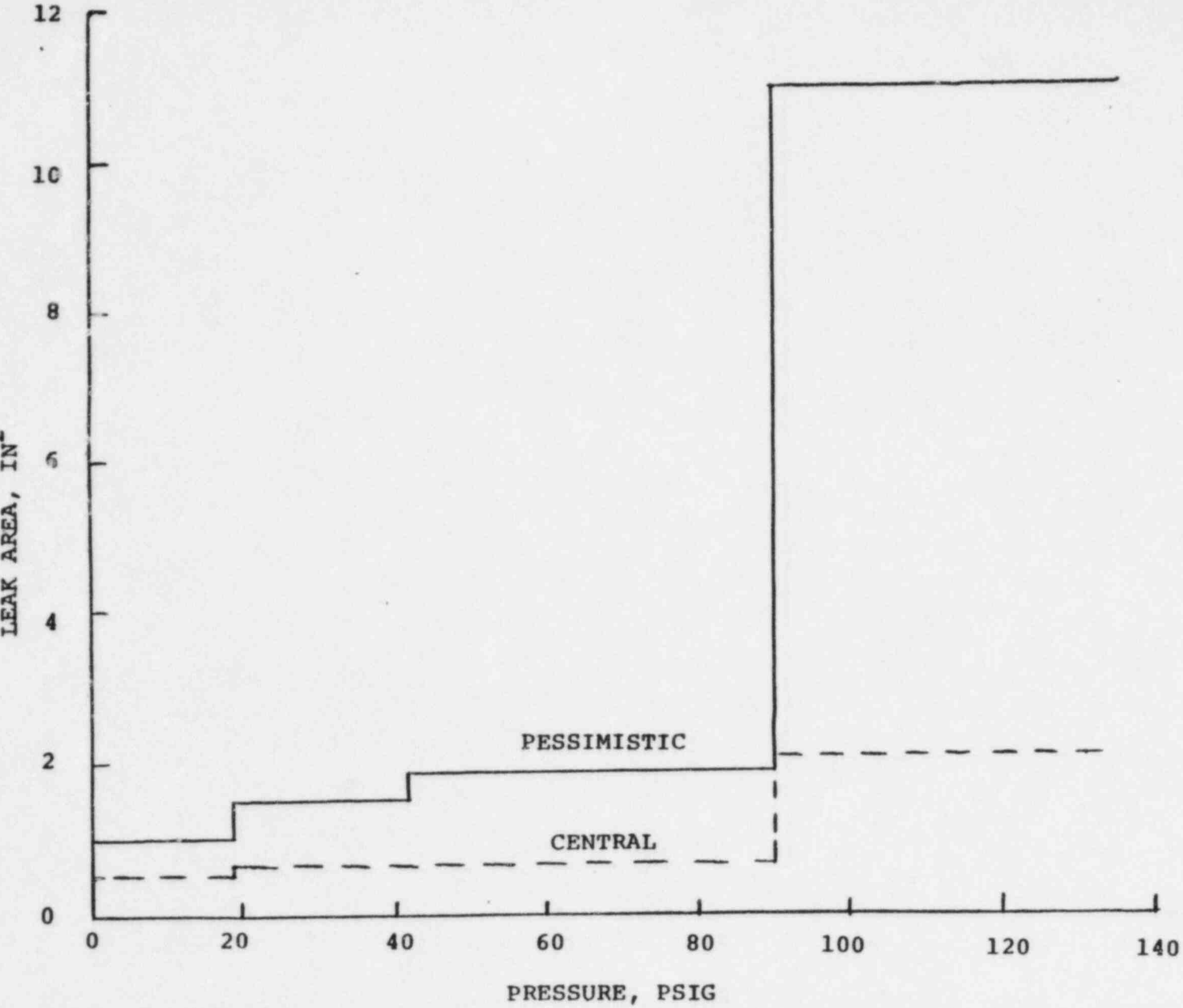

FIGURE 2.2. CPWG Pressure Dependent Leak Model for Zion.

AUG 1 5 1984

Sandia National Laboratories

Albuquerque, New Mexico 87185

date: August 9, 1984

to: Distribution

A S Benjamin

from: A. S. Benjamin, 6411

subject: Risk Perspective for NUREG-0956

Last April, Bob Bernero asked the Severe Accident Risk Reduction Program (SARRP) to apply its resources toward development of a risk perspective for NUREG-0956, the final report of the Accident Source Term Project Office (ASTPO). The primary objectives are as follows:

(1) For each of the six reference plants treated by ASTPO, identify the accident pathways (i.e., the combinations of accident sequences and containment events) that are important to risk.

(2) Estimate the frequencies of those accident pathways, utilizing to the maximum extent the results of the Containment Loads Working Group, Containment Performance Working Group, Accident Sequence Evaluation Program, and other NRC and Industry Studies.

(3) Ascertain how well the BMI-2104 source term calculations cover the risk-significant accident pathways, and identify pathways for which additional source term calculations are needed.

(4) Provide a letter report suitable as an appendix to NUREG-0956.

The study will be accomplished in two iterations. The first, completed July 31, 1984, provided preliminary estimates for four of the reference plants -- Surry, Zion, Peach Bottom, and Grand Gulf. The second, to be completed by November 30, 1984, will complete the analyses for these four plants and will cover the two remaining reference plants, Sequoyah and Limerick.

The attached draft document entitled "Containment Event Analysis and Estimation of Source Term Frequencies," fulfills our obligation for the July 31 iteration. It is the result of an intensive effort by members of the SARRP team conducted over

841813

a period of about 2 months. We have sent an advance copy to Walt Pasadag at NRC, at his request, and are providing copies to those within Sandia who are interested in this work. We would appreciate comments that will help us to optimize our product for the second iteration.

ASB:6411:cgt
Attachment

Distribution:
w/attachments

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