



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

MAY 21 1985

MEMORANDUM FOR: Hugh L. Thompson, Jr., Director  
Division of Licensing

Themis P. Speis, Director  
Division of Safety Technology

James P. Knight, Acting Director  
Division of Engineering

FROM: Robert M. Bernero, Director  
Division of Systems Integration

SUBJECT: PRELIMINARY REVIEW OF THE CONTAINMENT RESPONSE ANALYSES  
IN THE SHOREHAM PRA

We have completed our preliminary review of the containment response analyses in the Shoreham PRA. The Shoreham PRA study was performed by Science Applications Inc. (SAI). The Reactor Systems Branch, with BNL technical assistance, has reviewed the containment failure and radionuclide release analysis. The results of the BNL preliminary review are enclosed. The Reliability And Risk Assessment Branch (RRAB) with technical assistance from BNL has reviewed the internal events analysis to core melt. Their preliminary results were submitted to the Division of Licensing on December 31, 1984.

It may be noted that the BNL contract for the "back end" review is a limited one. BNL concentrated their review on comparisons to Limerick since BNL has gained extensive experience in the previous review. The BNL review (Enclosure 1) has thus concentrated on areas where there are analytical differences between the two PRAs or differences in containment design. The highlights of the "back end" review results to date are the following:

1. The dominant difference between Limerick and Shoreham estimated core melt frequencies is in the Class IV (see Enclosure 1 for definitions) ATWS with Shoreham being two orders of magnitude higher than Limerick. This is a risk dominant sequence because containment is predicted to fail prior to core melt. Much of the difference in ATWS frequency is due to the lack of an automatic, multiple train liquid poison injection system in Shoreham. The BNL estimate of the core melt frequency due to ATWS events is  $6.10^{-5}$ /reactor year. The BNL review indicates that ATWS events contribute about 40% to the total frequency. RSB believes that increasing the liquid poison injection rate to conform to the ATWS rule requirements would reduce the ATWS contribution to the total core melt frequency.

CONTACT: G. Thomas, RSB  
x29445

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2. The release fractions for Shoreham are several orders of magnitude lower than both WASH-1400 and the Limerick PRA. Most of this difference can be attributed to high pool Decontamination Factors (DFs) assumed by LILCO. BNL used SPARC to assess the potential for lower decontamination factors for a range of conditions.

The latest BNL decontamination factors are included as enclosure 2. These are time-averaged pool scrubbing factors based on SPARC (NUREG/CR-3317) and do not take credit for in-vessel retention and primary containment hold-up. Note that for the Class-I sequences the suppression pool is subcooled and the BNL decontamination factors are high and in substantial agreement with the Shoreham PRA results. However, for Class-IV sequences, the pool is heated to saturation before core melt and the BNL decontamination factors are much lower (600/100 vs 50/22). For the Class-IV ATWS with failure at the basemat ("y"), it is assumed that the pool is relocated to the annular region of the reactor building which surrounds the primary containment. Thus, the in-vessel release through the SRV's see the same scrubbing as the ex-vessel release through the vent pipes. The slight difference in the decontamination factors (9 compared to 14) for the two releases depend on gas blowing rates and carrier gas composition at the time the scrubbing occurs.

3. The Shoreham PRA is ambiguous on the possible failure location for containment overpressure failure. The location of failure is a key element for assessing the potential pool decontamination factors. The BNL technical assistance contract has been revised to provide an independent assessment of:
  - (1) The containment failure location and the containment failure modes.
  - (2) The pool thermal/hydraulic response, including possible water relocation for failures in the wetwell region.
  - (3) The pool thermal/hydraulic communication between the pedestal region and the outer annular region during debris/water interaction.
  - (4) The likelihood of secondary containment failure and the potential for pool bypass given secondary containment failure.

This work is expected to be completed by July 1985.

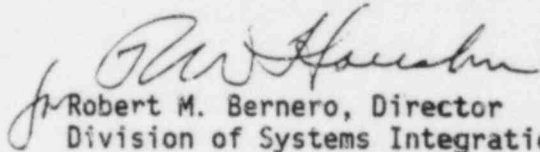
4. BNL is reviewing the impact on risk of the new source term methodology being developed by the NRC Accident Source Term Program Office (ASTPO). The reports issued by BCL (BMI-2104) presently do not assess the release rates for Mark-II containments, but the accident types and containment performance are similar to the Mark-I results. In addition, BCL is currently performing release estimates for selected sequences in a Mark-II geometry. BNL will evaluate the impact of these results on the Shoreham PRA. We note that the draft report from the American Physical Society (APS) emphasizes the importance of containment performance and has taken the position that much of the source term reduction has yet to be validated.

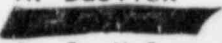
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5. In the preliminary review, BNL concurs with the Shoreham PRA release rate estimates for Class I and Class III accidents with early core melt and delayed containment failure. For these classes of accidents, the suppression pool remains subcooled and effectively reduces the source term. For the Class II loss-of-heat-sink transients (TW) the BNL audit calculations using SPARC indicate reduced decontamination factors compared to the Shoreham PRA. The combination of high frequency ( $1.2 \times 10^{-5}$ /ry) and relatively high release rates may result in a significant fraction of the risk for TW sequences in Shoreham. Note that the unusual geometry for the Shoreham containment (suppression pool vent pipes below the reactor vessel) negate the APS concerns for sustained core/concrete interaction and resultant vaporization release for these core melt classes.
6. Risk estimates for Class IV and Class V core melt sequences have the largest uncertainty. The preliminary BNL review indicates that the Class IV ATWS will have a much higher frequency (Item 1) and a lower decontamination factor (Item 2) than the estimates in the Shoreham PRA. The frequency of the Class V containment bypass sequences have been increased in the BNL review (to about  $4 \times 10^{-7}$ /ry) and may contribute significantly to the Shoreham risk profile.

We request that all feedback and comments from all NRR divisions and LILCO be forwarded to us by June 1, 1985, to allow us sufficient time for consideration in our final evaluation.

Enclosure 1 contains the preliminary report from BNL.

  
for Robert M. Bernero, Director  
Division of Systems Integration

cc: w/o enclosure  
H. Denton  
D. Eisenhower  
A. Thadani  
A. Buslick  
  
L. G. Hulman  
R. Caruso  
E. Chow  
F. Eltawila  
M. Wohl

BROOKHAVEN NATIONAL LABORATORY  
MEMORANDUM

DATE: September 27, 1984

TO: W. T. Pratt *WTP* *SYH*

FROM: K. R. Perkins and S. Y. Hsieh

SUBJECT: Preliminary Review of the Containment Response Analyses  
in the Shoreham PRA

INTRODUCTION

We have conducted a preliminary review of the containment response analyses contained in the Shoreham Probabilistic Risk Assessment.<sup>1</sup> A parallel effort sponsored by RRAB/DST/NRC is under way at BNL to review the event tree development and quantification. This "front end" evaluation is a much more extensive review than the present review and it has provided many valuable insights. We have concentrated our review on comparisons to Limerick since BNL has gained extensive experience in the previous review,<sup>2,3</sup> and the plants are very similar. Our review has thus concentrated on areas where there are analytical differences between the two PRAs or containment design differences. The degraded core frequencies for the four accident classes are shown in Fig. 1. The definitions used in the Shoreham classification scheme are included as Attachment 1. The dominant differences between these two estimated frequencies is in the Class IV ATWS with Shoreham being two orders of magnitude higher than Limerick. Class II loss of containment heat removal is also estimated to be higher in Shoreham than Limerick. In order to keep this comparison in perspective, a comparison of the results for all the available BWR PRAs is shown in Fig. 2. Note that the Limerick PRA gives substantially lower core melt frequencies than any of the other PRAs. However, methodological differences make direct comparison between the various PRAs difficult. The Limerick PRA used the basic approach and techniques of the Reactor Safety Study (WASH-1400)<sup>4</sup> but accounted for plant specific design differences between Limerick (BWR4 with a Mark-II containment) and the WASH-1400 plant (BWR4 with a Mark-I containment). The Shoreham PRA methodology is compared to WASH-1400 in Table 1.

The high frequency of ATWS events in Shoreham is of particular concern because of the potential for severe releases. Much of the difference in ATWS frequency can be attributed to the lack of an automatic poison injection system and to differences in the ADS inhibit logic. In other respects the scram systems used in Shoreham and Limerick are quite similar.



Having noted that the ATWS core melt frequency in Shoreham is relatively high, it is interesting to note that the calculated radiological impact is only moderate as indicated in Table 2. This is basically because of the large decontamination factors (DFs) calculated for Shoreham. Because of the different classification schemes used in the PRAs, it is difficult to make direct comparisons of the DFs. However, in order to get some perspective on the high DFs claimed for Shoreham, the release fractions for a typical ATWS sequence are compared in Table 3 for the three plants. Both WASH-1400 and the Limerick PRA calculated severe releases for these rapid sequences, but Shoreham calculates releases two orders of magnitude lower. We noted above that the Limerick PRA used WASH-1400 methods so that one would expect the source terms predicted in the two studies to be similar for compatible failure modes. However, from an inspection of Table 1 (Item F), it is clear that the Shoreham PRA used more recent methods to determine the radionuclide source terms and most of the reduction is apparently due to higher pool DFs. It will be important to verify that these reduction factors can be achieved under all sequence conditions and failure modes.

#### DISCUSSION

Of the five Shoreham plant accident classes, the final set of risk contributing accident sequences are chosen based on the ranking of importance of the product of the end state probability and source reduction factors. Fourteen risk contributing release categories and two non-risk contributing categories are defined. Three of these sixteen categories are Class IV accident sequences. They are SNP-10 ( $C_4R_4T_1-\gamma$ ), SNP-11 ( $C_4R_4T_1-\gamma$ ), and SNP-12 ( $C_4R_4T_1-\gamma'$ ;  $C_4R_4T_1-\gamma''$ ) as described in Table 4. Note that SNP-12 consists of both  $\gamma'$  and  $\gamma''$  scenarios where the  $\gamma''$  scenario assumes the wetwell failure below the waterline at the basemat. The DFs were calculated for the fourteen categories as shown in Table 5. Among the fourteen categories, three are Class IV accident sequences. However, the  $\gamma''$  sequence was not included in this table. The pool scrubbing DFs for the various accident sequences are summarized in Table 6 with the implication that the DF for  $\gamma''$  sequence are at least as much as the values in this table. The high suppression pool DFs of the Shoreham plant are based on the assumption that the pool is intact, and all fission products go through the pool [with the exception of Class IV ( $C_4R_4T_1-\gamma$ ) sequence in which 10% pool bypass is assumed] before entering the containment. In order to evaluate its high DF claims, the containment structural design of Shoreham and Limerick were examined and compared. The following preliminary assessment can be made.

1. The diaphragm floor at elevation 62'8" of the Shoreham plant was not anchored to the containment wall as in the Limerick design. The Shoreham containment wall displacement will expand outwardly under pressure as shown in Fig. 3. Based on this free standing diaphragm floor design, the PRA suggests that the most likely leakage paths will occur at the junction of the diaphragm floor and less likely at the basemat. Under this

failure condition, the suppression pool integrity and high scrubbing efficiency will probably be maintained. However, when gross containment failure occurs, as will be the case in many Class IV accidents, the basement-containment wall juncture was judged by Stone and Webster as the most probable location to fail (Appendix M of Ref. 1). Under such failure conditions (generally defined as the  $\gamma$ " scenario), the suppression pool water will be blown out into the surrounding chambers.

2. The suppression pool of both plants is surrounded by chambers; while the Limerick surrounding chamber is partitioned, the Shoreham surrounding chamber is a continuous annular-like space. Both surrounding chambers have drains. Limerick's PRA assumes drainage of the suppression pool in  $\gamma$ " sequence. Likewise, it is reasonable to assume that the Shoreham suppression pool will also be drained under such failure conditions ( $\gamma$ " sequence). If such is the case, the DF of Class IV  $\gamma$ " sequence should be evaluated explicitly. At present, the Shoreham PRA does not include the  $\gamma$ " scenario in any of the sixteen release calculations. Instead they are "binned" with the  $\gamma'$  sequences where the pool remains intact.
3. After the bottom head failure, the Shoreham PRA predicts that 90% of the core debris will flow to the suppression pool via the four downcomers underneath the vessel in the CRD room. The remaining 10% of the core debris will attack the concrete floor of the CRD room. Because of the limited amount of molten corium, the core-concrete interaction does not generate a substantial amount of gases to threaten the containment integrity.

The estimate of 90% of the core debris flowing into the pool is probably a very good estimate if the molten core debris can be treated as non-viscous incompressible fluid (as modeled in Appendix L of Ref. 1), since the remaining molten core on the concrete floor cannot be more than 1/2" deep before it spills over the downcomer's neck and flows into the pool. However, there is a wide range of possible debris conditions at the time of vessel failure.<sup>4</sup> Generally the high temperature molten debris (~4300F) is taken to be the limiting case. For Shoreham, however, the low temperature solid debris (~2700F) may be the worst case since very little debris would flow through the downcomers. Thus the effect of more than 10% of molten core remaining on the concrete floor should be addressed.

The revised geometry (see Fig. 4) of the downcomer vent pipes is intended to maximize corium flow into the pool but this also increases the potential for steam spikes and oxidation release.

1. The revised reactor pedestal geometry is not described adequately in the schematic (Fig. 4). Verify that the vent pipes and manways remain unblocked in the revised pedestal geometry.
2. Provide the estimate of the fraction of molten corium which is expected to spread out of the pedestal area through the open manways and vent pipes.
3. Verify that the downcomer vent pipes only protrude 1/2" above the diaphragm floor of the drywell as indicated in Fig. 4.
4. Section 3.6 of the PRA takes credit for containment leakage which will prevent gross containment failure for all pressurization rates except the very rapid pressurization associated with large breaks. However, the structural analysis by Stone and Webster (Appendix M) did not identify any significant source of leakage. The expected leakage source and the leakage rate as a function of pressure should be provided.
5. The basis for the partitioning between release category 10 and 11 (no pool bypass vs. partial pool bypass) should be provided. The phenomenological basis for the estimate of only 10% bypass should be provided. Preliminary results from IDCORE indicate that for some BWR sequences the vessel will fail with only 20% of the core molten. Presumably 80% of the melt release would bypass the SRV's and be released into the drywell.
6. The basis for the binning into release categories is poorly described and the transfer from Tables H.4-8 etc. (Attachment 2) into the 16 release categories is inscrutable. A table listing the specific sequences which are binned into each category should be provided.
7. The lack of  $R_5$  sequences in the release categories makes it apparent that these releases have been binned "downward" into the lesser release category  $R_4$ . The basis for this "downward" binning and any other sequences that are moved to less severe categories should be provided.
8. Table H.4-25 appears to be incomplete in that it does not include sequences D6 and D8. The completed table should be provided.
9. The source escape fractions used for end state screening (Table 3.6-10) appears to be quite arbitrary yet it greatly influences the importance ranking. In particular: the use of Z as the surrogate for melt release ignores the fact that there are noble gases in the melt release which will not be scrubbed at all; the use of a large scrubbing factor (500) for  $C_4$  transients is inappropriate since most of the melt release will be released directly to a failed containment; the reduction factor of 0.01 for  $\gamma$  failures is indefensible since the event tree precludes everything but large ruptures where the pool will be blown out into the reactor

building at high pressures. Taken in conjunction with the scrubbing factor of .002 the reduction factor of 0.01 implies double scrubbing with a decontamination factor of 50000 for an event in which the final level of water in the suppression pool is highly uncertain.

Table 3.6-10 should be replaced by a table with defensible reduction factors. As a minimum the table should include a separate category for  $C_u$  transients, which recognizes the defined sequence of events (containment failure before core melt). In addition, a detailed justification for each reduction factor should be provided along with the numerical results of the ranking process.

10. Sheet 1 of Figure H.4.2 has been reduced so that it is illegible. A full-size legible copy should be provided.

#### RECOMMENDATIONS

In addition to the above information, we feel that there are several areas which are important enough to warrant independent verification. The basis for our concern and the proposed resolution for each item is outlined below:

1. Core debris disposition: The partitioning of 90% of the core debris into the wetwell is highly speculative and assumes that debris will be nearly inviscid. In fact the molten core may be very viscous and may be solidified by quenching in the lower head of the vessel or on the drywell floor.

We propose to run a Class I accident sequence (e.g. TQUV) with 50% of the debris retained on the drywell floor in order to examine the potential for early release of fission products for this class of events.

2. The Shoreham PRA presents no quantitative analysis to preclude failure of the wetwell below the waterline. In fact, Appendix M indicates that the most likely failure location is at the bottom of the wetwell. A failure in this region would force the pool into the annular region of the reactor building surrounding the primary containment. If the reactor building does not fail and the drains are not on, the pool may still tend to mitigate releases from the containment as they are bubbled through the failure location into the reactor building.

We propose to use SPARC to address the DFs for the  $\gamma$  configuration assuming the pool is retained in the annular area surrounding the containment. We will also assess the significance of the assumption of no pool DF for this sequence (as assumed in the Limerick PRA). The partitioning between  $\gamma$  and non- $\gamma$  scenarios will be based on the applicant's response to questions 1 and 2 and the structural analysis of Appendix M. The



partitioning between scrubbing and no scrubbing for the  $\gamma$  scenario will be based upon our assessment of the possible pool configurations after a large rupture at the basemat at high containment pressure.

3. The release fractions for Shoreham are several orders of magnitude lower than both WASH-1400 and Limerick. Most of this difference can be attributed to high pool DFs based on limited experimental data as cited in Appendix N.

We propose to use SPARC to assess the potential for lower decontamination factors for a range of conditions. The calculations will emphasize vaporization release to a saturated pool since our previous experience indicates the potential for lower DFs under these conditions.

4. The issue of the energetics associated with steam explosions remains unresolved, but the issue is being addressed by the NRC as part of the Containment Loads Working Group (CLWG) effort. We propose to review the preliminary results of the CLWG effort to ensure that the NRC position is consistent with the low probability of containment failure ( $4 \times 10^{-4}$ ) or the low probability of an oxidation release given containment failure ( $6 \times 10^{-3}$ ) that is cited in Appendix L of the Shoreham PRA.

#### REFERENCES

1. Science Applications, Inc., Probabilistic Risk Assessment, Shoreham Nuclear Power Station, SAI-372-83-PA-01, June 1983.
2. H. Ludewig, J.W. Yang, and W.T. Pratt, "Containment Failure Mode and Fission Product Release Analysis for the Limerick Generating Station: Base Case Assessment," BNL Informal Report, BNL-NUREG-33835, April 1984.
3. I.A. Papazoglou, et al., "A Review of the Limerick Generating Station Probabilistic Risk Assessment," Brookhaven National Laboratory, NUREG/CR-3028, February 1983.
4. "Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014.
5. K.R. Perkins, G.A. Greene, and W.T. Pratt, "Appendix D-Standard Problem 4 (BWR Mark I)," Appendix D of the Containment Loads Working Group Standard Problem Results, to be published.

KRP:sm/tr

cc: R. A. Bari  
G. A. Greene  
D. Ilberg  
H. Ludewig  
K. Shiu  
J. W. Yang  
R. Youngblood  
W. S. Yu

Table 1

Major Changes in the Shoreham PRA Compared  
to the WASH-1400 Methodology

- a. New sequence initiators are defined and accident sequence models developed, including time phase event trees.
- b. The definition of generic accident release categories in WASH-1400 required lumping accident sequences with major differences in potential consequences and containment interactions into the same category for consequence evaluation. For the Shoreham evaluation, realistic and refined release categories are defined so that each unique sequence type could be evaluated separately assuring greater detail in defining the spectrum of radionuclide releases.
- c. Smoothing of probabilities among release categories was used in WASH-1400 to account for possible miscategorization of sequences and other uncertainties. This artifice is eliminated in the Shoreham evaluation because of the better definition of accident sequence release categories for consequence evaluation.
- d. Accident sequences are totally reevaluated using the latest BWR thermal hydraulic calculations for transients, LOCAs and ATWS.
- e. Component failure rate data and common mode failures are reevaluated based upon the latest data from operating nuclear plants.
- f. The radionuclide source term, release mechanisms, and removal mechanisms have been completely reevaluated to incorporate the latest experimental data and analytical methods in the characterization of source terms.
- g. The conservative estimates of the probability of the steam explosion leading directly to containment failure was reassessed. The steam explosion phenomenon leading directly to a containment failure and substantial oxidation release is realistically evaluated considering the specific Shoreham design. The probability of this event is reduced.
- h. The conservative assumption that all potential core damage sequences lead to a major release was reassessed. Detailed containment event trees were developed to appropriately characterize the accident sequences which could lead to a radionuclide release.

Table 2  
SUMMARY OF SIGNIFICANT RADIONUCLIDE INVENTORY  
RELEASE FRACTIONS BY RELEASE CATEGORY

RELEASE CATEGORY	ACCIDENT CLASSES CONTRIBUTING TO RELEASE CATEGORY	SIGNIFICANT ISOTOPE RELEASE FRACTION(a)		POTENTIAL RADIOLOGICAL IMPACT (b)	PRIORITY OF RELEASE
		Cs	Te		
1	I	$9.3 \times 10^{-5}$	$3.4 \times 10^{-3}$	Moderate	$2.5 \times 10^{-7}$
2	I	$3.3 \times 10^{-4}$	$9.1 \times 10^{-6}$	Minor	$2.8 \times 10^{-6}$
3	I	$1.5 \times 10^{-7}$	$1.4 \times 10^{-4}$	Minor	$2.3 \times 10^{-4}$
4	I	E	$1.2 \times 10^{-7}$	Minor	$1.9 \times 10^{-4}$
5	II	$1.5 \times 10^{-4}$	$2.5 \times 10^{-2}$	Moderate	$4.3 \times 10^{-7}$
6	II	$1.3 \times 10^{-4}$	$1.1 \times 10^{-5}$	Minor	$1.2 \times 10^{-4}$
7	III	$1.4 \times 10^{-1}$	$3.8 \times 10^{-2}$	Severe	$1.7 \times 10^{-7}$
8	III	$2.8 \times 10^{-5}$	$1.2 \times 10^{-4}$	Minor	$4.8 \times 10^{-7}$
9	III	$3.1 \times 10^{-7}$	$1.2 \times 10^{-4}$	Minor	$4.3 \times 10^{-7}$
10	IV, I, II	$2.3 \times 10^{-2}$	$2.4 \times 10^{-2}$	Moderate	$2.3 \times 10^{-6}$
11	IV	$1.6 \times 10^{-4}$	$2.3 \times 10^{-4}$	Moderate	$1.8 \times 10^{-6}$
12	IV	$1.7 \times 10^{-4}$	$2.9 \times 10^{-4}$	Minor	$4.3 \times 10^{-6}$
13	V	$1.3 \times 10^{-1}$	$3.0 \times 10^{-3}$	Severe	$2.5 \times 10^{-6}$
14	V	$2.5 \times 10^{-1}$	$3.0 \times 10^{-3}$	Severe	$1.2 \times 10^{-6}$
15	I, II, III, IV	E	$1.2 \times 10^{-4}$	Negligible	$4.3 \times 10^{-6}$
16	I, II, III, IV	E	E	Negligible	$1.9 \times 10^{-6}$

- (a) Release fractions based on core inventory; Cs and Te were chosen as a measure because these provide an estimate of the containment retention capability for fission product releases in-vessel and ex-vessel, respectively. The containment retention capability also provides a quantitative measure of the effectiveness of the suppression pool for the transient sequences.
- (b) The measure of potential impact is a relative measure of radiological doses to an exposed individual.
- Severe means that the noble gases and substantial fractions of the volatiles and actinides are released.
  - Moderate means that the noble gases and small fractions of the volatiles and actinides are released.
  - Minor means that primarily noble gases are released.
  - Negligible means that very small fractions of all radionuclide species are slowly leaked.
- (c) The release fractions for this radionuclide species are less than  $10^{-7}$  and are not presented here.

TABLE 3  
Comparison of Release Fractions for  
ATWS Sequences

Shoreham Release Category	Iodine or Cesium			Tellurium		
	WASH-1400	Limerick	Shoreham	WASH-1400	Limerick	Shoreham
11	.1 <sup>1</sup>	.07-.73 <sup>2</sup>	$1.6 \times 10^{-4}$	.3	.1-.76 <sup>2</sup>	$2.5 \times 10^{-2}$

<sup>1</sup>Based on comparable BWR release assuming elemental iodine.

<sup>2</sup>Depending on failure mode.



Table 4

## DESCRIPTION OF THE RELEASE CATEGORIES IDENTIFIED FOR THE SHOREHAM PRA (Sheet 1 of 4)

RELEASE CATEGORY	GENERAL DESCRIPTION	DOMINANT ACCIDENT SEQUENCE CONTRIBUTION BASIS FOR IN-PLANT ANALYSIS	EET PROGRESSION PATH DESIGNATOR
SNPS-1	This release category is representative of Class I accident sequences involving a transient event leading to core meltdown where the containment fails to isolate or fails by overpressure early in the accident sequence leading to a leakage type release from the drywell.	Loss of Off-site Power Initiator, failure to recover Division I or II electric power, failure of high pressure injection systems, failure of ADS, failure to isolate in the drywell.	$C_{1R}T_1 - \delta$ Early Failure in the Drywell
SNPS-2	This release category is representative of Class I accident sequences involving a transient event where the containment fails to isolate or fails by overpressure early, leading to a leakage type release from the wetwell.	RB flood which partially drains suppression pool, failure of high and low pressure injection systems, failure to isolate in the wetwell.	$C_{1R}T_1 - \delta$ Early Failure in the Wetwell
SNPS-3	This release category involves a core meltdown for a Class I accident sequence in which the containment fails in the long term leading to a leakage path from the drywell. The long time to containment failure is expected to reduce the airborne fission products in containment substantially prior to release.	Loss of condenser vacuum, failure of high pressure injection systems, failure of ADS, the containment is intact during the significant fission product release periods.	$C_{1R}T_4 - \gamma$ Late Failure in the Drywell
SNPS-4	This release category involves a core meltdown for a Class I accident sequence in which the containment fails in the long term leading to a leakage path from the wetwell. The long time to containment failure is expected to reduce the airborne fission products in containment substantially prior to release.	Loss of condenser vacuum, failure of high pressure injection systems, failure of ADS, the containment is intact during the significant fission product release periods.	$C_{1R}T_4 - \gamma'$ $C_{1R}T_4 - \gamma''$ Late Failure in the Wetwell

Table 4

## DESCRIPTION OF THE RELEASE CATEGORIES IDENTIFIED FOR THE SHOREHAM PRA (Sheet 2 of 4)

RELEASE CATEGORY	GENERAL DESCRIPTION	DOMINANT ACCIDENT SEQUENCE CONTRIBUTION BASIS FOR IN-PLANT ANALYSIS	CEI PROGRESSION PATH DESIGNATOR
SNPS-5	This category involves Class II accident sequences in which the containment fails due to loss of decay heat removal capability followed by core meltdown. The containment failure is assumed to occur in the drywell and the fission products airborne in containment would not be significantly reduced.	Loss of condenser vacuum; failures of RHC, RHCSC, and PCS; failure of coolant injection following containment failure; the suppression pool is effective in retaining the fission products released from the core region in-vessel.	$C_{2R}V_1 - \gamma$ Failure in the Drywell
SNPS-6	This category involves Class II accident sequences in which the containment fails due to loss of decay heat removal capability followed by core meltdown. The containment failure is assumed to occur in the wetwell and the fission products airborne in containment would not be significantly reduced.	Loss of condenser vacuum; failures of RHC, RHCSC, and PCS; failure of coolant injection following containment failure; the suppression pool is effective in retaining the fission products released from the core region in-vessel.	$C_{2R}V_1 - \gamma$ $C_{2R}V_1 - \gamma$ Failure in the Drywell
SNPS-7	This release category is representative of a Class III accident sequence in which the containment fails early in the accident due to inadequate pressure suppression capability. The fission products released from the core region are discharged directly to the drywell atmosphere and are not significantly attenuated prior to leakage from the drywell. This category includes Large LOCA and RPV failure accident sequences, which challenge containment integrity early in the sequence.	Large LOCA, failure of vapor suppression, early overpressure failure of containment.	$C_{3R}V_1 - \gamma$ Early failure in the Drywell
SNPS-8	This release category involves a Class III accident sequence in which the containment fails in the long term leading to a leakage path from the drywell. The long time to containment failure is expected to reduce the airborne radionuclide material inventory in containment prior to its leakage to the environment.	Medium LOCA, failure of high and low pressure injection systems, the containment is intact during the radionuclide release period from the fuel.	$C_{3R}V_4 - \gamma$ Late failure in the Drywell

Table 4

## DESCRIPTION OF THE RELEASE CATEGORIES IDENTIFIED FOR THE SHOREHAM PRA (Sheet 3 of 4)

RELEASE CATEGORY	GENERAL DESCRIPTION	DOMINANT ACCIDENT SEQUENCE CONTRIBUTION BASIS FOR IN-PLANT ANALYSIS	CEP PROGRESSION PATH DESIGNATION
SNPS-9	This release category involves a Class III accident sequence in which the containment fails in the long term leading to a leakage path from the wetwell. The long time to containment failure is expected to reduce the airborne radionuclide material inventory in containment prior to its leakage to the environment.	Medium LOCA, failure of high and low pressure injection systems, the containment is intact during the radionuclide release period from the fuel.	$C_{341}T_4 - Y^+$ $C_{341}T_4 - Y^-$ Late Failure in the Wetwell
SNPS-10	This release category is representative of transient events involving Class I, Class II, and Class IV accident sequences where the suppression pool is partially bypassed and the containment integrity is lost early in the accident sequence.	MSIV Closure ATWS, failure of SLC, failure of all injection systems following containment failure, 10% of the fission products released from the fuel in-vessel is transported directly to the drywell bypassing the suppression pool.	$C_{441}T_1 - Y$ Early Failure in the Drywell
SNPS-11	This release category involves a Class IV accident sequence in which the containment fails by a failure to scram and remove decay heat, followed by core meltdown. The containment failure is assumed to occur in the drywell and the fission products are not significantly attenuated prior to its leakage to the environment.	MSIV Closure ATWS, failure of SLC, failure of all injection systems following containment failure, the fission products released from the fuel in-vessel are totally transported to the containment through the suppression pool.	$C_{441}T_1 - Y$ Early Failure in the Drywell
SNPS-12	This release category involves a Class IV accident sequence in which the containment fails by a failure to scram and remove decay heat, followed by core meltdown. The containment failure is assumed to occur in the wetwell and the fission products are significantly attenuated prior to its leakage to the environment.	MSIV Closure ATWS, failure of SLC, failure of all injection systems following containment failure, the fission products released from the fuel in-vessel are totally transported to the containment through the suppression pool.	$C_{441}T_1 - Y^+$ $C_{441}T_1 - Y^-$ Early Failure on the Wetwell

Table 4

## DESCRIPTION OF THE RELEASE CATEGORIES IDENTIFIED FOR THE SHOREHAM PRA (Sheet 4 of 4)

RELEASE CATEGORY	GENERAL DESCRIPTION	DOMINANT ACCIDENT SEQUENCE CONTRIBUTION BASIS FOR IN-PLANT ANALYSIS	CEP PROGRESSION PATH DESIGNATOR
SNPS-13	This release category is representative of Class V accident sequences which involve core meltdown following a LOCA outside containment. The SRVs are actuated in order to mitigate the release of fission products to the environment by providing an alternative path into the containment (i.e., suppression pool) during the in-vessel release period.	Interfacing LOCA, the suppression pool is partially effective in mitigating releases.	C <sub>5</sub> D <sub>4</sub> T <sub>1</sub> A
SNPS-14	This release category is representative of Class V accident sequence which involve core meltdown following a LOCA outside containment. The SRVs are assumed not to be opened, and the fission products released from the fuel totally bypass the containment.	Interfacing LOCA, failure of SRVs.	C <sub>5</sub> D <sub>4</sub> T <sub>1</sub> B
SNPS-15	This release category is representative of the terminated core meltdown accident sequences in which the containment remains intact and the release of radionuclides to the environment would be very small, and determined by leakage to the reactor building.	Loss of condenser vacuum, failure of high pressure injection systems, failure of ADS, containment integrity is maintained.	C <sub>1</sub> D <sub>5</sub> T <sub>4</sub> - E
SNPS-16	This release category is representative of the terminated core meltdown accident sequences in which the containment remains intact and the release of radionuclides to the environment would be very small, and determined by leakage to the reactor building. Furthermore, the releases would be filtered by the standby filter systems.	Loss of condenser vacuum, failure of high pressure injection systems, failure of ADS, containment integrity is maintained.	C <sub>1</sub> D <sub>5</sub> T <sub>4</sub> - E



Table 5  
SHOREHAM SUMMARY OF DEGRADED CORE ACCIDENT ANALYSIS CASES

SEQUENCE DESIGNATOR	CONTAINMENT FAILURE DESIGNATOR(a)	PRIMARY SYSTEM DEPOSITION(b)	SUPPRESSION POOL DECONTAMINATION FACTORS		CODES USED (c)
			SRV	VENTS	
C <sub>1</sub> R <sub>4</sub> T <sub>1</sub> - δ	CI-DW	T	600	100	M,C
C <sub>1</sub> R <sub>4</sub> T <sub>1</sub> - β	CI-WW	T	120(d)	20 (d)	M,C
C <sub>1</sub> R <sub>4</sub> T <sub>4</sub> - γ	OP-DW	T	600	100	M,C
C <sub>1</sub> R <sub>4</sub> T <sub>4</sub> - γ'	OP-WW	T	600	100	M,C
C <sub>2</sub> R <sub>4</sub> T <sub>1</sub> - γ	OP-DW	T	600	100	M,C
C <sub>2</sub> R <sub>4</sub> T <sub>1</sub> - γ'	OP-WW	T	600	100	M,C
C <sub>3</sub> R <sub>4</sub> T <sub>1</sub> - γ	OP-DW	L	-	100	M,C
C <sub>3</sub> R <sub>4</sub> T <sub>4</sub> - γ	OP-DW	L	-	100	M,C
C <sub>3</sub> R <sub>4</sub> T <sub>4</sub> - γ'	OP-WW	L	-	100	M,C
C <sub>4</sub> R <sub>4</sub> T <sub>1</sub> A-γ	OP-DW	T(e)	600	100	M,C
C <sub>4</sub> R <sub>4</sub> T <sub>1</sub> B-γ	OP-DW	T	600	100	M,C
C <sub>4</sub> R <sub>4</sub> T <sub>1</sub> - γ''	OP-WW	T	600	100	M,C
C <sub>5</sub> R <sub>4</sub> T <sub>1</sub> - A	BP	L(f)	2000	100	M,C,R
C <sub>5</sub> R <sub>4</sub> T <sub>1</sub> - B	BP	L	-	100	M,C,R

- (a) CI = Containment Isolation Failure  
 OP = Containment Overpressure Failure  
 DW = Containment Breach is in the Drywell (T)  
 WW = Containment Breach is in the Wetwell (T')
- (b) T = Transient event with primary system intact and an effective retention of 80%.  
 L = LOCA event with the effective primary system retention of 30 and 10% for the vapors and aerosols respectively.
- (c) M = MARCH  
 C = CORRAL  
 R = Contest module of RADAP-1
- (d) Pool scrubbing effectiveness is reduced due to the reduced submergence height.  
 (e) Ten percent of the fission products released from the core region within the primary system bypasses the pool.  
 (f) Fifty percent of the fission products released from the fuel is directed into the containment by opening all the SRV discharge lines during core heat up and meltdown.

Table 6.  
SHOREHAM POOL SCRUBBING DECONTAMINATION FACTORS

EVENT	PATHWAY	
	SRV	DOWNCOMERS
Class 1, 2, 4	3000	100
Class 3	NA	1000*, 100

Table 3.6-10

## ESTIMATED SOURCE ESCAPE FRACTIONS FOR USE IN CET END STATE SCREENING

Containment Event tree

GLORIC SOURCE TYPE	FUEL RELEASE	SOURCE FACTOR	RE-VESSSEL ESCAPE FUNCTION	POOL SCRAMBLING	CONTAINMENT FAILURE TIME PHASE				PRIMARY CONTAINMENT FAILURE MODES				REACTOR BUILDING MODALIZATION	RDSYS. FILTER EFFECTIVENESS
					11*	12	13	14	LEAKAGE	OVERPRESSURE FAILURE BREAK SIZE	CONTAINMENT BREAK LOCATION	WV*		
TRANSIENT:  $C_1, C_2, C_4$	Gap (1c)**	0.005	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
	Melt (1)	1.0	0.20	0.002	1.0	0.5	0.1	0.01	0.05	0.05	1.0	0.5	0.5	0.001
	Oxidation (1c-Ro)	0.03	-	-	-	-	0.5	0.01	0.05	0.05	1.0	0.5	0.5	0.001
	Dispersion (1c)	0.09	-	-	-	-	-	0.10	0.05	0.05	1.0	0.5	0.5	0.001
LOCA  $C_3, C_5$	Gap (1c)	0.005	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
	Melt (1)	1.0	0.90	-	1.0	0.5	0.1	0.01	0.05	0.05	1.0	0.5	0.5	0.001
	Oxidation (1c-Ro)	0.05	-	-	-	-	0.5	0.01	0.05	0.05	1.0	0.5	0.5	0.001
	Dispersion (1c)	0.09	-	-	-	-	-	0.10	0.05	0.05	1.0	0.5	0.5	0.001

\* If time phase includes failure to isolate the containment immediately following the initiating event.

\*\* Contingency measure is expressed in terms of the fission product species escape fraction.

\*\*\* The C<sub>5</sub> accident class would bypass containment during in-vessel meltdown. Therefore, the re-normal mechanisms associated with the containment for the gap and melt would not be applicable.

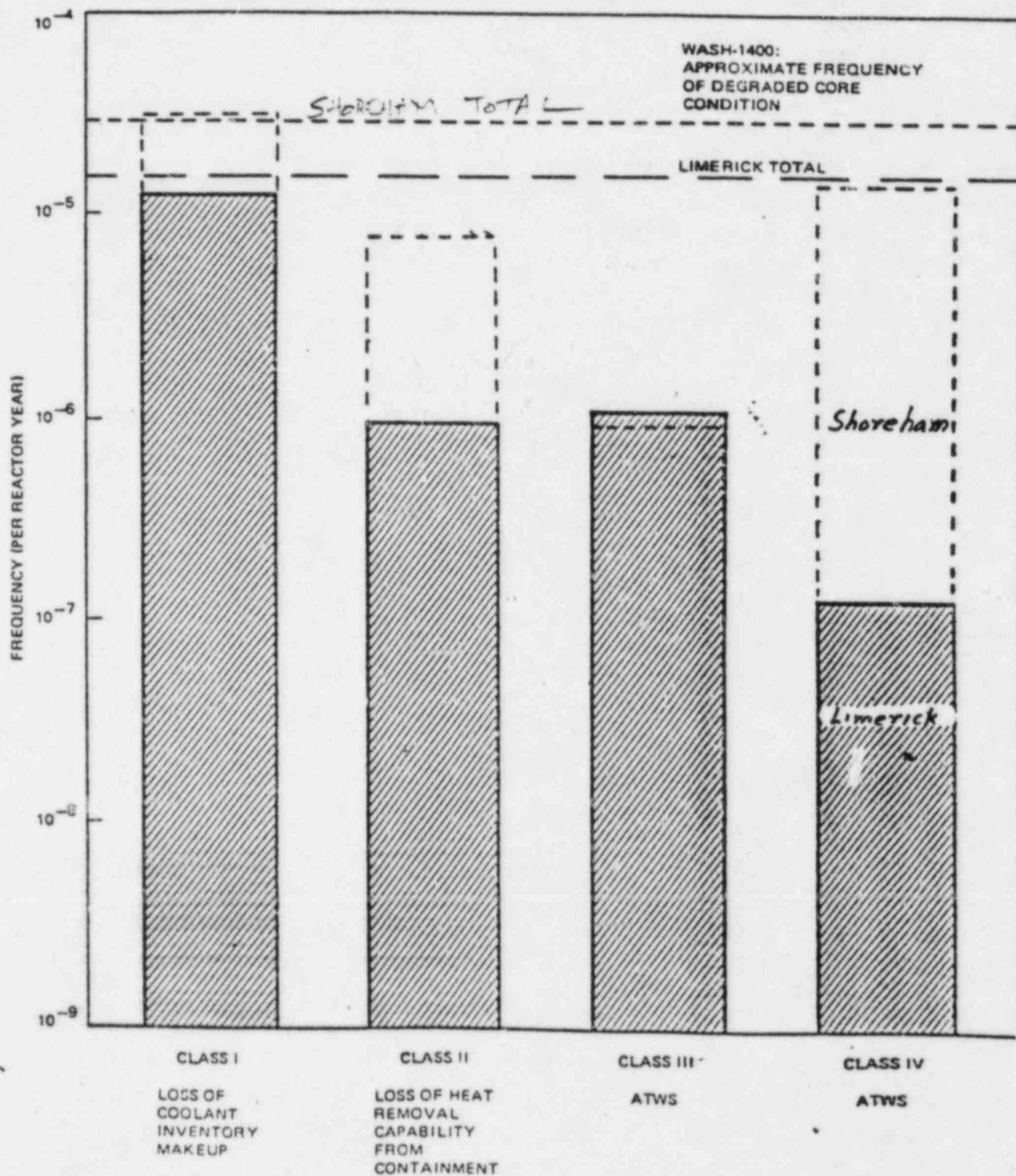


Figure 1 Summary of the accident sequence frequencies leading to degraded core conditions summed over all accident sequences within a class.



NOTE: In the other available BWR PRA's there is no clear distinction between core vulnerable and core melt end states.

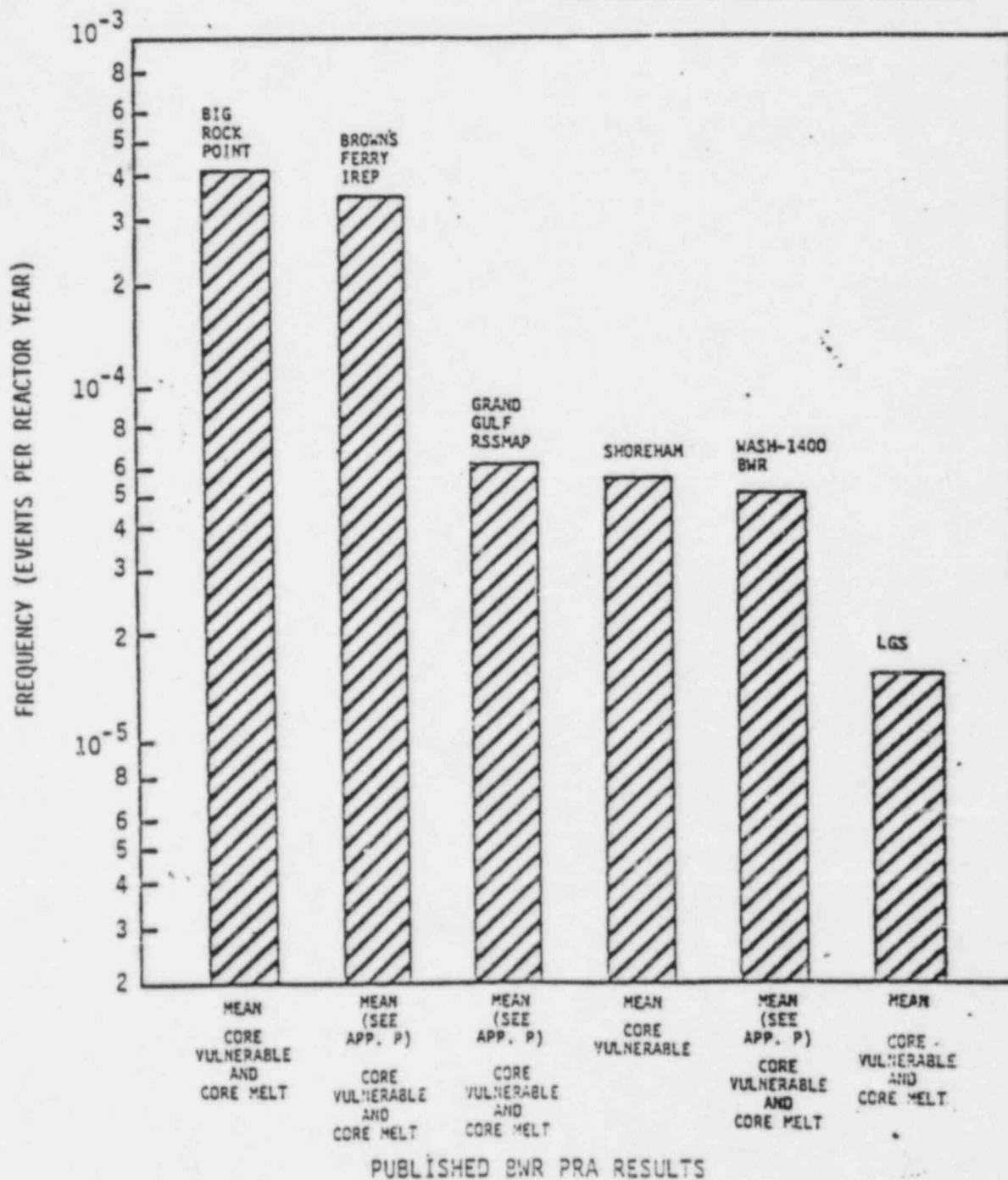


Figure 2 Comparison of Frequency of Core Vulnerable/Core Melt from Published BWR PRAs.

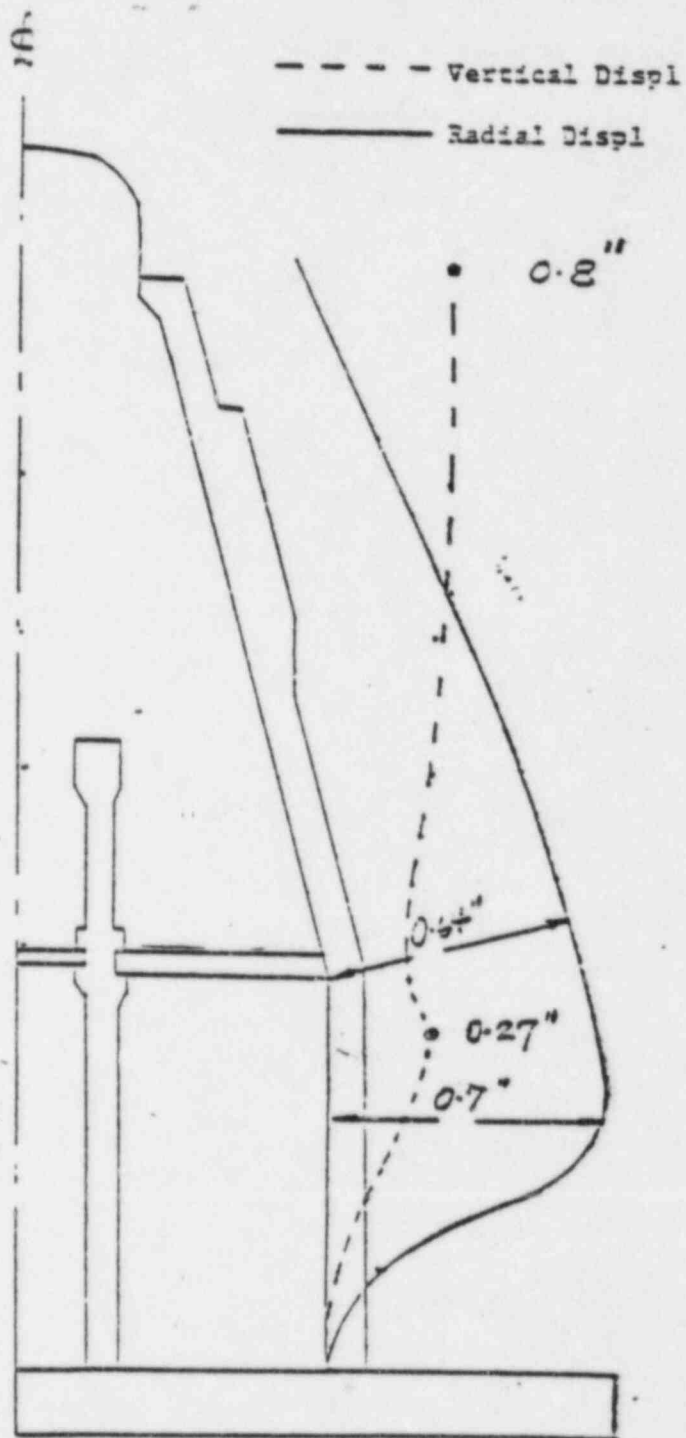


FIGURE 3 - GROWTH OF CONTAINMENT UNDER 120 PSI INTERNAL PRESSURE

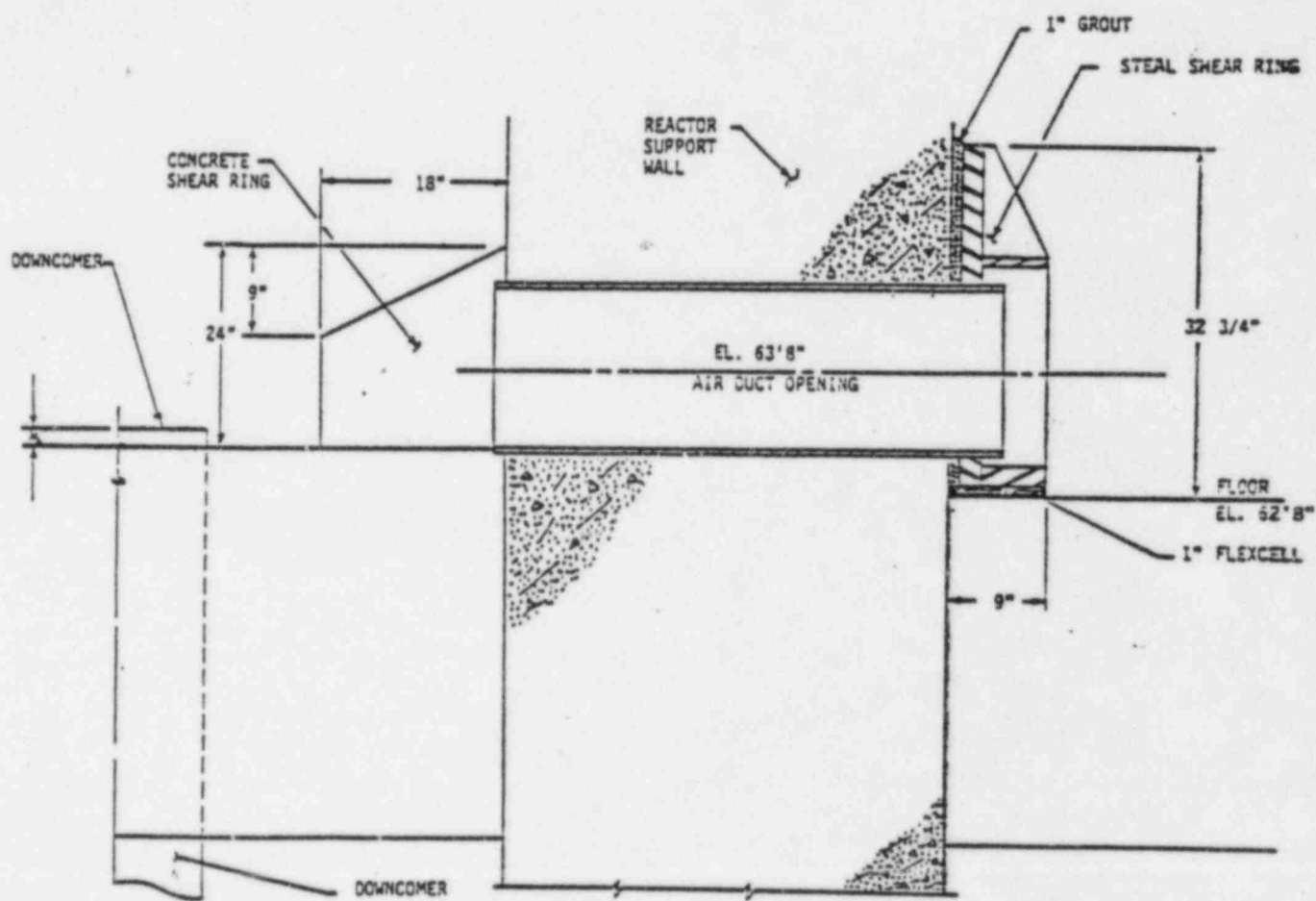


Figure 4 Reactor Pedestal Schematic Showing Reactor Downcomer Geometry

ATTACHMENT 1

Definitions Used in the Shoreham PRA Accident Sequence  
Classification

Table 3. 3-1  
ACCIDENT SEQUENCE CLASSIFICATIONS USED IN THE SYSTEMS EVALUATION

ACCIDENT SEQUENCE DESIGNATION	PHYSICAL BASIS FOR CLASSIFICATION	SYSTEM LEVEL CONTRIBUTING EVENT SEQUENCE	REPRESENTATIVE SEQUENCE FOR CLASS
Class I (C <sub>1</sub> )	Relatively rapid core melt; containment intact at core melt and at initially low pressure; involves a release pathway from the vessel to the suppression pool	Transients involving loss of inventory makeup; medium and small LOCA events involving SRV actuation with loss of inventory makeup; transients involving loss of scram function and inability to provide sufficient coolant makeup	Transient with loss of high and low pressure coolant makeup
Class II (C <sub>2</sub> )	Relatively slow core melt due to lower decay heat power; containment is failed prior to core melt; involves a release pathway from the vessel to the suppression pool	Transients or LOCAs involving loss of containment heat removal; inadvertent SRV opening accidents with inadequate heat removal capability	Transient with loss of residual heat removal
Class III (C <sub>3</sub> )	Relatively rapid core melt; containment intact at core melt, but at initially high internal pressure; involves a release from the vessel to the drywell	Large LOCA with insufficient coolant makeup; small and medium LOCAs with failure of the SRVs to actuate and long-term loss of inventory makeup; RPV failures with insufficient coolant makeup	Large LOCA with loss of low pressure ECCS
Class IV (C <sub>4</sub> )	Relatively rapid core melt; containment fails prior to core melt due to overpressure; involves a release pathway from the vessel to the suppression pool	Transients involving loss of scram function and loss of containment heat removal or all reactivity control; transients with loss of scram function followed by actuated depressurization	Transient with failure of NPS and failure of SICs
Class V (C <sub>5</sub> )	Relatively rapid core melt; containment failed from initiation of accident due to equipment failure; involves a release pathway from the vessel which bypasses the containment	LOCAs outside containment with insufficient coolant makeup to core; interfacing system LOCAs with insufficient coolant makeup	LOCA in main steam lines with failure of MSIV closure and loss of ECCS



Table 3.6-9  
SUMMARY OF CET CATEGORIZATION

DESIGNATOR	DESCRIPTION	ATTRIBUTE
	GENERIC ACCIDENT SEQUENCE CLASS	
C <sub>1</sub>	Class I	Pool scrubbing prior to transport in contain- ment
C <sub>2</sub>	Class II	Pool scrubbing prior to transport in contain- ment
C <sub>3</sub>	Class III	Pool scrubbing is by- passed prior to transport in contain- ment
C <sub>4</sub>	Class IV	Pool scrubbing prior to transport in contain- ment
C <sub>5</sub>	Class V	Pool scrubbing and containment are bypassed
	RELEASE TYPES	
R <sub>1</sub>	Gap Release	Recovered accident after initial core overheat
R <sub>2</sub> , R <sub>3</sub>	Gap and Melt plus oxidation release	Recovered accident after significant core melting
R <sub>4</sub> , R <sub>5</sub>	Gap, melt, and vaporization release plus oxidation release	Unrecovered meltdown accident

Table 3.6-9  
SUMMARY OF CET CATEGORIZATION

DESIGNATOR	DESCRIPTION	ATTRIBUTE
	GENERIC ACCIDENT SEQUENCE CLASS	
C <sub>1</sub>	Class I	Pool scrubbing prior to transport in contain- ment
C <sub>2</sub>	Class II	Pool scrubbing prior to transport in contain- ment
C <sub>3</sub>	Class III	Pool scrubbing is by- passed prior to transport in contain- ment
C <sub>4</sub>	Class IV	Pool scrubbing prior to transport in contain- ment
C <sub>5</sub>	Class V	Pool scrubbing and containment are bypassed
	RELEASE TYPES	
R <sub>1</sub>	Gap Release	Recovered accident after initial core overheating
R <sub>2</sub> , R <sub>3</sub>	Gap and Melt plus oxidation release	Recovered accident after significant core melting
R <sub>4</sub> , R <sub>5</sub>	Gap, melt, and vaporization release plus oxidation release	Unrecovered meltdown accident

Table 3.6-9 (Continued)  
SUMMARY OF CET CATEGORIZATION

DESIGNATOR	DESCRIPTION	ATTRIBUTE
	CONTAINMENT FAILURE TIME	
T <sub>1</sub>	Time phase T <sub>1</sub>	Containment is failed before core degradation
T <sub>2</sub>	Time phase T <sub>2</sub>	Containment fails during core meltdown in vessel
T <sub>3</sub>	Time phase T <sub>3</sub>	Containment fails during core-concrete interaction
T <sub>4</sub>	Time phase T <sub>4</sub>	Containment fails in the long term
	CONTAINMENT FAILURE MODE	
BP, CI	Containment Bypass or Isolation Failure	Containment isolation fails or containment is bypassed
δ	Leakage in the Dry- well	Leakage sufficient to preclude overpressure failure
θ	Leakage in the Wet- well	Leakage sufficient to preclude overpressure failure
OP	Overpressurization Failure	Containment over- pressure failure, small or large break
γ	Gross Containment Failure Location	Drywell
γ'	Gross Containment Failure Location	Wetwell airspace
γ''	Gross Containment Failure Location	Wetwell below the waterline

Table 3.6-1  
TYPES OF POTENTIAL RELEASE FROM FUEL

NR	No release
R <sub>1</sub>	Core heatup (gap)
R <sub>2</sub>	Core heatup and melt release (gap and melt)
R <sub>3</sub>	Core heatup and melt release with potential for oxidation release (gap, melt and oxidation)
R <sub>4</sub>	Core heatup and melt release, and vaporization release (gap, melt and vaporization)
R <sub>5</sub>	Core heatup and melt release, oxidation release, and vaporization release (gap, melt, oxidation and vaporization)

Table 3.6-2  
DISCRETE TIME PERIODS DEFINED TO MODEL THE VARYING EFFECTS OF CHANGES  
IN CONTAINMENT FAILURE TIMING

T <sub>1</sub>	From the time of accident initiation to initiation of core overheating
T <sub>2</sub>	From the time of initiation of core overheating until the time of pressure vessel failure
T <sub>3</sub>	From the time of vessel failure until soon after vessel failure or when core-concrete interaction occurs
T <sub>4</sub>	Long after vessel failure or vaporization release has occurred

ATTACHMENT 2

Conditional Probabilities for all Release Categories



Table II.4-8

## CONDITIONAL PROBABILITIES FOR CLASS I, PLANT DAMAGE STATE A RELEASE CATEGORIES

TIME/LOCATION RELEASE POTENTIAL	T <sub>1</sub>			T <sub>2</sub>			T <sub>3</sub>			T <sub>4</sub>			NO CONTAINMENT FAILURE	CONTAINMENT FAILURE (ISOLATION FAILURE)
	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*		
SMALL BREAKS	R1	1.1E-3	1.1E-3										0.10	3.4E-4
	R1*												-	-
	R1**												-	-
	R1***												-	-
	R2												4.2E-2	4.0E-5
	R2*												2.8E-2	2.6E-5
	R2**												-	9.8E-6
	R2***												-	6.5E-6
	R3												0.17	1.6E-4
	R3*												0.11	1.1E-4
	R3**												-	4.0E-5
	R3***												-	2.6E-5
	R4	1.1E-5	1.1E-5											4.0E-5
	R4*	7.4E-6	7.4E-6											2.6E-5
LARGE BREAKS	R4**	2.8E-6	2.8E-6											9.8E-6
	R4***	1.9E-6	1.9E-6											6.5E-6
	R5	4.4E-5	4.4E-5											1.6E-4
	R5*	3.0E-5	3.0E-5											1.1E-4
	R5**	1.1E-5	1.1E-5											4.0E-5
	R5***	7.4E-6	7.4E-6											2.6E-5
	R6													
	R6*													
	R6**													
	R6***													
	R7													
	R7*													
	R7**													
	R7***													

\* Partial pool bypass.  
 \*\* No secondary con-  
 tainment filtering.  
 \*\*\* Partial pool bypass,  
 no filtering.

Table II.4-9  
CONDITIONAL PROBABILITIES FOR CLASS I, PLANT DAMAGE STATE B RELEASE CATEGORIES

PIPE LOCATION RELEASE POSITION	I <sub>1</sub>			I <sub>2</sub>			I <sub>3</sub>			I <sub>4</sub>			ID CONTAINMENT FAILURE	CONTAINMENT BYPASS (1500 ATOM FAILURE)
	DU	MI	MI*	DU	MI	MI*	DU	MI	MI*	DU	MI	MI*		
SMALL BREAKS	01													0.50
	01*													-
	01**													-
	01***													-
	02													2.4E-2
	02*													1.6E-2
	02**													6.0E-3
	02***													2.0E-3
	03													9.6E-2
	03*													6.4E-2
	03**													2.4E-2
	03***													1.6E-2
LARGE BREAKS	04													2.4E-2
	04*													1.6E-2
	04**													6.0E-3
	04***													2.0E-3
	05													9.6E-2
	05*													6.4E-2
	05**													2.4E-2
	05***													1.6E-2
	06													
	06*													
	06**													
	06***													

\* Partial pool bypass.  
\*\* No secondary con-  
tainment filtering.  
\*\*\* Partial pool bypass,  
no filtering.

Table II.4-10

## CONDITIONAL PROBABILITIES FOR CLASS I, PLANT DAMAGE STATE C RELEASE CATEGORIES

TIME/LOCATION RELEASE POSITIONAL	T <sub>1</sub>			T <sub>2</sub>			T <sub>3</sub>			T <sub>4</sub>			NO CONTAINMENT FAILURE	CONDITIONAL PROBABILITY (ISOLATION FAILURE)
	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*		
SMALL BIOL RES	01	-	-	-	-	-	-	-	-	-	-	-	-	-
	01	4.1E-5	4.1E-5	-	-	-	9.7E-4	-	-	-	-	-	3.0E-2	6.0E-5
	01*	-	-	-	-	-	5.5E-4	-	-	-	-	-	2.0E-2	3.0E-5
	01**	-	-	-	-	-	2.4E-4	-	-	-	-	-	-	1.4E-5
	01***	-	-	-	-	-	1.6E-4	-	-	-	-	-	-	9.3E-6
	02	-	-	-	-	-	3.8E-3	-	-	-	-	-	0.12	2.4E-4
	02*	-	-	-	-	-	2.6E-3	-	-	-	-	-	8.0E-2	1.5E-4
	02**	-	-	-	-	-	9.6E-4	-	-	-	-	-	-	5.5E-5
	02***	-	-	-	-	-	6.4E-4	-	-	-	-	-	-	3.7E-5
	03	-	-	-	-	-	1.4E-3	-	-	-	-	-	-	6.0E-5
	03*	-	-	-	-	-	9.7E-4	-	-	-	-	-	-	3.0E-5
	03**	-	-	-	-	-	3.6E-4	-	-	-	-	-	-	1.4E-5
	03***	-	-	-	-	-	2.4E-4	-	-	-	-	-	-	9.3E-6
	04	4.3E-7	4.3E-7	-	-	-	5.8E-3	-	-	-	-	-	-	2.3E-4
LARGE BIOL RES	04*	2.9E-7	2.9E-7	-	-	-	3.8E-3	-	-	-	-	-	-	1.5E-4
	04**	1.1E-7	1.1E-7	-	-	-	1.4E-3	-	-	-	-	-	-	6.0E-5
	04***	7.2E-8	7.2E-8	-	-	-	9.6E-4	-	-	-	-	-	-	3.0E-5
	05	1.7E-6	1.7E-6	-	-	-	9.6E-4	-	-	-	-	-	-	3.0E-5
	05*	1.2E-6	1.2E-6	-	-	-	3.0E-3	-	-	-	-	-	-	6.0E-5
	05**	4.3E-7	4.3E-7	-	-	-	1.4E-3	-	-	-	-	-	-	3.0E-5
	05***	2.9E-7	2.9E-7	-	-	-	9.6E-4	-	-	-	-	-	-	3.0E-5
	06	-	-	-	-	-	-	-	-	-	-	-	-	-
	06*	-	-	-	-	-	-	-	-	-	-	-	-	-
	06**	-	-	-	-	-	-	-	-	-	-	-	-	-
	06***	-	-	-	-	-	-	-	-	-	-	-	-	-
	07	-	-	-	-	-	-	-	-	-	-	-	-	-
	07*	-	-	-	-	-	-	-	-	-	-	-	-	-
	07**	-	-	-	-	-	-	-	-	-	-	-	-	-
	07***	-	-	-	-	-	-	-	-	-	-	-	-	-

\* Partial pool bypass.  
 \*\* No secondary con-  
 tainment filtering.  
 \*\*\* Partial pool bypass,  
 no filtering.

Table II.4-11  
CONDITIONAL PROBABILITIES FOR CLASS I, PLANT DAMAGE STATE D RELEASE CATEGORIES

TYPE/LOCATION	T <sub>1</sub>			T <sub>2</sub>			T <sub>3</sub>			T <sub>4</sub>			NO CONTAINING FAILURE	CONTAINING BYPASS (IF SUCH ACTION FAILURE)
	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*		
SMALL BREAKS	01	-	-	-	-	-	-	-	-	-	-	-	9.1E-2	6.7E-2
	01	7.1E-2	7.1E-2	-	-	-	-	-	-	-	-	-	-	-
	01*	-	-	-	-	-	-	-	-	-	-	-	-	-
	01**	-	-	-	-	-	-	-	-	-	-	-	-	-
	01***	-	-	-	-	-	-	-	-	-	-	-	-	-
	02	-	-	-	-	-	-	-	-	-	-	-	-	-
	02*	-	-	-	-	-	-	-	-	-	-	-	-	-
	02**	-	-	-	-	-	-	-	-	-	-	-	-	-
	02***	-	-	-	-	-	-	-	-	-	-	-	-	-
	03	-	-	-	-	-	-	-	-	-	-	-	-	-
LARGE BREAKS	04	3.4E-3	3.4E-3	-	-	-	5.0E-6	5.0E-6	-	5.0E-2	5.0E-2	-	-	-
	04*	2.3E-3	2.3E-3	-	-	-	3.3E-6	3.3E-6	-	3.3E-2	3.3E-2	-	-	-
	04**	0.7E-4	0.7E-4	-	-	-	1.3E-6	1.3E-6	-	1.3E-2	1.3E-2	-	-	-
	04***	5.0E-4	5.0E-4	-	-	-	0.5E-7	0.5E-7	-	0.4E-3	0.4E-3	-	-	0.10
	05	3.0E-4	3.0E-4	-	-	-	5.7E-5	5.7E-5	-	5.6E-3	5.6E-3	-	-	-
	05*	2.5E-4	2.5E-4	-	-	-	3.0E-5	3.0E-5	-	3.7E-3	3.7E-3	-	-	-
	05**	9.5E-5	9.5E-5	-	-	-	1.5E-5	1.5E-5	-	1.4E-3	1.4E-3	-	-	1.1E-2
	05***	6.3E-5	6.3E-5	-	-	-	9.7E-6	9.7E-6	-	9.3E-4	9.3E-4	-	-	-
	06	-	-	-	-	-	-	-	-	-	-	-	-	-
	06*	-	-	-	-	-	-	-	-	-	-	-	-	-
LARGE BREAKS	07	-	-	-	-	-	-	-	-	-	-	-	-	-
	07*	-	-	-	-	-	-	-	-	-	-	-	-	-
	07**	-	-	-	-	-	-	-	-	-	-	-	-	-
	07***	-	-	-	-	-	-	-	-	-	-	-	-	-
	08	-	-	-	-	-	-	-	-	-	-	-	-	-
	08*	-	-	-	-	-	-	-	-	-	-	-	-	-
	08**	-	-	-	-	-	-	-	-	-	-	-	-	-
	08***	-	-	-	-	-	-	-	-	-	-	-	-	-
	09	-	-	-	-	-	-	-	-	-	-	-	-	-
	09*	-	-	-	-	-	-	-	-	-	-	-	-	-

\* Partial pool bypass.  
\*\* No secondary con-  
tainment filtering.  
\*\*\* Partial pool bypass,  
no filtering.

Table II.4-13  
CONDITIONAL PROBABILITIES FOR CLASS II RELEASE CATEGORIES

TIME / LOCATION	I <sub>1</sub>			I <sub>2</sub>			I <sub>3</sub>			I <sub>4</sub>			CONTAINED RELEASE	CONTAINMENT BYPASSED, X <sub>1</sub>
	DM	WM	WM*	DM	WM	WM*	DM	WM	WM*	DM	WM	WM*		
SMALL BREAKS	0.34	0.34												
LARGE BREAKS														

\* Partial pool bypass.  
\*\* No secondary containment filtering.  
\*\*\* Partial pool bypass, no filtering.



Table II.4-19  
CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE A RELEASE CATEGORIES

TIME/LOCATION	T <sub>1</sub>			T <sub>2</sub>			T <sub>3</sub>			T <sub>4</sub>			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, T <sub>4</sub>
	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*		
SMALL BREAKS	01	3.4E-5												
	01*													
	01**													
	01***													
	02													
	02*													
	02**												0.40	4.3E-4
	02***													
	03													
	03*													
	03**													
	03***												0.10	4.0E-5
	04													
	04*	2.7E-6												
	04**													
	04***	6.0E-7												
LARGE BREAKS	05													
	05*	3.0E-7												
	05**													
	05***	7.5E-8												
	06													
	06*													
	06**													
	06***													
	07													
	07*													
	07**													
	07***													
	08													
	08*													
	08**													
	08***													
LARGE BREAKS	09													
	09*													
	09**													
	09***													
	10													
	10*													
	10**													
	10***													
	11													
	11*													
	11**													
	11***													
	12													
	12*													
	12**													
	12***													
LARGE BREAKS	13													
	13*													
	13**													
	13***													
	14													
	14*													
	14**													
	14***													
	15													
	15*													
	15**													
	15***													
	16													
	16*													
	16**													
	16***													

\* No suppression pool scrubbing.  
\*\* No secondary containment filtering.  
\*\*\* No scrubbing, no filtering.

Table II.4-20  
CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE B RELEASE CATEGORIES

INITIATION RELEASE POTENTIAL	T <sub>1</sub>			T <sub>2</sub>			T <sub>3</sub>			T <sub>4</sub>			NO CONTAINER FAILURE	CONTAINER BYPASSED, T <sub>5</sub>
	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*	DU	WM	WM*		
SMALL BREAKS	00	1.0E-3											0.30	
	01													
	01*													
	01**													
	01***													
	02													
	02*												0.20	4.3E-4
	02**													
	02***													1.1E-4
	03													
	03*													
	03**													
	03***													
	04												7.0E-2	4.0E-5
	04*													1.2E-5
LARGE BREAKS	04**	0.1E-5												
	04***													
	04***	2.0E-5												
	05													
	05*	9.0E-6												
	05**													
	05***	2.3E-6												
	06													
	06*													
	06**													
	06***													
	07													
	07*													
	07**													
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	18*													
	18**													
	18***													
	19													
	19*													
	19**													
	19***													

\* No suppression pool scrubbing.  
\*\* No secondary containment filtering.  
\*\*\* No scrubbing, no filtering.

Table II.4-21  
CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE C RELEASE CATEGORIES

TIME/LOCATION	I <sub>1</sub>			I <sub>2</sub>			I <sub>3</sub>			I <sub>4</sub>			NO CONTAINER FAILURE	CONTAINER BYPASSED, F <sub>1</sub>
	DM	WM	WM*	DM	WM	WM*	DM	WM	WM*	DM	WM	WM*		
00	3.4E-5	-	-	-	-	-	-	-	-	-	-	-	-	-
01	-	-	-	-	-	-	-	-	-	-	-	-	-	-
01*	-	3.4E-5	-	-	-	-	-	-	-	-	-	-	-	2.2E-4
01***	-	-	-	-	-	-	-	-	-	-	-	-	-	5.4E-5
01****	-	-	-	-	-	-	-	-	-	-	-	-	-	2.4E-5
02	-	-	-	-	-	-	-	-	-	-	-	-	-	9.4E-6
02*	-	-	-	-	-	-	-	-	-	-	-	-	-	-
02**	-	-	-	-	-	-	-	-	-	-	-	-	-	-
02***	-	-	-	-	-	-	-	-	-	-	-	-	-	-
03	-	-	-	-	-	-	-	-	-	-	-	-	-	-
03*	-	-	-	-	-	-	-	-	-	-	-	-	-	-
03**	-	-	-	-	-	-	-	-	-	-	-	-	-	-
03***	-	-	-	-	-	-	-	-	-	-	-	-	-	-
04	2.7E-6	2.7E-6	-	-	-	-	3.0E-5	3.0E-5	-	-	-	-	-	-
04*	-	-	-	-	-	-	-	-	-	-	-	-	-	-
04**	-	-	-	-	-	-	7.7E-6	7.7E-6	-	-	-	-	-	-
04***	6.0E-7	6.0E-7	-	-	-	-	-	-	-	-	-	-	-	-
05	-	-	-	-	-	-	-	-	-	-	-	-	-	-
05*	3.0E-7	3.0E-7	-	-	-	-	1.9E-6	1.9E-6	-	-	-	-	-	-
05**	-	-	-	-	-	-	-	-	-	-	-	-	-	-
05***	7.5E-8	7.5E-8	-	-	-	-	4.0E-5	4.0E-5	-	-	-	-	-	-
06	-	-	-	-	-	-	-	-	-	-	-	-	-	-
06*	-	-	-	-	-	-	-	-	-	-	-	-	-	-
06***	-	-	-	-	-	-	-	-	-	-	-	-	-	-
07	-	-	-	-	-	-	-	-	-	-	-	-	-	-
07*	-	-	-	-	-	-	-	-	-	-	-	-	-	-
07***	-	-	-	-	-	-	-	-	-	-	-	-	-	-
08	1.1E-6	2.3E-6	9.0E-6	8.9E-5	1.0E-5	7.1E-5	3.0E-5	7.5E-6	3.0E-5	0.10	3.6E-2	-	-	-
08*	-	-	-	-	-	-	-	-	-	-	-	-	-	-
08**	-	-	-	-	-	-	-	-	-	-	-	-	-	-
08***	-	-	-	-	-	-	-	-	-	-	-	-	-	-
09	1.3E-6	2.5E-7	1.0E-6	9.9E-6	2.0E-6	7.9E-6	2.4E-6	4.7E-6	1.9E-6	4.6E-2	9.0E-3	3.6E-2	-	-
09*	-	-	-	-	-	-	-	-	-	-	-	-	-	-
09**	-	-	-	-	-	-	-	-	-	-	-	-	-	-
09***	-	-	-	-	-	-	-	-	-	-	-	-	-	-

\* No suppression pool scrubbing.  
 \*\* No secondary containment filtering.  
 \*\*\* No scrubbing, no filtering.

Table II.4-22  
CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE D RELEASE CATEGORIES

LINE LOCATION	I <sub>1</sub>			I <sub>2</sub>			I <sub>3</sub>			I <sub>4</sub>			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, I <sub>5</sub>
	DN	WM	WM*	DN	WM	WM*	DN	WM	WM*	DN	WM	WM*		
SMALL BREAKS	01													
	01*													
	01**													
	01***													
	02													
	02*													4.3E-4
	02**													-
	02***													1.1E-4
	03													-
	03*													4.0E-5
	03**													-
	03***													1.2E-5
LARGE BREAKS	04													-
	04*													4.3E-4
	04**													-
	04***													1.7E-4
	05													-
	05*													4.0E-5
	05**													-
	05***													1.2E-5
	06													
	06*													
	06**													
	06***													

\* No suppression pool scrubbing.  
 \*\* No secondary containment filtering.  
 \*\*\* No scrubbing, no filtering.

Table II.4-24  
CONDITIONAL PROBABILITIES FOR CLASS IV RELEASE CATEGORIES

TIME / LOCATION	Y <sub>1</sub>			Y <sub>2</sub>			Y <sub>3</sub>			Y <sub>4</sub>			CONTAINED RELEASE	CONTINGENT BYPASS, Y <sub>1</sub>
	BU	UJ	UJ*	DU	UJ	UJ*	DU	UJ	UJ*	DU	UJ	UJ*		
SMALL BOLARS														
00	9.0E-5	9.0E-5												1.1E-3
01														
01*														
01**														
01***														
02														
02*														
02**														
02***														
03														
03*														
03**														
03***														
04	9.6E-7	9.6E-7												1.2E-5
04*	6.4E-7	6.4E-7												7.1E-6
04**	2.4E-7	2.4E-7												2.9E-6
04***	1.6E-7	1.6E-7												1.9E-6
05	3.0E-6	3.0E-6												4.6E-5
05*	2.6E-6	2.6E-6												3.1E-5
05**	9.6E-7	9.6E-7												1.2E-6
05***	6.4E-7	6.4E-7												7.1E-6
06														
01**														
01***														
02**														
02***														
03**														
03***														
04**	6.0E-2	1.8E-2	1.8E-2											
04***	6.0E-2	6E-2	3.7E-2											
05**	0.24	4.8E-2	0.19											
05***	0.16	3.7E-2	0.13											
LARGE BOLARS														

\* Partial pool bypass.  
\*\* No secondary containment filtering.  
\*\*\* Partial pool bypass, no filtering.



Table II.4-26  
CONDITIONAL PROBABILITIES FOR CLASS V RELEASE CATEGORIES

TIME/LOCATION RELEASE POSITION	T <sub>1</sub>			T <sub>2</sub>			T <sub>3</sub>			T <sub>4</sub>			CONTAINED RELEASE	CONTAINMENT BYPASSED, T <sub>1</sub>
	DM	UM	UM*	DM	UM	UM*	DM	UM	UM*	DM	UM	UM*		
SMALL BREAKS	NR													
	R1													
	R1*													
	R1**													
	R1***													
	R2													
	R2*													
	R2**													
	R2***													
	R3													
	R3*													
	R3**													
LARGE BREAKS	R3***													
	R4													
	R4*													
	R4**													
	R4***													
	R5													
	R5*													
	R5**													
	R5***													
	NR													
	R1**													
	R1***													
	R2**													
	R2***													
	R3**													
	R3***													
	R4**													
	R4***													
	R5**													
	R5***													

\*No suppression pool scrubbing.  
\*\*No secondary containment filtering.  
\*\*\*No scrubbing, no filtering.

ENCLOSURE 2

REQUEST FOR INFORMATION

1. Table II of Appendix M gives different pressure limits for the longitudinal reinforcement bars at the base of the containment and in the wetwell region. However, the longitudinal bars appear to be continuous and should therefore have the same stress. Please explain the basis for the different results.
2. Table II of Appendix M indicates that the shear bars at the base and drywell head have the lowest pressure holding capability (121 psi and 120 psi, respectively) but the discussion indicates that the additional reinforcement will preclude this failure mode. Since the containment failure mode is a key ingredient of the release estimates, please provide a quantitative estimate of the additional shear strength provided by the non-shear reinforcement bars.

3. If shear failure is precluded as discussed in Section 3.2 of Appendix M, "it appears that the ultimate capacity is controlled by the yield of the longitudinal and the hoop bars at about 123 psi." These two failure modes appear to be very important to subsequent fission product release (particularly for Class IV ATWS) since they will both occur in the wetwell region. Please provide an estimate of the size, location and direction (vertical or horizontal) containment failures for each of the three possible failure modes.
4. Section 3.6 of the PRA takes credit for containment leakage which will prevent gross containment failure for all pressurization rates except the very rapid pressurization associated with large breaks. However, the structural analysis by Stone and Webster (Appendix M) did not identify any significant source of leakage. The basis for the expected leakage source and the leakage rate as a function of pressure should be provided.
5. The basis for the partitioning between release category 10 and 11 (no pool bypass vs. partial pool bypass) should be provided. The phenomenological basis for the estimate of only 10% bypass should be provided. Preliminary results from IDCOR indicate that for some BWR sequences the vessel will fail with only 20% of the core molten. Presumably 80% of the melt release would bypass the SRV's and be released into the drywell.

6. The basis for binning into release categories is poorly described and the transfer from Tables H.4-8 etc. into the 16 release categories is difficult to interpret. A table listing the specific sequences which are binned into each category should be provided.
7. The lack of  $R_5$  sequences in the release categories makes it apparent that these releases have been binned "downward" into the lesser release category  $R_4$ . The basis for this "downward" binning and any other sequences that are moved to less severe categories should be provided.
8. Table H.4-25 appears to be incomplete in that it does not include sequences D6 and D8. The completed table should be provided.
9. The source escape fractions used for end state screening (Table 3.6-10) appears to be quite arbitrary yet it greatly influences the importance ranking. In particular: the use of I as the surrogate for melt release ignores the fact that there are noble gases in the melt release which will not be scrubbed at all; the use of a large scrubbing factor (500) for  $C_4$  transients is inappropriate since most of the melt release will be released directly to a failed containment; the reduction factor of 0.01 for  $\gamma$  failures is indefensible since the event tree precludes everything but large ruptures where the pool will be blown out into the reactor building at high pressures.

Table 3.6-10 should be replaced by a table with defensible reduction factors. As a minimum the table should include a separate category for  $C_4$  transients, which recognizes the defined sequence of events (containment failure before core melt). In addition, a detailed justification for each reduction factor should be provided along with the numerical results of the ranking process. This revised table will provide the basis for our independent importance ranking based on revised estimates of accident frequency and reduction factors.

10. Sheet 1 of Figure H.4.2 has been reduced so that it is illegible. A full-size legible copy should be provided.
11. Appendix L provides a detailed discussion of the disposition of the corium (90% is expected to go down the vent pipes) based on the revised reactor pedestal geometry illustrated in Figure L.3-2. However, this figure is inconsistent with other descriptions of the geometry (e.g., Figure 2.3-2) and provides inadequate information for an independent assessment of the corium disposition. Please provide detailed (as built) drawings of the vent pipes and their covers within and external to the reactor pedestal region. Include a description of whether the air ducts and manways in the reactor support wall will be blocked during operation.
12. Provide the estimate of the fraction of the molten corium which is expected to spread out of the pedestal area through the open manways and air ducts in the reactor support wall.