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

NOV 26 1984

MEMORANDUM FOR: Themis P. Speis, Director
Division of Safety Technology, NRR

Ormon E. Bassett, Director
Division of Accident Evaluation, RES

Frank P. Gillespie, Director
Division of Risk Analysis, RES

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

SUBJECT: PROPOSED COMMISSION PAPER ON THE REGULATORY USES OF SOURCE TERM RESEARCH

The NUREG-0956 is now scheduled to be published in early 1985, and will summarize the radiological source term research results.

Upon publication of NUREG-0956, we anticipate that licensees may seek changes in their Technical Specifications, or exemptions from criteria in many areas that relate to the new source term, such as filter testing and efficiency requirements and containment integrity requirements. The potential scope of these requests could be broad since many of our Tech Spec limiting conditions or testing requirements are based upon a radiological source term. While such changes may be appropriate, there will also be a change in emphasis to a more risk based approach to regulation. Specifically, while relaxation in some areas may be appropriate, our understanding of accident risks indicates a need to emphasize other areas such as containment integrity. In order to handle this potential influx of requests in a manner that is both technically consistent with the material reported in NUREG-0956, and consistent in implementation, I propose to prepare a Commission Policy Paper to accompany the forthcoming NUREG-0956. The Policy Paper would outline the manner in which the new radiological source term would be handled in the licensing arena. The Policy Paper would thus explain to the public and the industry how the Commission intends to utilize the research results, to scope the areas in which changes would be accommodated, and to identify the process to be used to effect any changes.

The Policy Paper, to be comprehensive, should represent the thinking of a broad constituency within the staff. I request, therefore, your designation of knowledgeable first line managers (preferably a Branch Chief or Section Leader) to participate in the development and coordination of the Policy Paper. I suggest that participation include representatives from the Fuel Systems Research Branch (ASTPO), Reactor Risk Branch, Reliability and Risk Assessment Branch, and three branches in my Division (the Accident Evaluation Branch, Containment Systems Branch and the Reactor Systems Branch). I suggest all activities, including review of early drafts, should be coordinated through the line

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organization to help ensure a staff consensus before final management review.

The draft of the paper should be scheduled for senior management review by March 1, 1985. I have asked L. G. Hulman to coordinate this effort. Please inform Mr. Hulman of your nominations, or me if there are conflicts. Similarly, by copy of this memo, the AD for Reactor Systems should identify the Containment Systems Branch and Reactor Systems Branch representatives.

A statement of objectives and considerations for use in drafting the Policy Paper are attached.

Robert M. Bernero

Robert M. Bernero, Director
Division of Systems Integration

Attachment:
As Stated

cc: H. R. Denton
R. B. Minogue
R. C. DeYoung
E. Case
D. Ross
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M. Silberberg
A. Thadani
W. Minners
B. Sheron
W. Butler
D. Mathews
J. Malaro
W. Pasedag
L. G. Hulman

CHARTER FOR MANAGEMENT GROUP TO DEVELOP A
COMMISSION POLICY PAPER ON INTEGRATION OF THE
RADIOLOGICAL SOURCE TERM RESEARCH RESULTS
INTO THE REGULATORY PROCESS

1. Become sufficiently familiar with the ASTPO sponsored methodology, APS review, WASH-1400 source terms, TID 14844 use and related risk considerations, to judge policy and practice implications;

2. Identify areas of existing policy and practice influenced by source term research using NUREG-0771 (FOR COMMENT) and the attached summary memo as a starting point for identification (note that both NUREG-0771 and the attached provide such a profile through 1981, but require updating considerations in such areas as severe accident rulemaking, safety goals, safety issue prioritization and resolution and emergency planning);

3. Identify future policy and practice areas potentially influenced by the evolving source term research; e.g., proposed safety goals, siting, environmental statements, severe accident policy, and DL Task Force on Tech Specs.;

4. For each area of regulation that is influenced by the research that may or obviously will have a significant effect on regulatory practice now in place;

   a. describe the current regulatory requirement or practice,

   b. make a tentative evaluation of how it might change based upon new source term information,

   c. make a preliminary estimate of costs and savings achieved by b. above, and

   d. make a recommendation on whether to pursue regulatory change in the area and how it should be done.

5. Identify actions that might be taken by OCA and OPA; and

6. Provide a plan or policy statement for evaluating applicant/licensee requests for deviations from present practice.

Shieu
10/2/84

BROOKHAVEN NATIONAL LABORATORY

MEMORANDUM

DATE: September 27, 1984

TO: W. T. Pratt

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

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.[1] 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,[2,3] 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)[4] 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.

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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 ($C_4R_4T_1$-γ), SNP-11 ($C_4R_4T_1$-γ), and SNP-12 ($C_4R_4T_1$-γ'; $C_4R_4T_1$-γ'') as described in Table 4. Note that SNP-12 consists of both γ' and γ'' scenarios where the γ'' 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 γ'' 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 γ'' 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 ($C_4R_4T_1$-γ) 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 basemat-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 γ'' 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 γ'' sequence. Likewise, it is reasonable to assume that the Shoreham suppression pool will also be drained under such failure conditions (γ'' sequence). If such is the case, the DF of Class IV γ'' sequence should be evaluated explicity. At present, the Shoreham PRA does not include the γ'' scenario in any of the sixteen release calculations. Instead they are "binned" with the γ' 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.[4] 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.

INFORMATION NEEDS

In view of the previous discussions, we would like to request the following information:

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 γ" 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_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.

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 γ'' 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 γ'' and non-γ'' 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 γ" 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 (4×10^{-4}) or the low probability of an oxidation release given containment failure (6×10^{-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
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[redacted]
J. W. Yang
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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 ISOTOPIC RELEASE FRACTION(a)		POTENTIAL RADIOLOGICAL IMPACT(b)	FREQUENCY OF RELEASE
		Cs	Te		
1	I	9.3×10^{-5}	3.4×10^{-3}	Moderate	2.6×10^{-7}
2	I	3.3×10^{-4}	9.1×10^{-5}	Minor	2.0×10^{-6}
3	I	1.5×10^{-7}	1.4×10^{-6}	Minor	2.3×10^{-6}
4	I	ϵ	1.2×10^{-7}	Minor	5.9×10^{-6}
5	II	1.5×10^{-4}	2.5×10^{-2}	Moderate	6.3×10^{-7}
6	II	1.2×10^{-6}	1.1×10^{-5}	Minor	1.2×10^{-6}
7	III	1.6×10^{-1}	3.8×10^{-2}	Severe	$1.[illegible] \times 10^{-7}$
8	III	2.8×10^{-5}	1.2×10^{-4}	Minor	6.1×10^{-7}
9	III	3.1×10^{-7}	1.2×10^{-6}	Minor	6.7×10^{-7}
10	IV,I,II	2.3×10^{-2}	2.5×10^{-2}	Moderate	3.1×10^{-6}
11	IV	1.6×10^{-4}	2.5×10^{-2}	Moderate	3.8×10^{-6}
12	IV	1.7×10^{-4}	2.9×10^{-4}	Minor	6.2×10^{-6}
13	V	1.3×10^{-1}	3.0×10^{-3}	Severe	2.5×10^{-6}
14	V	2.6×10^{-1}	3.0×10^{-3}	Severe	1.1×10^{-8}
15	I,II,III,IV	ϵ	1.2×10^{-8}	Negligible	8.3×10^{-6}
16	I,II,III,IV	ϵ	ϵ	Negligible	1.9×10^{-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.

ϵ - The release fractions for this radionuclide species are less than 10^{-9} 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[1]	.07-.73[2]	1.6×10^{-1}	.3	.1-.76[2]	2.5×10^{-2}

[1]Based on comparable BWR release assuming elemental iodine.

[2]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	CET 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.	$C_1R_4T_1 - \delta$ 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.	$C_1R_4T_1 - \partial$ 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.	$C_1R_4T_4 - \gamma$ 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.	$C_1R_4T_4 - \gamma'$ $C_1R_4T_4 - \gamma''$ 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	CET 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 RHR, RCICSC, 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.	$C_2R_4T_1 - \gamma$ 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 RHR, RCICSC, 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.	$C_2R_4T_1 - \gamma'$ $C_2R_4T_1^- - \gamma''$ 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 sequence 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.	$C_3R_4T_1 - \gamma$ 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.	$C_3R_4T_4 - \gamma$ 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	CET PROGRESSION PATH DESIGNATOR
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.	$C_3R_4T_4 - \gamma'$ $C_3R_4T_4 - \gamma''$ 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.	$C_4R_4T_1 - \gamma$ 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.	$C_4R_4T_1 - \gamma$ 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.	$C_4R_4T_1 - \gamma'$ $C_4R_4T_1 - \gamma''$ Early Failure un 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	CET 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.	$C_5R_4T_1$ 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.	$C_5R_4T_1$ 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.	$C_1R_5T_4$ - 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.	$C_1R_5T_4$ - 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	
$C_1R_4T_1 - \delta$	CI-DW	T	600	100	M,C
$C_1R_4T_1 - \partial$	CI-WW	T	120(d)	20 (d)	M,C
$C_1R_4T_4 - \gamma$	OP-DW	T	600	100	M,C
$C_1R_4T_4 - \gamma'$	OP-WW	T	600	100	M,C
$C_2R_4T_1 - \gamma$	OP-DW	T	600	100	M,C
$C_2R_4T_1 - \gamma'$	OP-WW	T	600	100	M,C
$C_3R_4T_1 - \gamma$	OP-DW	L	-	100	M,C
$C_3R_4T_4 - \gamma$	OP-DW	L	-	100	M,C
$C_3R_4T_4 - \gamma'$	OP-WW	L	-	100	M,C
$C_4R_4T_1A - \gamma$	OP-DW	T(e)	600	100	M,C
$C_4R_4T_1B - \gamma$	OP-DW	T	600	100	M,C
$C_4R_4T_1 - \gamma'$	OP-WW	T	600	100	M,C
$C_5R_4T_1 - A$	BP	L(f)	2000	100	M,C,R
$C_5R_4T_1 - B$	BP	L	-	100	M,C,R

(a) CI - Containment Isolation Failure
OP - Containment Overpressure Failure
DW - Containment Breach is in the Drywell (γ)
WW - Containment Breach is in the Wetwell (γ')

(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 - Contempt module of RACAP-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

GENERIC SEQUENCE TYPE	FUEL RELEASE	SOURCE FACTOR	IN-VESSEL ESCAPE FRACTION	POOL SCRUBBING	CONTAINMENT FAILURE TIME PHASE T1*	T2	T3	T4	PRIMARY CONTAINMENT FAILURE MODES: LEAKAGE	OVERPRESSURE FAILURE BREAK SIZE: SMALL	LARGE	CONTAINMENT BREAK LOCATION: DW	WW	WW'	REACTOR BUILDING NODALIZATION	RBSVS FILTER EFFECTIVENESS
Transient:	Gap (Xe)**	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	1.0	1.0
C_1,C_2,C_4	Melt (I)	1.0	0.20	0.002	1.0	0.5	0.1	0.01	0.05	0.05	0.9	1.0	0.5	0.01	0.5	0.001
	Oxidation (Te-Ru)	0.03	-	-	-	-	0.5	0.01	0.05	0.05	0.9	1.0	0.5	0.01	0.5	0.001
	Vaporization (Te)	0.09	-	-	-	-	-	0.10	0.05	0.05	0.9	1.0	0.5	0.01	0.5	0.001
LOCA	Gap (Xe)	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	1.0	1.0
C_3,C_5***	Melt (I)	1.0	0.90	-	1.0	0.5	0.1	0.01	0.05	0.05	0.9	1.0	0.5	0.01	0.5	0.001
	Oxidation (Te-Ru)	0.05	-	-	-	-	0.5	0.01	0.05	0.05	0.9	1.0	0.5	0.01	0.5	0.001
	Vaporization (Te)	0.09	-	-	-	-	-	0.10	0.05	0.05	0.9	1.0	0.5	0.01	0.5	0.001

* T1 time phase includes failure to isolate the containment immediately following the initiating event.

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

***The C_5 accident class would bypass containment during in-vessel meltdown. Therefore, the removal mechanisms associated with the containment for the gap and melt would not be applicable.

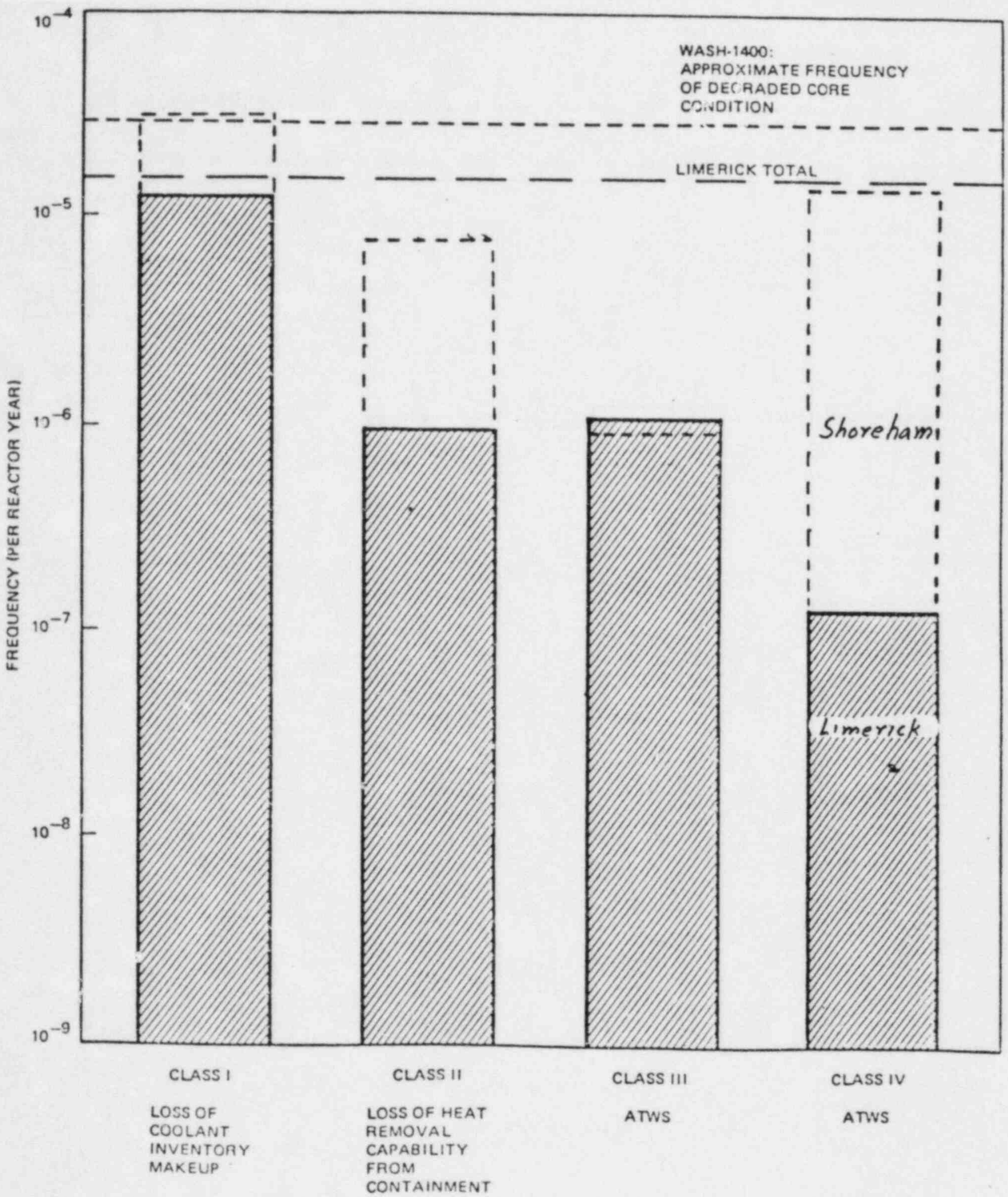

Figure 1 Summary of the accident sequence frequencies leading to degraded core conditions summed over all accident sequences within a class.

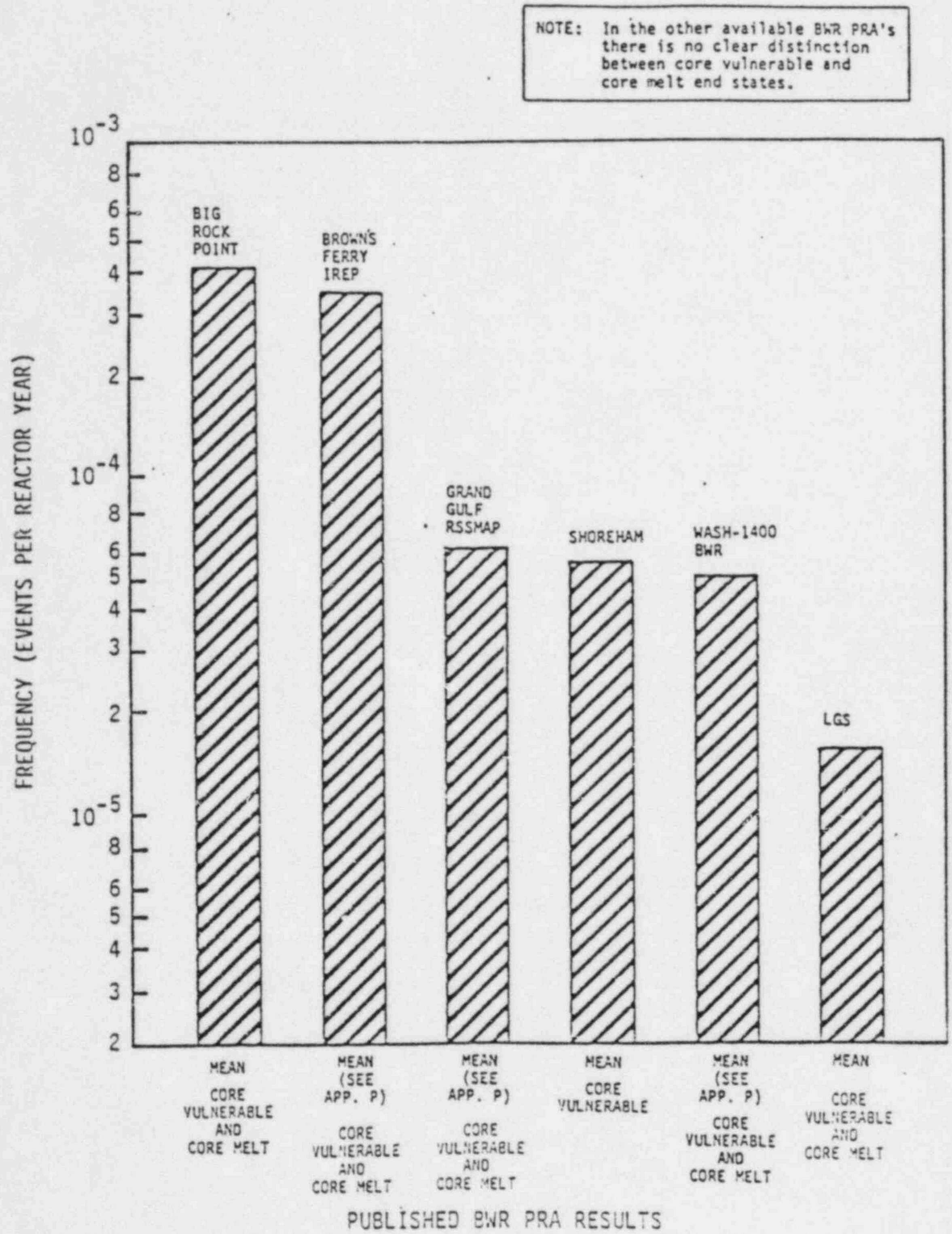

Figure 2 Comparison of Frequency of Core Vulnerable/Core Melt from Published BWR PRAs.

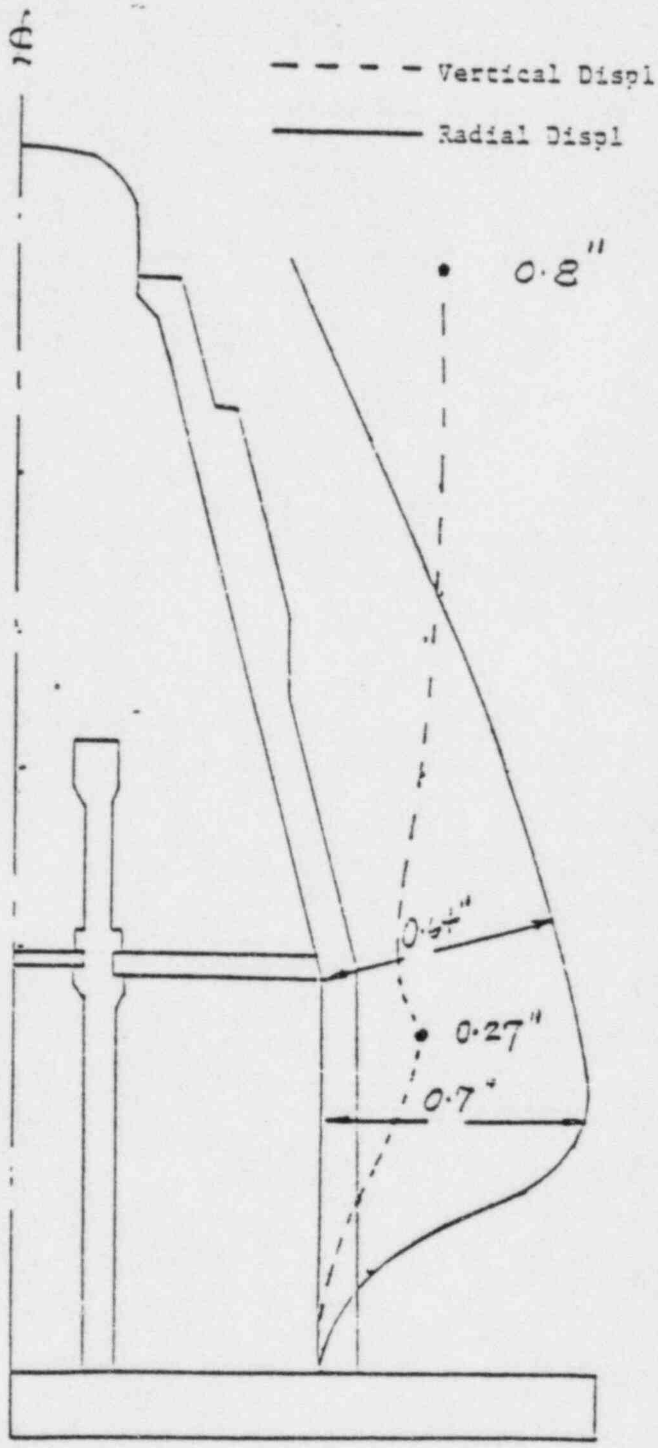

FIGURE 3 - GROWTH OF CONTAINMENT UNDER 120 PSI INTERNAL PRESSURE

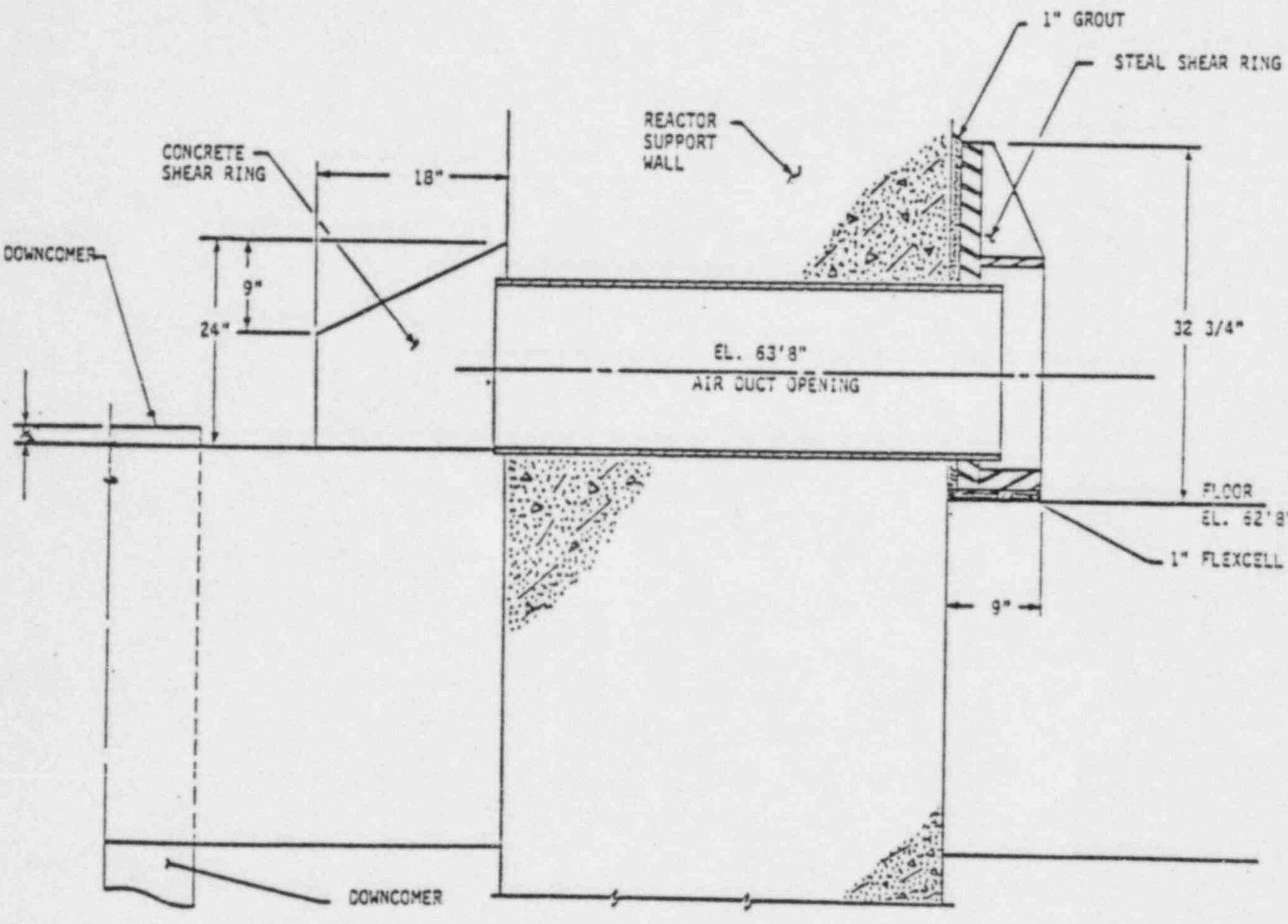

Figure 4 Reactor Pedestal Schematic Showing Reactor Downcomer Geometry

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 DESIGNATOR	PHYSICAL BASIS FOR CLASSIFICATION	SYSTEM LEVEL CONTRIBUTING EVENT SEQUENCE	REPRESENTATIVE SEQUENCE FOR CLASS
Class I (C_1)	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_2)	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_3)	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 LOCAs 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_4)	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 RPS and failure of SLCS
Class V (C_5)	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_1	Class I	Pool scrubbing prior to transport in containment
C_2	Class II	Pool scrubbing prior to transport in containment
C_3	Class III	Pool scrubbing is bypassed prior to transport in containment
C_4	Class IV	Pool scrubbing prior to transport in containment
C_5	Class V	Pool scrubbing and containment are bypassed
	RELEASE TYPES	
R_1	Gap Release	Recovered accident after initial core overheat
R_2, R_3	Gap and Melt plus oxidation release	Recovered accident after significant core melting
R_4, R_5	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_1	Class I	Pool scrubbing prior to transport in containment
C_2	Class II	Pool scrubbing prior to transport in containment
C_3	Class III	Pool scrubbing is bypassed prior to transport in containment
C_4	Class IV	Pool scrubbing prior to transport in containment
C_5	Class V	Pool scrubbing and containment are bypassed
	RELEASE TYPES	
R_1	Gap Release	Recovered accident after initial core overheat
R_2,R_3	Gap and Melt plus oxidation release	Recovered accident after significant core melting
R_4,R_5	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_1	Time phase T_1	Containment is failed before core degradation
T_2	Time phase T_2	Containment fails during core meltdown in vessel
T_3	Time phase T_3	Containment fails during core-concrete interaction
T_4	Time phase T_4	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
∂	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_1	Core heatup (gap)
R_2	Core heatup and melt release (gap and melt)
R_3	Core heatup and melt release with potential for oxidation release (gap, melt and oxidation)
R_4	Core heatup and melt release, and vaporization release (gap, melt and vaporization)
R_5	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_1	From the time of accident initiation to initiation of core overheating
T_2	From the time of initiation of core overheating until the time of pressure vessel failure
T_3	From the time of vessel failure until soon after vessel failure or when core-concrete interaction occurs
T_4	Long after vessel failure or vaporization release has occurred

ATTACHMENT 2

Conditional Probabilities for all Release Categories

Table H.4-8

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

	TIME/LOCATION	T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, T_1 (ISOLATION FAILURE)
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	NR													-	-
	R1	1.1E-3	1.1E-3											0.30	3.6E-4
	R1*													-	-
	R1**													-	-
	R1***													-	-
	R2							6.7E-7	6.7E-7					4.2E-2	4.0E-5
	R2*							4.5E-7	4.5E-7					2.8E-2	2.6E-5
	R2**							1.7E-7	1.7E-7					-	9.8E-6
	R2***							1.1E-7	1.1E-7					-	6.5E-6
	R3							3.7E-5	3.7E-5					0.17	1.6E-4
	R3*							2.5E-5	2.5E-5					0.11	1.1E-4
	R3**							9.2E-6	9.2E-6					-	4.0E-5
	R3***							6.1E-6	6.1E-6					-	2.6E-5
	R4	1.1E-5	1.1E-5					1.0E-6	1.0E-6		8.3E-3	8.3E-3			4.0E-5
	R4*	7.4E-6	7.4E-6					6.7E-7	6.7E-7		5.5E-3	5.5E-3			2.6E-5
	R4**	2.8E-6	2.8E-6					2.5E-7	2.5E-7		2.1E-3	2.1E-3			9.8E-6
	R4***	1.9E-6	1.9E-6					1.7E-7	1.7E-7		1.4E-3	1.4E-3			6.5E-6
	R5	4.4E-5	4.4E-5					5.5E-5	5.5E-5		3.3E-2	3.3E-2			1.6E-4
	R5*	3.0E-5	3.0E-5					3.7E-5	3.7E-5		2.2E-2	2.2E-2			1.1E-4
	R5**	1.1E-5	1.1E-5					1.4E-5	1.4E-5		8.3E-3	8.3E-3			4.0E-5
	R5***	7.4E-6	7.4E-6					9.2E-6	9.2E-6		5.5E-3	5.5E-3			2.6E-5
LARGE BREAKS	NR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	4.7E-5	9.3E-6	3.7E-5	8.4E-6	1.7E-6	6.7E-6	2.1E-6	4.2E-7	1.7E-6	1.1E-2	2.1E-3	8.4E-3		
	R4***	3.1E-5	6.2E-6	2.5E-[illegible]	5.6E-6	1.1E-6	4.5E-6	1.4E-6	2.8E-7	1.1E-6	7.3E-3	1.4E-3	5.6E-3		
	R5**	1.9E-4	3.7E-5	1.5E-4	3.4E-5	6.7E-6	2.7E-5	1.2E-4	2.3E-5	9.2E-5	4.3E-2	8.3E-3	3.3E-3		
	R5***	1.2E-4	2.5E-5	9.8E-5	2.2E-5	4.5E-6	1.8E-5	7.7E-5	1.5E-5	6.1E-5	2.9E-2	5.5E-3	2.2E-3		

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

Table II.4-9

CONDITIONAL PROBABILITIES FOR CLASS I, PLANT DAMAGE STATE B RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED T_1 (ISOLATION FAILURE)
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	NR														-
	R1														0.50
	R1*														-
	R1**														-
	R1***														-
	R2														2.4E-2
	R2*														1.6E-2
	R2**														6.0E-3
	R2***														2.0E-3
	R3														9.6E-2
	R3*														6.4E-2
	R3**														2.4E-2
	R3***														1.6E-2
	R4														2.4E-2
	R4*														1.6E-2
	R4**														6.0E-3
	R4***														2.0E-3
	R5														9.6E-2
	R5*														6.4E-2
	R5**														2.4E-2
	R5***														1.6E-2
LARGE BREAKS	NR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**														
	R4***														
	R5**														
	R5***														

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

Table H.4-10

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

	TIME/LOCATION	T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED,T_1 (ISOLATION FAILURE)
	RELEASE POTENTIAL	DW	WW	WW*	DW	WW	WW*	DW	WW	WW*	DW	WW	WW*		
SMALL BREAKS	HR	-													
	H1	4.1E-5	4.1E-5												
	H1*	-													
	H1**	-													
	H1***	-													
	H2	-						9.7E-4	9.7E-4					3.0E-2	6.0E-5
	H2*							6.5E-4	6.5E-4					2.0E-2	3.8E-5
	H2**	-						2.4E-4	2.4E-4					-	1.4E-5
	H2***							1.6E-4	1.6E-4					-	9.3E-6
	H3	-						3.8E-3	3.8E-3					0.12	2.3E-4
	H3*							2.6E-3	2.6E-3					8.0E-2	1.5E-4
	H3**	-						9.6E-4	9.6E-4					-	5.5E-5
	H3***							6.4E-4	6.4E-4					-	3.7E-5
	H4	4.3E-7	4.3E-7					1.4E-3	1.4E-3		1.1E-2	1.1E-2			6.0E-5
	H4*	2.9E-7	2.9E-7					9.7E-4	9.7E-4		7.1E-3	7.1E-3			3.8E-5
	H4**	1.1E-7	1.1E-7					3.6E-4	3.6E-4		2.7E-3	2.7E-3			1.4E-5
	H4***	7.2E-8	7.2E-8					2.4E-4	2.4E-4		1.8E-3	1.8E-3			9.3E-6
	H5	1.7E-6	1.7E-6					5.8E-3	5.8E-3		4.3E-2	4.3E-2			2.3E-4
	H5*	1.2E-6	1.2E-6					3.8E-3	3.8E-3		2.8E-2	2.8E-2			1.5E-4
	H5**	4.3E-7	4.3E-7					1.4E-3	1.4E-3		1.1E-2	1.1E-2			6.0E-5
	H5***	2.9E-7	2.9E-7					9.6E-4	9.6E-4		7.1E-3	7.1E-3			3.8E-5
LARGE BREAKS	HR														
	H1**														
	H1***														
	H2**														
	H2***														
	H3**														
	H3***														
	H4**	3.6E-6	7.2E-7	2.9E-6	1.2E-5	2.4E-6	9.5E-6	3.0E-3	6.0E-4	2.4E-3	1.4E-2	2.7E-3	1.1E-2		
	H4***	2.4E-6	4.8E-7	1.9E-6	7.9E-6	1.6E-6	6.3E-6	2.0E-3	4.0E-4	1.6E-3	9.3E-3	1.8E-3	7.3E-3		
	H5**	1.4E-5	2.9E-6	1.2E-5	4.8E-5	9.5E-6	3.8E-5	1.2E-2	2.4E-3	9.6E-3	5.5E-2	1.1E-2	4.3E-2		
	H5***	9.6E-6	1.9E-6	7.7E-6	3.2E-5	6.3E-6	2.5E-5	8.0E-3	1.6E-3	6.4E-3	3.7E-2	7.1E-3	2.8E-2		

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

Table H.4-11

CONDITIONAL PROBABILITIES FOR CLASS I, PLANT DAMAGE STATE D RELEASE CATEGORIES

LINE/LOCATION		T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, T_1 (ISOLATION FAILURE)
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	NR	-	-											-	-
	R1	7.1E-2	7.1E-2											9.1E-2	6.7E-2
	R1*	-	-											-	-
	R1**	-	-											-	-
	R1***	-	-											-	-
	R2	-													
	R2*														
	R2**	-													
	R2***														
	R3	-													
	R3*														
	R3**	-													
	R3***														
	R4	3.4E-3	3.4E-3					5.0E-6	5.0E-6		5.0E-2	5.0E-2			-
	R4*	2.3E-3	2.3E-3					3.3E-6	3.3E-6		3.3E-2	3.3E-2			-
	R4**	8.7E-4	8.7E-4					1.3E-6	1.3E-6		1.3E-2	1.3E-2			0.10
	R4***	5.8E-4	5.8E-4					8.5E-7	8.5E-7		8.4E-3	8.4E-3			-
	R5	3.8E-4	3.8E-4					5.7E-5	5.7E-5		5.6E-3	5.6E-3			-
	R5*	2.5E-4	2.5E-4					3.8E-5	3.8E-5		3.7E-3	3.7E-3			-
	R5**	9.5E-5	9.5E-5					1.5E-5	1.5E-5		1.4E-3	1.4E-3			1.1E-2
	R5***	6.3E-5	6.3E-5					9.7E-6	9.7E-6		9.3E-4	9.3E-4			-
LARGE BREAKS	NR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	1.4E-2	2.8E-3	1.1E-2	2.8E-5	5.6E-6	2.2E-5	6.3E-6	1.3E-6	5.0E-6	6.3E-2	1.3E-2	5.0E-2		
	R4***	9.5E-3	1.9E-3	7.6E-3	1.9E-5	3.7E-6	1.5E-5	4.2E-6	8.4E-7	3.3E-6	4.2E-3	8.4E-3	3.3E-2		
	R5**	1.6E-3	3.2E-4	1.3E-3	3.1E-6	6.2E-7	2.5E-6	7.2E-5	1.4E-5	5.7E-5	7.0E-3	1.4E-3	5.6E-3		
	R5***	1.1E-3	2.1E-4	8.4E-4	2.1E-6	4.1E-7	1.7E-6	4.8E-5	9.5E-6	3.8E-5	4.7E-3	9.3E-4	3.7E-3		

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

Table H.4-13

CONDITIONAL PROBABILITIES FOR CLASS II RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3			T_4			CONTAINED RELEASE	CONTAINMENT BYPASSED, T_1
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	NR	0.34	0.34												
	R1														
	R1*														
	R1**														
	R1***														
	R2														
	R2*														
	R2**														
	R2***														
	R3														
	R3*														
	R3**														
	R3***														
	R4	3.6E-3	3.6E-3												1.2E-5
	R4*	1.5E-3	1.5E-3												1.9E-5
	R4**	9.0E-4	9.0E-4												2.9E-6
	R4***	1.5E-3	1.5E-3												4.8E-6
	R5	1.4E-2	1.4E-2												4.6E-5
	R5*	6.0E-3	6.0E-3												1.9E-5
	R5**	3.6E-3	3.6E-3												1.2E-5
	R5***	1.5E-3	1.5E-3												4.8E-6
LARGE BREAKS	NR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	1.5E-2	3.0E-3	1.2E-2											
	R4***	2.5E-2	5.0E-3	2.0E-2											
	R5**	6.0E-2	1.2E-2	4.8E-2											
	R5***	2.5E-2	5.0E-3	2.0E-2											

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

Table H.4-19

CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE A RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, T_1
	RELEASE POTENTIAL	DW	WW	WW*	DW	WW	WW*	DW	WW	WW*	DW	WW	WW*		
SMALL BREAKS	HR														
	R1	3.4E-5	3.4E-5												
	R1*														
	R1**														
	R1***														
	R2							-	-					-	-
	R2*							7.0E-6	7.0E-6					0.40	4.3E-4
	R2**							-	-					-	-
	R2***							1.7E-6	1.7E-6					-	1.1E-4
	R3							-	-					-	-
	R3*							8.0E-5	8.0E-5					0.10	4.8E-5
	R3**							-	-					-	-
	R3***							2.0E-5	2.0E-5					-	1.2E-5
	R4	-	-					-	-		-	-			-
	R4*	2.7E-6	2.7E-6					1.0E-5	1.0E-5		7.8E-2	7.8E-2			4.3E-4
	R4**	-	-					-	-		-	-			-
	R4***	6.8E-7	6.8E-7					2.6E-6	2.6E-6		2.0E-2	2.0E-2			1.1E-4
	R5	-	-					-	-		-	-			-
	R5*	3.0E-7	3.0E-7					1.2E-4	1.2E-4		2.0E-2	2.0E-2			4.8E-5
	R5**	-	-					-	-		-	-			-
	R5***	7.5E-8	7.5E-8					3.0E-5	3.0E-5		5.0E-3	5.0E-3			1.2E-5
LARGE BREAKS	HR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	-	-	-	-	-	-	-	-	-	-	-	-		
	R4***	1.1E-5	2.3E-6	9.0E-6	8.9E-5	1.8E-5	7.1E-5	2.2E-5	4.4E-6	1.7E-5	0.10	2.0E-2	7.9E-2		
	R5**	-	-	-	-	-	-	-	-	-	-	-	-		
	R5***	1.3E-6	2.5E-7	1.0E-6	9.9E-6	2.0E-6	7.9E-6	2.5E-4	5.0E-5	2.0E-4	2.5E-2	5.0E-3	2.0E-2		

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

Table H.4-20

CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE B RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, T_1
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	NR													-	
	R1	1.0E-3	1.0E-3											0.30	
	R1*													-	
	R1**													-	
	R1***													-	
	R2							-	-					-	-
	R2*							4.9E-6	4.9E-6					0.28	4.3E-4
	R2**							-	-					-	-
	R2***							1.2E-6	1.2E-6					-	1.1E-4
	R3							-	-					-	-
	R3*							5.6E-5	5.6E-5					7.0E-2	4.8E-5
	R3**							-	-					-	-
	R3***							1.4E-5	1.4E-5					-	1.2E-5
	R4	-	-					-	-		-	-			-
	R4*	8.1E-5	8.1E-5					7.0E-6	7.0E-6		5.5E-2	5.5E-2			4.3E-4
	R4**	-	-					-	-		-	-			-
	R4***	2.0E-5	2.0E-5					1.8E-6	1.8E-6		1.4E-2	1.4E-2			1.1E-4
	R5	-	-					-	-		-	-			-
	R5*	9.0E-6	9.0E-6					8.4E-5	8.4E-5		1.4E-2	1.4E-2			4.8E-5
	R5**	-	-					-	-		-	-			-
	R5***	2.3E-6	2.3E-6					2.1E-5	2.1E-5		3.5E-3	3.5E-3			1.2E-5
LARGE BREAKS	NR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	-	-	-	-	-	-	-	-	-	-	-	-		
	R4***	3.3E-4	6.9E-5	2.7E-4	6.2E-5	1.3E-5	5.0E-5	1.5E-5	3.1E-6	1.2E-5	7.0E-2	1.4E-2	5.6E-2		
	R5**	-	-	-	-	-	-	-	-	-	-	-	-		
	R5***	3.9E-5	7.8E-6	3.1E-5	6.9E-6	1.4E-6	5.5E-6	1.8E-4	3.5E-5	1.4E-4	1.8E-2	3.6E-3	1.4E-2		

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

Table H.4-21

CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE C RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, T_1
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	HR	-	-												
	R1	3.4E-5	3.4E-5												
	R1*														
	R1**														
	R1***														
	R2														-
	R2*														2.2E-4
	R2**														-
	R2***														5.4E-5
	R3														-
	R3*														2.4E-5
	R3**														-
	R3***														9.6E-6
	R4	-	-					-	-		-	-			-
	R4*	2.7E-6	2.7E-6					3.0E-5	3.0E-5		0.14	0.14			6.5E-4
	R4**	-	-					-	-		-	-			-
	R4***	6.8E-7	6.8E-7					7.7E-6	7.7E-6		3.6E-2	3.6E-2			1.6E-4
	R5	-	-					-	-		-	-			-
	R5*	3.0E-7	3.0E-7					1.9E-4	1.9E-4		3.6E-2	3.6E-2			7.2E-5
	R5**	-	-					-	-		-	-			-
	R5***	7.5E-8	7.5E-8					4.8E-5	4.8E-5		9.0E-3	9.0E-3			1.8E-5
LARGE BREAKS	HR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	-	-	-	-	-	-	-	-	-	-	-	-		
	R4***	1.1E-5	2.3E-6	9.0E-6	8.9E-5	1.8E-5	7.1E-5	3.8E-5	7.5E-6	3.0E-5	0.18	3.6E-2	0.14		
	R5**	-	-	-	-	-	-	-	-	-	-	-	-		
	R5***	1.3E-6	2.5E-7	1.0E-6	9.9E-6	2.0E-6	7.9E-6	2.4E-4	4.7E-5	1.9E-4	4.5E-2	9.0E-3	3.6E-2		

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

Table H.4-22

CONDITIONAL PROBABILITIES FOR CLASS III, PLANT DAMAGE STATE D RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3			T_4			NO CONTAINMENT FAILURE	CONTAINMENT BYPASSED, T_1
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	HR														
	R1														
	R1*														
	R1**														
	R1***														
	R2														-
	R2*														4.3E-4
	R2**														-
	R2***														1.1E-4
	R3														-
	R3*														4.8E-5
	R3**														-
	R3***														1.2E-5
	R4														-
	R4*														4.3E-4
	R4**														-
	R4***														1.1E-4
	R5														-
	R5*														4.8E-5
	R5**														-
	R5***														1.2E-5
LARGE BREAKS	HR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	-	-	-											
	R4***	0.45	8.9E-2	0.36											
	R5**	-	-	-											
	R5***	5.0E-2	1.0E-2	4.0E-2											

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

Table H.4-24

CONDITIONAL PROBABILITIES FOR CLASS IV RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3			T_4				
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	CONTAINED RELEASE	CONTAINMENT BYPASSED, T_1
SMALL BREAKS	NR	9.0E-5	9.0E-5												
	R1														1.1E-3
	R1*														
	R1**														
	R1***														
	R2														
	R2*														
	R2**														
	R2***														
	R3														
	R3*														
	R3**														
	R3***														
	R4	9.6E-7	9.6E-7												1.2E-5
	R4*	6.4E-7	6.4E-7												7.7E-6
	R4**	2.4E-7	2.4E-7												2.9E-6
	R4***	1.6E-7	1.6E-7												1.9E-6
	R5	3.8E-6	3.8E-6												4.6E-5
	R5*	2.6E-6	2.6E-6												3.1E-5
	R5**	9.6E-7	9.6E-7												1.2E-5
	R5***	6.4E-7	6.4E-7												7.7E-6
LARGE BREAKS	NR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**	6.0E-2	1.2E-2	4.8E-2											
	R4***	4.0E-2	8E-3	3.2E-2											
	R5**	0.24	4.8E-2	0.19											
	R5***	0.16	3.2E-2	0.13											

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

Table H.4-26

CONDITIONAL PROBABILITIES FOR CLASS V RELEASE CATEGORIES

	TIME/LOCATION	T_1			T_2			T_3'			T_4'			CONTAINED RELEASE	CONTAINMENT BYPASSED, T_1
	RELEASE POTENTIAL	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'	DW	WW	WW'		
SMALL BREAKS	NR														
	R1														
	R1*														
	R1**														
	R1***														
	R2														
	R2*														
	R2**														
	R2***														
	R3														
	R3*														
	R3**														
	R3***														
	R4														--
	R4*														--
	R4**														--
	R4***														0.80
	R5														--
	R5*														--
	R5**														--
	R5***														0.20
LARGE BREAKS	NR														
	R1**														
	R1***														
	R2**														
	R2***														
	R3**														
	R3***														
	R4**														
	R4***														
	R5**														
	R5***														

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