DRAFT

REVIEW AND CRITIQUE OF PREVIOUS PROBABILISTIC ACCIDENT ASSESSMENTS FOR THE SHOREHAM NUCLEAR POWER STATION

Report on Task 1 of the Project

"Consequence Assessment for Suffolk County Radiological Emergency Response Plan"

VOLUME I: MAIN REPORT

Prepared for

Radiological Emergency Response Plan
Steering Committee

Frank Jones, Chairman
Office of the County Executive
County of Suffolk
H. Lee Dennison Building
Hauppauge, New York 11788

Robert J. Budnitz
Principal Author

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Stan Fabic
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September 17, 1982

Future Resources Associates, Inc.

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1.0 Introduction

This report summarizes the work carried out by Future Resources Associates, Inc. (FRA) under contract to Suffolk County, New York on the project entitled "Consequence Assessment for Suffolk County Radiological Emergency Response Plan." The overall goal of the project is to provide Suffolk County with technical support in its development of an emergency response plan for the Shoreham Nuclear Power Station, in particular by providing technical input as to the probabilities, severities, and radiological dispersion characteristics of potential large accidents at Shoreham. Shoreham is a boiling water reactor (BWR 4-Mark II) in final stages of construction on the north shore of Long Island, New York, with an electrical gross power rating of 846 megawatts. The reactor was manufactured by the General Electric Company and the architect-engineering work has been done by Stone & Webster. The location of the reactor on Long Island and the site plan for the Shoreham facility itself are shown on the next two pages: these are reproduced directly from Figures 2-1 and 2-3 of the NRC's "Safety Evaluation Report" for Shoreham (Ref. 6).

This project is a joint one involving a single unified scope of work under two contracts, one with FRA and the other one with Finlayson & Associates of Cerritos, California; under the arrangement with Suffolk County, Fred C. Finlayson of Finlayson & Associates has assumed overall responsibility to

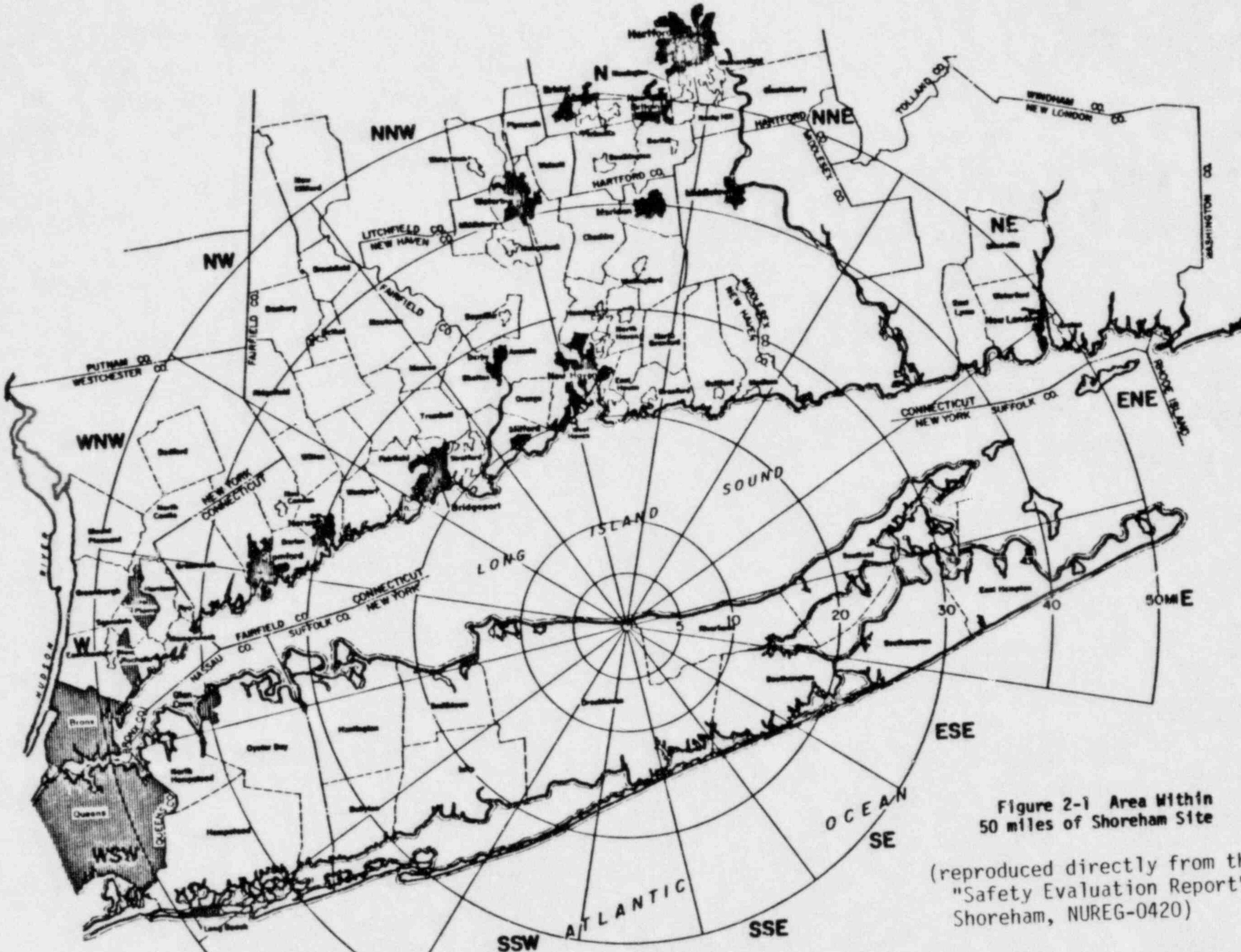

Figure 2-1 Area Within 50 miles of Shoreham Site

(reproduced directly from the NRC's "Safety Evaluation Report" on Shoreham, NUREG-0420)

(reproduced directly from NUREG-0420, the NRC's "Safety Evaluation Report" for the Shoreham plant)

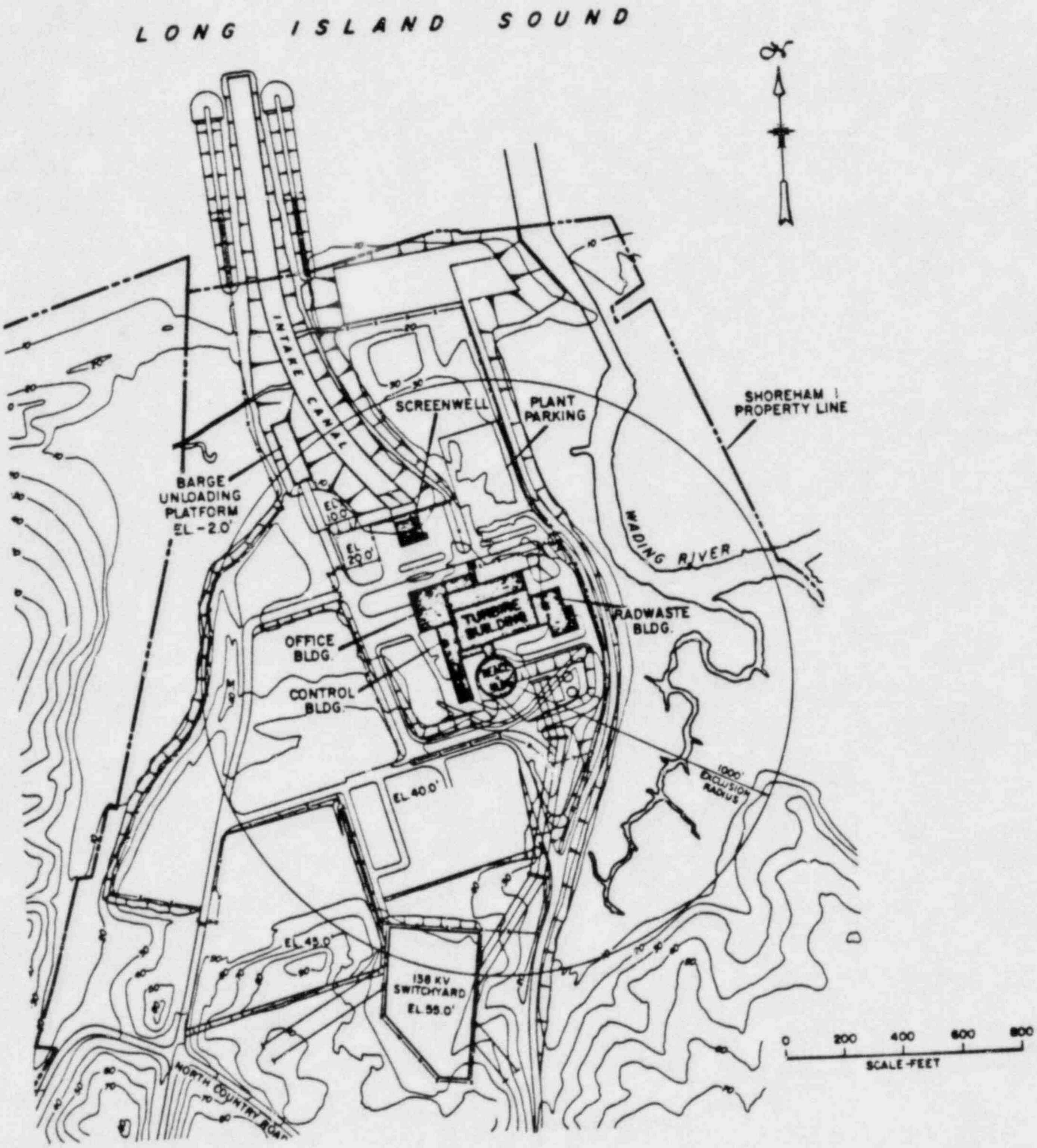

Figure 2-3 Plant Layout

coordinate the two contract efforts. The unified scope of work of the two contracts is reproduced in Appendix A of this report, and covers four tasks. FRA, under the technical direction of Robert J. Budnitz, has been principally responsible for the work under Task 1, while Finlayson & Associates is principally responsible for Tasks 2 and 3; in each task the other party has played a supporting role. Task 4, involving project management, integration, documentation, and technology transfer, has been a shared responsibility.

The contract was signed in late June 1982, and work under it has taken place predominantly in the months of June, July, and August, 1982 with this report due in draft form on September 15.

The review team made a site visit to the Shoreham plant in late June. Preliminary discussions of findings have taken place on an almost continuous basis, informally and verbally, between Dr. Budnitz and Dr. Finlayson in order to assure that both parts of the overall project are integrated effectively. Both parties agree that the integration has been successfully accomplished.

The FRA responsibility within the study has been predominantly to carry out Task 1, "Review and Critique of Previous Probabilistic Risk Analysis," and this report covers the work that FRA has accomplished in carrying out Task 1. Task 1 has consisted largely of a review of the preliminary draft report
coordinate the two contract efforts. The unified scope of work of the two

entitled "Probabilistic Risk Assessment, Shoreham Nuclear Power Station," that was carried out by the San Jose, California office of Science Applications, Inc. (SAI) for Long Island Lighting Company (LILCO), the owner of the Shoreham facility. To carry out this review task, FRA has utilized a team of four individuals who possess expertise in various aspects of water reactor safety and in particular of probabilistic risk ssessment (PRA). The FRA work has been led by Robert J. Budnitz, President of FRA, and has included Howard E. Lambert and Peter R. Davis, FRA consultants, and Stan Fabic of Dynatrek, Inc. (Rockville, MD), an FRA subcontractor.

The scope of the FRA work in Task 1 has been that of an independent review, which must be understood as quite different from an independent analysis of the potential accidents at the Shoreham plant. The purpose of the review has been to ascertain whether the PRA results obtained by SAI in their large and voluminous study are sufficiently reliable to form an acceptable basis for the County's emergency planning work. Because the scope of work has not included significant independent analysis, it is important to realize that its conclusion cannot be considered as a "stand-alone" conclusion. That is, it depends heavily upon the quality of the detailed work by SAI. This point was made in the original proposal to the County, where it was pointed out that "if major flaws are indentified in the earlier studies, . . . it may be necessary to devote a higher level of effort to this project." Thus the basis for the project has been an assumption that the SAI work under review is a credible effort, up to the standards of the state-of-the-art, and requiring no major upgrading. The

FRA team has attempted to challenge this assumption by carrying out a critical review with the intent to uncover facets of the SAI analysis that might contain inadequate methodology, inappropriate assumptions, errors of omission/ commission, or biases.

1.1 Overall Conclusion

The assumption that the SAI work is basically sound has turned out to be correct, in the opinion of Future Resources Associates, and we believe that the following conclusion is essentially the most important overall summary of the results of our study: FRA concludes that the overall results of the Shoreham PRA, contained in the preliminary draft report by Science Applications, represent reasonable conclusions as to the likelihood and magnitude of releases from large accidents at the Shoreham reactor.

As discussed in the text below, we believe that the likelihood of core melt accidents is somewhat higher, and the magnitude of radiation releases somewhat smaller than found by SAI. However, we believe that the differences should not cause Suffolk County to modify their emergency preparedness activities significantly compared to what they would do by using the SAI results directly as published. Subsidiary to this important conclusion are our conclusions that the methodology used in the SAI study is at the level of the state-of-the-art of reactor risk assessment at the present time, and that SAI's application of this methodology to Shoreham has generally been a competent one.

1.2 Limitations

There are limitations to our acceptance of the SAI study, and these are based on limitations within the study itself. The most important of these study limitations, from the point of view of its applicability to the County's emergency planning needs, is the absence of any analysis of accidents arising from the set of "external initiating events" that could initiate accidents from outside the plant. Most important of these external initiators is earthquakes, with high winds (hurricanes, tornadoes) and floods also of possible concern. In addition, internally-initiated fires are not treated. Because these types of accidents could be important risk contributors, their omission means that neither LILCO nor the County has a fully satisfactory set of important accidents to use as a basis for emergency planning.

There are other limitations to the SAI study, including the appro imations, conservative in nature, as to the fission product source term; the way the very numerous accidents were grouped into classes for ease of treatment in the within-plant and containment phenomenological analysis; the absence of a plant-specific failure data base (since Shoreham has not yet operated); and other issues involving methodological approximations made by the study team. Fortunately, it is our view that none of these limitations, with the exception of the conservative approximations as to the fission product source term, is sufficiently important that it might significantly alter the substance of the County's emergency planning effort. In any event, while we found a few places where the SAI report could be improved, we also believe that many of the report's limitations are not SAI's limitations *per se* but rather are limitations of the present state-of-the-art of PRA generally.

Our own study within Future Resources Associates, unfortunately, has had limitations of its own. The most important of these has been the relatively short time period (about 2-1/2 months) and relatively small level of effort that has been devoted to the project. Although the contract has been sizable from Suffolk County's perspective, the total level of professional effort devoted to reviewing SAI's draft PRA report has been only a few percent of the effort that SAI spent on carrying out the PRA itself and, of course, such a few percent effort cannot reasonably be expected to study every facet of the problem. This limitation is, however, balanced in part by the fact that the review team has had considerable experience in the PRA field, which has enabled the effort to be focused on what are thought to be the main issues.

Another limitation that hampered our group's work during the first month, but was cleared up in mid-July, was a restriction on us that effectively barred direct contact with the SAI PRA study team. After it was lifted, our interaction with the SAI team was a full and open technical exchange of questions and issues, at the level of professional mutual respect that we found refreshingly matter-of-fact. We wish to acknowledge SAI's full cooperation, for which we are grateful.

There are substantial uncertainties associated with the numerical conclusions that SAI quotes in its report. The uncertainties arise from several sources, including the validity of the data base, approximations in the accident sequence fault-tree/event-tree modeling, uncertainties and gaps in our understanding of within-plant accident phenomena, and incomplete understanding of the role of human error and human ingenuity in reactor accidents. Our general view is that the treatment of uncertainty within SAI's study is a reasonable one, including as it does advanced methods for estimating contributions from various sources. We have concentrated on those uncertainties that could particularly affect what Suffolk County might do or decide in the context of its emergency response plan. The discussion in the main body of this report will be in that context.

1.3 Objective of the Report

It is important to state clearly the objective of FRA's review work, which has been to provide Suffolk County with an independent technical opinion as to the probabilities and magnitudes of large potential accidents at Shoreham. FRA's work under Task 1 of its contract has concentrated on ascertaining, through independent technical review, whether the preliminary draft version of the LILCO- supported probabilistic risk assessment carried out by SAI provides a technically sound basis for emergency response planning.

1.4 Organization of the Report

The body of the report to follow will be organized generally along the lines of the scope of work (see Appendix A) for Task 1 of the overall project.

The technical issues that we have covered within the SAI report can be conveniently separated into the following:

- Are all important potential accident sequences considered?
- Are the calculations of the accident probabilities correct?
- Are the accident phenomena within the plant treated correctly?
- Are the magnitudes of the calculated potential radiation releases correct?

The treatment of environmental transport of radioactive materials after release, and their impact on populations, is the subject of Tasks 2 and 3 of this study and is not covered in this report.

1.5 Summary of SAI's Results

Before continuing with the results of FRA's review, it is useful to reproduce here the main results of the SAI study, in tabular and figure form. These, on the following pages, are reproduced directly from the SAI draft report. The first page reproduces SAI's Table 4.1 and Figure 4.1, in which the frequencies of core vulnerable conditions are shown. Of particular interest are the uncertainty ranges of the SAI results. The second page reproduces SAI's Table 4.2, which contains the detailed numerical results for SAI's five accident classes. Core vulnerable frequencies are shown along with contingent probability of core melt, and the characteristics of the releases are also shown. The details of Table 4.2 will be discussed below in the body of our review.

(reproduced directly from SAI's report on Shoreham)

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Table 4.1

SUMMARY OF THE FREQUENCIES OF CORE VULNERABLE CONDITIONS BY ACCIDENT CLASS

GENERALIZED CLASS CLASS	CLASS	FREQUENCY OF CORE VULNERABLE (PER REACTOR YEAR)
Loss of Coolant Makeup	I	2.7E-5
Loss of Containment Heat Removal	II	1.1E-5
LOCA	III	3.6E-7
ATWS W/O Poison Injection	IV	6.1E-6
LOCA Outside Containment	V	2.0E-8
Total Core Vulnerable Frequency (Per Rx Yr)		4.4E-5

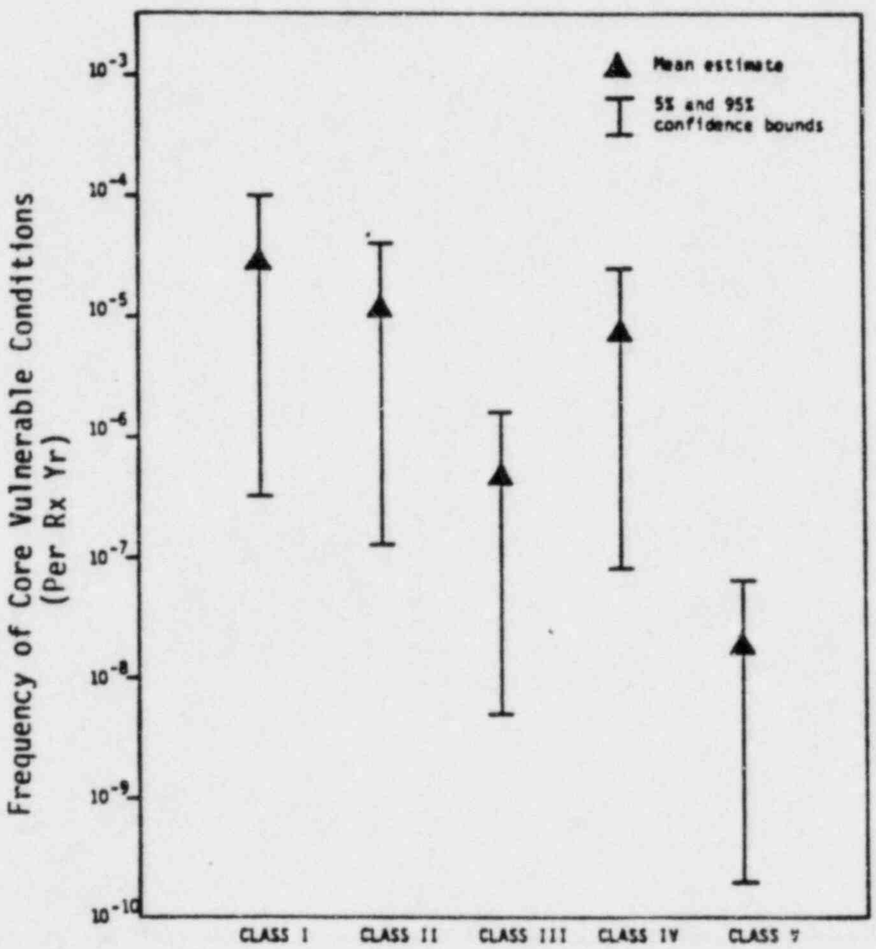

Figure 4.1 Summary of Core Vulnerable Frequencies, Including the Uncertainty Characterization

(reproduced directly from the SAI report)

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Table 4.2

SUMMARY OF SNPS PRA RESULTS, OVERPRESSURE EVENTS

ACCIDENT CLASS	(a) CORE VULNERABLE FREQUENCY	(b) CONDITIONAL FREQUENCY OF RELEASE	(c) RELEASE BOUNDS	(d) ACCIDENT RELEASE DESIGNATOR	(e) FRACTION	(f) TIME OF RELEASE	(g) DURATION OF RELEASE	(h) WARNING TIME FOR EVACUATION	(i) ELEV OF RELEASE	(j) AVG ENERGY OF RELEASE	RADIONUCLIDE RELEASE FRACTIONS: Kr Xe	OI	I[l]	I	Cs[m]	Te[n]	Sr[o]	Ru[p]	La[q]
1	2.7×10^{-5}	8.0×10^{-2}	UPPER	$C1R,T_{\gamma\gamma'}$	.33	2	6[u]	1.5	60	10	.99	3.0×10^{-4}	2.4×10^{-2}	2.1×10^{-4}	3.9×10^{-2}	1.7×10^{-1}	3.9×10^{-3}	2.2×10^{-2}	3.8×10^{-3}
			BASE CASE	$C1R,T_{\gamma\gamma'}$	.34	8	1	7	60	6	.98	2.97×10^{-2}	1.8×10^{-2}	1.5×10^{-3}	5.1×10^{-2}	2.1×10^{-1}	5.1×10^{-3}	2.5×10^{-2}	5.1×10^{-3}
			LOWER	$C1R,'T_{\gamma C}$	.33	8	1	7	75	.3[v]	.15	9.6×10^{-6}	2.0×10^{-4}	1.4×10^{-7}	5.7×10^{-6}	2.4×10^{-5}	5.1×10^{-7}	3.0×10^{-6}	5.1×10^{-7}
2	1.1×10^{-5}	7.0×10^{-2}	UPPER	$C2R,YSFT_{\gamma\gamma'}$	.1	25	20	2	60	50	.98	6.4×10^{-4}	1.15×10^{-2}	3.1×10^{-2}	8.2×10^{-2}	1.5×10^{-1}	3.0×10^{-3}	1.5×10^{-2}	2.3×10^{-3}
			BASE CASE	$C2R,T_{\gamma\gamma'}$	.8	30	20	2	60	50	.98	6.4×10^{-4}	1.14×10^{-2}	3.1×10^{-2}	8.1×10^{-2}	1.5×10^{-1}	2.9×10^{-3}	1.5×10^{-2}	2.2×10^{-3}
			LOWER	$C2R,'T_{\gamma\gamma'}$	.1	30	30	2	60	3[v]	.96	6.4×10^{-4}	3.1×10^{-3}	1.4×10^{-2}	3.1×10^{-2}	3.5×10^{-2}	6.4×10^{-4}	2.5×10^{-3}	3.2×10^{-4}
3	7.[illegible]	8.4×10^{-1}	UPPER	$C3R,T_{\gamma\gamma'}$	.33	2	6[u]	1.5	60	10	.99	6.3×10^{-3}	8.0×10^{-3}	4.7×10^{-3}	4.4×10^{-2}	1.7×10^{-1}	5.0×10^{-3}	2.0×10^{-2}	3.9×10^{-3}
			BASE CASE	$C3R,T_{\gamma\gamma'}$	.34	8	1	7	60	6	.99	6.3×10^{-3}	1.1×10^{-2}	3.8×10^{-4}	4.6×10^{-2}	1.7×10^{-1}	4.2×10^{-3}	2.1×10^{-2}	4.1×10^{-3}
			LOWER	$C3R,T_{\gamma CF}$	.33	8	1	7	75	.3[v]	.49	3.1×10^{-3}	1.5×10^{-3}	2.9×10^{-5}	1.0×10^{-2}	4.3×10^{-2}	1.1×10^{-3}	5.2×10^{-3}	1.0×10^{-3}
4	6.1×10^{-6}	4.3×10^{-1}	UPPER	$C4R,YSFT_{\gamma\gamma'}$	.1	1.5	10[u]	1	60	60	.99	6.4×10^{-4}	2.3×10^{-2}	3.5×10^{-2}	1.0×10^{-1}	2.1×10^{-1}	4.1×10^{-3}	2.2×10^{-2}	3.4×10^{-3}
			BASE CASE	$C4R,T_{\gamma\gamma'}$	.8	1.5	10[u]	1	60	60	.99	6.4×10^{-4}	2.3×10^{-2}	3.5×10^{-2}	1.0×10^{-1}	2.1×10^{-1}	4.1×10^{-3}	2.2×10^{-3}	3.4×10^{-3}
			LOWER	$C4R,T_{\gamma CF}$	.1	1.5	10[u]	1	60	3[v]	.98	6.4×10^{-4}	7.4×10^{-3}	1.3×10^{-2}	2.9×10^{-2}	4.2×10^{-2}	6.9×10^{-4}	4.0×10^{-3}	4.1×10^{-4}
5	2.0×10^{-8}	≈1.0	UPPER	$C5R,Z$[r]	.1	1	1	0	60	6	.99	6.3×10^{-3}	6.6×10^{-2}	6.5×10^{-1}	6.9×10^{-1}	5.9×10^{-1}	1.8×10^{-1}	3.9×10^{-2}	6.1×10^{-3}
			BASE CASE	$C5R,Z$[s]	.45	1	1	0	50	6	.99	5.1×10^{-3}	4.6×10^{-1}	6.6×10^{-2}	6.9×10^{-1}	5.9×10^{-1}	1.8×10^{-1}	3.9×10^{-2}	6.1×10^{-3}
			LOWER	$C5R,Z$[t]	.45	1	2.5	0	75	.3[v]	.94	2.5×10^{-3}	3.5×10^{-1}	4.0×10^{-2}	4.3×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	1.2×10^{-2}	2.0×10^{-3}

FOOTNOTES

(a) Obtained from functional event tree quantification.
(b) Given that the core has become vulnerable, this is the conditional probability that a substantial release occurs.
(c) Releases are treated as distributed with simple discrete model (see uncertainties). Values are shown for the best estimate and two bounding cases.
(d) See Appendix B for discussion of nomenclature.
(e) Treated as a discrete distribution with three values this is the conditional probability of either the point estimate or the bounds.
(f) In hours
(g) In hours
(h) In hours
(i) In meters
(j) In 10^6 BTU/hr
(k) Background on the isotope groups and release mechanisms is presented in WASH-1400, Appendix VII.
(l) Includes I and Br.
(m) Includes Cs and Rb
(n) Includes Te and Sb
(o) Includes Ba and Sr
(p) Includes Ru, Rh, Co, Mo, Tc
(q) Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm
(r) Case where significant deflagration has occurred in secondary containment causing a reducing atmosphere to exist; difference is in chemical form of iodine.
(s) Case where no significant deflagration in secondary containment has occurred.
(t) Extrapolated from data for release from containment terminated after .3 to .6 hr.
(u) Release characteristics are such that an initial puff occurs during the first hour followed by another puff towards the end of the release duration.
(v) Low energy represents slower blowdown after a smaller containment breach.

2.0 Are All Important Accident Sequences Considered?

The answer to this question, as mentioned above in the introduction to this report, is negative. In particular, there has been no treatment of internally-initiated fires, nor of any externally-initiated events, the most important of which are earthquakes, floods, and high winds (hurricanes, tornadoes). These omissions are discussed in the SAI report (P. 1-13), and were beyond the scope of that study. The implications of these omissions will be discussed below, but first we will consider whether any important internally-initiated accidents (besides fires) seem to have been omitted.

On this latter point, we have concluded that the SAI draft report has apparently considered all of what the reactor safety community considers the important internally-initiated accidents. Specifically, we have not found any internally-initiated sequences likely to contribute significantly to the overall risks that have been omitted.

Of course, the unconscious omission of important sequences known to others in the reactor safety community would be quite unlikely in a study of this kind . . . the safety community maintains close enough and open enough communications that any new or unusual accident sequences would almost surely have cometo the attention of the study team or its outside consultants. So our conclusion comes as no surprise.

This observation of the apparent completeness within SAI's draft report does not mean that there are no important sequences omitted. It only means that we are unaware of any, nor do we believe that anyone else in the reactor safety community has thought of any. The strength of our convictions about completeness is based on the general observations that, over nearly a decade of time since the initial probabilistic accident delineation work of WASH-1400 (Ref. 1), there has been hardly any addition to the list of important internally-initiated accidents that WASH-1400 considered. But, on the other hand, who knows whether or when a new sequence will arise, either from operating experience or from analysis?

The grouping into sequences that the SAI team used is discussed in detail in Section 4.1 later in this report. The five classes are described in SAI's Table 3.3.1, which is reproduced on the next page.

It is important to recognize in this connection, however, that the approximately 1600 reactor years of commercial nuclear power operating experience worldwide without a serious accident leading to off-site consequences is a statistically significant data base providing evidence that the result for Shoreham is unlikely to underestimate the probability of serious accidents by a <u>large</u> factor, unless Shoreham is somehow very untypical in its risk profile of the entire group of reactors; of course, it is just this question that is being addressed by the Shoreham-specific risk assessment.

Returning to the known "external" omissions (fires, earthquakes, high winds, floods), the SAI team acknowledged in their report (P. 1-13) that

(reproduced directly from the Shoreham PRA by SAI)

Table 3.3.1

GENERIC ACCIDENT SEQUENCE CLASSES

GENERIC ACCIDENT SEQUENCE DESIGNATOR	PHYSICAL BASIS FOR CLASSIFICATION	SYSTEM LEVEL CONTRIBUTING EVENT SEQUENCE	REPRESENTATIVE SEQUENCE FOR CLASS
Class I (C1)	Relatively fast core melt; containment intact at core melt and at low pressure	Transients involving loss of inventory makeup; small-small LOCA events involving 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 (C2)	Relatively slow core melt due to lower decay heat power; containment failed prior to core melt	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 (C3)	Relatively fast core melt; containment intact at core melt, but at high internal pressure	Large LOCAs with insufficient coolant makeup; transients with loss of heat removal and long-term loss of inventory makeup; RPV failures with insufficient coolant makeup	Large LOCA with loss of low pressure ECCS
Class IV (C4)	Relatively fast core melt; containment fails prior to core melt due to overpressure	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 of SLCS
Class V (C5)	Relatively fast core melt; containment failed from initiation of accident due to equipment failures	LOCAs outside containment with insufficient coolant makeup to core; RPV failures which result in immediate containment failure	LOCA in main steam lines with failure of MSIV closure and loss of ECCS

these were consciously and specifically excluded. The reasons for the omission are probably a combination of two things: first is the fact that treating external events and fires in a PRA is considerably different than treating internal initiators, requiring a different methodology, large additional manpower resources different in character from the rest of the study, and yielding results of even greater uncertainty than the uncertainty in the rest of the PRA analysis. These large uncertainties are because the methodologies are immature and the data base weak. Second, the SAI study team believed when the project began in 1980 that these external events and fires were not as important contributors to risk as internally-initiated accidents, nor as amenable to cost-beneficial risk reduction (Ref. 2).

In the intervening two years, our ability to analyze the risk contribution from fires and externally-initiated events has advanced considerably. Benchmarks of this advance include the recent publication by Pickard, Lowe, & Garrick of seismic and fire PRAs at both the Zion and Indian Point reactor stations (each a two-unit station with Westinghouse PWR's); the completion of an important NRC-sponsored seismic methodology development effort, the Seismic Safety Margin Research Program at Lawrence Livermore National Laboratory; and the inclusion by consensus within the draft PRA Procedures Guide (Ref. 3) of an "acceptable" methodology for PRA analysis of earthquake- and fire-initiated accidents. With these methodological advances have come the first probability-based insights into the quantitative contribution to risk from these sources, albeit with very large uncertainties and important conservatisms in the analyses. To the surprise of some in the reactor community, the Zion and Indian Point studies have told us that neither earthquakes nor fires can be neglected as contributors at those power stations to overall public risk. Their apparent preeminence at Zion and Indian Point is partially because internally-initiated accidents were found to be much less important than was found in WASH-1400.

Fortunately for the purposes of this report, one member of the PRA review team (Budnitz) has recently been reviewing the Indian Point PRA specifically from the perspective of the risk posed by earthquakes and high winds (but <u>not</u> fires). His basic conclusion vis-a-vis earthquake-initiated and wind-initiated accidents at Indian Point is that the methodology is clearly adequate to tell us what types of accident sequences are likely to be of most concern, and whether these initiators pose important safety problems (at Indian Point, they <u>do</u>). But the methodologies do not seem to be mature enough to provide reliable <u>quantitative</u> calculations of the probability of core melt . . . thus any <u>numerical</u> comparison of internally-initiated core melt frequencies or public risks with those from earthquakes and winds is of little value.

An important observation from the Indian Point study is that accidents initiated by earthquakes and winds seem to involve phenomena quite similar to those involved in the ensemble of internally-initiated accidents: that is, the accident phenomena themselves do not seem to form a different set of phenomena that must be considered separately in the sense of comprising different types of releases. If this observation is generally valid, then the main impact on overall PRA results would be to increase the probability of releases already treated in the analysis, with less impact on the spectrum of releases and consequences.

What insights about Shoreham can be obtained by transfer to Shoreham of our recent increased understanding of externally-initiated accidents at other plants? Unfortunately, not much. This is in part because there is not yet available any external-events PRA analysis for any BWR reactor (the PWR's analyzed to date are in detail not remotely similar to the Shoreham design), and in part because the accident-initiating events found to be important at Zion and Indian Point are quite site- and design-specific in detail, involving features that are unlikely to be reflected at another plant.

Much thought has been given by the PRA review team as to why useful insights applicable to Shoreham cannot be reliably gained from studying external-events PRA analyses at other reactors. Our negative conclusion arises basically from our belief that if earthquakes, high winds, or fires give rise to important accident sequences at Shoreham, the sequences themselves are likely to be idiosyncratic to Shoreham, or in some cases possibly generic to BWR's (or BWR Mark II reactors) as a class. Absent any specific analysis, we conclude that the contribution of these sources to residual public risk from Shoreham is simply not quantitatively known, in terms of either probability or character of consequences.

The insight (at the PWR's studied elsewhere) that the accident phenomena are not qualitatively different in kind from those arising from internal sources is, of course, a reasonable one consistent with the intuition of most students of the problem. If this insight were to hold at Shoreham, then emergency response plans based on accident scenarios from internal initiators would likely afford reasonable protection from these other types of accidents as well, provided that the special external circumstances surrounding a large earthquake or hurricane are adequately included in any response plans, but the "if" in this proposition could be a weak reed.

3.0 Are the SAI Calculations of the Accident Probabilities Correct?

To answer this question, the FRA team reviewed the methodology used by the SAI analysts, considered the validity of the numerical data base, and repeated selected calculations to ascertain how sensitive some of the results were. Because the numerical quantification of literally hundreds of sequences, within many different event trees, was beyond our capability, we are not able to affirm the specific validity of each SAI accident sequence. However, we believe that such a detailed review has not been necessary to satisfy Suffolk County's objective.

3.1 Methodology Considerations

The methodology of the SAI analysis includes numerous advances over the WASH-1400 analysis of Peach Bottom in 1973-1974. We concur in the judgments of the SAI team that the use of these advances improves the analysis. An example of the different approach taken at Shoreham is the incorporation of certain important support systems (such as instrument air, AC power, DC battery power) consistently within the fault trees rather than having some of these in the event trees, as in WASH- 1400. This approach allows for easier analysis, but carries with it an increased risk that common dependencies might be missed through oversight if the analyst gets sloppy or forgetful. We believe that SAI's approach is a valid one, which can produce valid results if executed with care.

Another advance is the differentiation in the Shoreham study between accidents leading to a "core-vulnerable" condition and those that proceed further to "core melting." In the WASH- 1400 approach, any accident proceeding as far as what SAI here calls a "core-vulnerable" condition was assumed to proceed all the way to "core melt"; that is, there was no differentiation in WASH-1400 between the two conditions and the more conservative approach was taken. The reason for this was that in 1973-74 there was no existing methodology for consistently calculating the differentiation between these two conditions, and the WASH-1400 study team was not able to develop one within the resources and technical knowledge of the time. Now, eight years later, this differentiation is feasible, and the SAI analysts have developed it further and applied it in the Shoreham analysis.

In our opinion, this differentiation is an advance: it was understood in the WASH-1400 period that the conservative core melt assumption was not correct, and the Three Mile Island accident told us this as well. (The TMI accident, if treated with the WASH-1400 methodology, is called a full "core melt," although it was certainly not.) For purposes of comparison, the probability of "core-vulnerable" conditions at Shoreham is what should be compared to "core melt" in WASH-1400. The numerical results of the Shoreham analysis show that the fraction of "core-vulnerable" sequences that proceed to "core melt" ranges from several percent (for what are called Class I and II events) to 100% (for the Class V category). In Section 4.4 below, we will discuss in detail the question of whether SAI's quantitative conclusions in this area are valid.

Still another methodological advance is in the way uncertainties are estimated. Since the WASH-1400 team pioneered with their analysis, a variety of different methods have arisen for getting a numerical handle on these uncertainties. The approach taken by SAI in the Shoreham study is to use Monte Carlo methods to model how much difference in the final results would arise from various changes at the front end, or in the data base. This seems to be a reasonable approach, and its conclusions are also quite reasonable (see Table 3.8.1 on Page 3-172 of the SAI main report), but we did not review it in detai because of our judgment that the conclusions were reasonable.

Of course, there are still methodological problems that temper the confidence we might have as to the validity of PRA results for frequency of, say, core-vulnerable or core melt states. Among these are a variety of issues in the arena of human factors, and these methodological issues stand apart from issues of human error quantification, about which we will say more below as part of our discussion of the error data base.

A key example of methodological inadequacies is the failure of PRA methods generally to consider adequately the ability of reactor operators to cope, through improvisation and ingenuity outside of standard procedures, with accident sequences as they develop in real time. The coping must surely allow operators to terminate some sequences successfully that otherwise would develop further into high-risk accidents. Yet we do not know how much difference this inadequacy makes to the final results. On the other side of the same coin is the possibility that even well-trained operators, being fallible humans like the rest of us, might significantly aggravate a sequence that in the standard PRA analysis is terminated successfully without damage.

Another methodological limitation in a related arena is the present inability to model control system failures well enough. The event-tree/fault-tree approach intrinsically views accidents through the concept of system functional failures caused by underlying component and support system failures or unavailabilities. In its present state of maturity, the methodology can only incorporate control system failures through _ad hoc_ analysis of when, and how, multiple component system failures might arise from control system failures. While this _ad hoc_ analysis is probably adequate for most sequences, we do not know whether it does acceptably well when "time is of the essence," that is when rapidly developing events, especially in the early stages of some accident classes, could severely tax the operators' capabilities or the resilience of some components.

Time sequencing issues also underlie uncertainties in _event tree delineation_ (and, to a lesser degree, fault-tree delineation). The event trees are written down in a time-ordered fashion, but issues can rise of "which failure occurs first" and of whether recovery might occur later in a sequence. Again, whether this set of issues makes a significant difference to the final results is not known, although our experience leads us to expect that differences are unlikely to be large.

As described in the NRC's PRA Procedures Guide, quality assurance is very important in the generation and analysis of fault trees. For example, if the same event appearing in two or more places in a fault tree were mis-typed when entered into a computer, one can make an error as large as a factor of 100 (usually non-conservatively) in computing accident probabilities. (Such numerical errors would probably show up, of course, in the final analysis.) It is important to note that we did not check SAI's fault trees, nor did we run an independent computer analysis of their fault trees.

None of these methodological limitations is special to the Shoreham PRA analysis by SAI: all are generally shared by other PRA studies on other reactors. Indeed, progress is gradually improving our confidence that these limitations are not important enough to invalidate our confidence in the conclusions of PRA . . . but they do temper our confidence, in an unquantifiable way.

The tenor of the above discussion reveals our overall conclusion about the methodology used by SAI in the Shoreham analysis. That conclusion is that the SAI methodology, though it suffers from some generic limitations common to all reactor safety PRA's to date, nevertheless represents the present state-of-the-art; it includes important methodological advances in some areas. We believe that it is an acceptable basis for estimating the probabilities of the important internally-initiated accident sequences at Shoreham.

3.2 Data Base Considerations

The validity of the failure data base is vital to the validity of any PRA analysis. A couple of issues in our present case represent limitations that might be important. First and foremost, the Shoreham reactor is still in final stages of construction, so we have no data on failures at Shoreham: the failure data must generally come from industry-wide sources. But operating experience has told us that some failure and unavailability rates can vary widely from plant to plant, and we don't know whether Shoreham will be above or below average.

The other side of this coin is that significant advances have been made recently in identifying below-average design, maintenance, and test practices, by study of LERs and plant-specific attention to determining root causes of system and component failures. To the extent that these activities represent improvements, a new plant such as Shoreham can take advantage of this experience to improve its performance over the average performance of plants already running.

Human error quantification is the other arena where data base issues seem to represent a limitation. The SAI study team has used the accepted industry-wide reference for most of its human factors failure data (Ref. 4), but these are widely understood to contain large uncertainties. In particular, the failure data are lumped into broad categories whose applicability to specific sequence situations at Shoreham must surely be only approximate. Also, how Shoreham's operators will behave, compared to industry averages, is of course not known.

Despite the limitations just discussed, it is our conclusion that the Shoreham PRA analysis under review has used state-of-the art data bases generally. We have found some specific cases (see below) where we do not agree with certain specific numerical values, but we conclude that the data used are a generally acceptable basis for estimating accident probabilities at Shoreham.

3.3 Specific Accident Sequences -- An Eclectic Critique

To review quantitatively all of the important accident sequences that SAI quantified in its Shoreham analysis was not possible (and in our opinion not necessary) within the scope of this report. We took the approach of studying what SAI wrote down in their event-tree delineation, and using our experience and judgment to ascertain whether the approach and results were roughly congruent with our expectations. This is what characterized our accident-sequence review.

This activity consumed a reasonable fraction of all the effort spent on this project. In the course of it we found that most of what we studied was unarguably consistent with our experience and understanding, while there were only a few cases to the contrary (albeit important cases). For example, we have already mentioned above that we examined the list of internal accident initiators to ascertain whether SAI had left out any from the list that we would have used, and we indicated that they had not (except fires). We also attempted to look at some of the numerical values that SAI used in order to see whether they agreed with our collective experience. In the course of this review, we found ourselves comfortable with almost all of the numbers we examined, the exceptions being noted below.

Perhaps the most important part of the PRA analysis where we find that our review does not agree with the SAI draft report concerns their analysis of internal flooding (their section 3.4.4.1 and Appendix G). In particular, consider the situation in which portions of crucial safety systems are disassembled for routine maintenance during plant operation. If during this disassembly the valved-off component is accidentally reconnected to its pipe (such as accidentally opening a motor-operated valve that has been closed to allow the maintenance), then release of water through the opened valve will occur. If the mistake is not promptly corrected by re-closing the opened valve, an internal flood can result; such a flood at "Level 8" inside the reactor building can quickly inundate several pieces of critical safety equipment, and its analysis as a special issue was deemed so important that an entire Appendix G is devoted to it by SAI.

A detailed numerical analysis of this issue, by H. E. Lambert of our review team, is presented in Appendix D to this report. We will briefly discuss here the main issues and conclusion. We believe that SAI's analysis contains some errors that underestimate the internal flood frequency, the correction of which would raise the core-vulnerable frequencies for Classes I and II.

The events leading up to a disabling flood are as follows:

Event A: On-line maintenance of some critical system
Event B: During maintenance, the system is disassembled
Event C: Inadvertent opening of isolation valve, causing the flood
Event D: Failure to reclose valve within specified time period
Event E: Operator erroneously isolates the power conversion system during flooding causing the accident because the heat sink is lost

We believe that the initiating event is Event C: the occurrence together of Events A and B defines a vulnerable system state that permits C to initiate the flood. Because Event C is an initiating event, we must compute its frequency, and also the failure on demand of Event D. Event C's frequency (in units of events per year, or the like) must include the pre-existing presence of the vulnerability-inducing states A and B. The units of the events, in the calculation, should be maintenance acts per year (Event A); probability of system disassembly given maintenance (Event B); conditional probability of inadvertent isolation valve opening given maintenance with disassembly (Event C); conditional probability of failure on demand to reclose the opened valve (Event D); and conditional probability of erroneous operator isolation of the power conversion system (Event E), which then would initiate the accident. We believe that SAI has incorrectly used system unavailability for Event A, and that their calculated result is about a factor of 100 lower than the value we obtain, all of which is explained in more detail in Appendix D to this report.

In addition, we believe that the SAI team has used an incorrect value for the Event D failure (probability to reclose the accidentally open valve): SAI uses 0.05 for this probability whereas their primary reference source (Ref. 4) uses 0.25, if one accepts that there will be highly stressful conditions during the period when the operator action will be required (see Appendix D for details).

An important assumption made by SAI is that flooding to the six-foot level will not result in automatic closure of the MSIV's. (SAI does assume, however, that reactor trip will occur.) It is important to verify that the assumption regarding automatic MSIV closure is true. Otherwise the power conversion system is lost and the only normally available coolant makeup system is the condensate system. In this case the accident frequency caused by flooding would increase by an additional factor of about 10, and a design change might be necessary to overcome the problem just discussed.

If our flooding analysis is correct, then the internal flooding accident frequencies are <u>at least of the order of a few times 10^{-5} per reactor year</u>, and as described in our Appendix D could be much higher, depending upon how human error rates are quantified. These accident sequences then become dominant for both Class I and Class II type accidents.

There is another part of the SAI analysis where our review team disagrees with the SAI work. Our difference of opinion enters critically into a key class of accident sequences, the so-called ATWS group (Anticipated

Transients Without SCRAM), which comprise Class IV of the five classes into which SAI grouped the Shoreham accidents. These sequences arise when one of several anticipated transients cannot be properly controlled because the "SCRAM" system (which inserts the control rods) fails to shut down the chain reaction. If this event were to occur, back-up engineered functions are brought into action to bring the reactor to safe shutdown: these are discussed on pages 3-94 ff. of the SAI report and include the alternate rod insertion system; the standby liquid control system to inject boron poison into the core; trip of the recirculation pumps; and operator procedures. Of course, the critical failure that drives this sequence is <u>the failure to SCRAM on demand</u>, for which there are essentially no empirical data upon which to base an analysis.

The SAI report selects as its value for failure to SCRAM on demand the likelihood 3×10^{-5} (about one failure in 33,000 demands). In its discussion (see p. B-111 ff), the report points out that "The calculation of Scram system reliability has been an issue which has taken on both technical and philosophical aspects over the last seven years." This is because within the regulatory arena the issue of whether reactors are adequately safe against ATWS events has been one of the most hotly-contested issues in recent years, whose regulatory resolution is still not complete. Unfortunately, there is also no consensus in the safety community about how to go about calculating the value for this failure nor about which value should be used in analyses such as the Shoreham case.

While admitting that the situation requires a good deal of judgment, we believe that the value chosen by SAI is too high, by a factor in the range of 3 to 10 . . . that is, we would have chosen a value of 1×10^{-5} to 3×10^{-6} instead of their 3×10^{-5}. This judgment, based upon nearly a decade of study of the issue, cannot be defended analytically; but we do believe that the 3×10^{-5} value, which SAI seems to state (p. B-115) is chosen with a conservative bias, is too high. (Of course, we are aware that certain industry calculations fall in the range of 10^{-6}, or sometimes much smaller.)

The impact of this judgment on SCRAM failure probability is practically linear in the results of core-vulnerable and core-melt frequency for Class IV. However, even though our best judgment would lower the likelihood of Class IV accidents, we believe that it is prudent for Suffolk County to base its emergency planning on the Class IV core melt values as quoted in the SAI draft report . . . this approach takes cognizance of the observation that our difference of opinion with SAI is a purely judgmental one.

There were other areas that we reviewed which are important to the SAI analysis. For example, as part of our review we studied the discussion on <u>success criteria</u> found on pp. 1-22 to 1-27 of the SAI draft report. It is clear that in any analysis of accident sequences, whether the outcome is "successful" (that is, leads to an adequately cooled core without vulnerability to core melt) depends crucially on whether the design requires, say, all 4 low-pressure pumps or, say, only 1 out of 4 to cope with a large LOCA. The success criteria chosen by the SAI analysis team have been taken, according to their report, from manufacturer's (GE's) reports and analyses, and the report comments (page 1-22) that "it is believed that the success criteria so defined tend to be conservative." We were not able to assess this claim independently, although it makes some sense to us because the postulated conditions in the GE analysis are limited in a conservatively biased direction. We wish to call attention to this observation as a possible source of uncertainty in the SAI results, which if corrected would lower their calculated core melt values.

We also looked at the way human errors are incorporated into the component and system failures. Here our understanding is that the SAI team has basically relied on the standard data source, the work by Swain and Guttmann (Ref.4), for the human failure data. Since this is the standard work and there is no better source at the present time, we have no quarrel with the SAI team's decision here, but we feel it important to point out that there are very large uncertainties in the values for human failures that Swain and Guttmann quote. Even in the situations in which these values are applicable, there are large variations in the error rates from one human to another, but the stickier problem is that the actual environments in which the human errors will occur in reactor accidents are not those in which the error rates used are necessarily directly applicable. Of course, this is a problem that is generic to reactor PRAs as a class, and not special to the Shoreham effort under review here. But again we wish to call attention to this issue as a source of possible high uncertainty in the SAI results.

Certain other assumptions, necessary in the SAI analysis in most cases for want of any better approach, merit brief discussion. On page 1-30, for example, the SAI report says that "The failure of display of information to the operator is treated as a random independent failure or set of failures and is not dependent on the accident sequence." The limitations in this assumption are clear: that is, independence is obviously not fully true for some sequences, but we understand (and concur with) the reasons why the assumption is made. Also (p. 1-28), "The maintenance contributions in the fault tree model are modeled as mutually exclusive among certain systems consistent with the Limiting Conditions of Operation. An example of this would be that HPCI is not allowed in maintenance if RCIC is unavailable." While this seems reasonable, it is clearly non-conservative to some degree, since LCOs are not always obeyed. Another guideline used in the SAI analysis (p. 1-28) is that "Plant components are presumed to meet all performance requirements consistent with licensing." This assumption implies that, although a component or system might be <u>unavailable</u>, if it is operating it will perform essentially at least as well as the licensing requirement. Again, this reasonable approach contains some (unquantifiable) error.

3.4 Implications of the Review of Specific Accident Sequences

Our review of specific accident sequences described in the SAI draft PRA report has resulted in several specific comments that have been discussed in sections 3.1 - 3.3 just above. Here we will consider the implications of these findings for Suffolk County's emergency planning activities.

Our most important quantitative finding is an apparent error in the SAI calculation of the probability of severe internal flooding: we have discussed this in section 3.3 and in Appendix D of this report. If our interpretation of the frequency of flooding at elevation 8 of the reactor building is correct, then the frequency of core-vulnerable conditions in accident Classes I and II is raised considerably: mostly in Class I in our opinion, with some effect on Class II as well. The SAI report gives as its best-estimate for core-vulnerable frequency in Classes I and II the following:

Class I	2.7×10^{-5} per year
Class II	1.1×10^{-5} per year

The conditional probabilities of core melt are given as 7% and 8% for these two classes, respectively.

If our analysis is correct, these core-vulnerable frequencies would be considerably higher. Because we have not been able to carry out a complete systems analysis, we believe that our choosing a number and sticking to it as "our value" is inappropriate: we believe it is more appropriate that LILCO and SAI carry out a proper analysis. We not only believe that the core-vulnerable numbers will rise, but also that the conditional probability of core melt might increase: to be precise, if flooding dominates Classes I and II, then the flooding sequences should be used as the basis for computing the conditional probabilities, with the possibility that core-melt might be more probable than 7-8% given the amount of equipment taken out of service by a flood at elevation 8.

With these provisos, we believe that it is prudent for Suffolk County to use a number in the range of 10^{-4} per year as the likelihood of core melt from both Class I and Class II: use of this value as input to the planning basis for the County emergency planning activities will prudently allow the County to protect against accidents at Shoreham within Classes I II. Also, because no core-vulnerable/core-melt analysis has been done for these internal flooding sequences, we believe it is prudent to ignore this factor for the time being in these classes. (As it turns out, using these higher values probably makes almost no difference to the Suffolk County emergency planning activities, because Classes I and II dominate the accident planning basis one way or the other.)

For the other accident Classes, the only one where we believe the SAI results are probably not correctly representative of the real accident frequency is Class IV (the ATWS accident group). However, as discussed earlier, even though our best estimate would lower the SAI values for core-vulnerable frequency by a factor of about 3 to 10, we recommend that Suffolk C ty use the SAI values as a prudent planning basis.

Our review revealed some impor ant qualitative conclusions that should be me ned. Most importan of these is that we have found the SAI analysis of a de t seque es and their probabilities to be on the whole a fully competent, state-of-the-art analysis. Also, we believe that there are major uncertainties in the results of these analyses today, not only the analyses by SAI but analyses by all teams that carry out reactor PRAs: we have touched on some of the underlying reasons for these in earlier discussion. Nevertheless, we believe that the SAI group has taken some care to understand and quote the ranges of their uncertainties, and we find their uncertainty analysis for the accident sequences acceptable.

4.0 Are the Accident Phenomena Within the Plant Treated Correctly?

The approach taken by our review team in reviewing the accident phenomena parts of the SAI draft PRA was to ascertain whether the analytical methods employed are reasonably consistent with the present state-of-the-art methods now used for these calculations. Members of the FRA team are quite familiar with the level of understanding within the broader reactor safety community about these phenomena, including a recognition of the significant limitations to our present understanding.

The basic conceptual problem being addressed here is as follows: we postulate the unlikely situation that the Shoreham reactor has reached a physical state in which a core-degradation accident is underway. The accident sequence that has led to this physical state is assumed to be understood, in the sense that the event-tree/fault-tree analysis has revealed to us a specific sequence of equipment failures and human errors leading up to the onset of core degradation. (It is not necessary for the purposes of this part of the analysis to know the probability of the sequence.) The problem is to attain acceptable understanding of the physical sequence of events following the equipment failures modeled in the event-tree/fault-tree analysis, beginning with the onset of core degradation and continuing through until the reactor core is either securely cooled, or has melted (wholly or partially), releasing a fraction of its fission products to the reactor vessel, the containment structures, and possibly the environment.

In order to carry out quantitative calculations of the phenomena, one requires a variety of data, some conceptual understanding of the physical and chemical events, and a calculational model embodying this understanding. The calculational model should incorporate not only the physical reality of the reactor plant itself, but also the functioning (correct or degraded) of various engineered systems within the plant.

In an ideal analysis, it would be desirable to calculate the sequence of phenomena for every important accident sequence that follows core degradation. Unfortunately, though every accident sequence is different from every other one, such a massive effort would be beyond the calculational ability of any team of analysts today. Also, because our understanding of thse phenomena is limited it would not make much sense to model each specific sequence separately: differences among similar sequences are far less significant than uncertainties in our understanding.

For these reasons, PRA analysts generally have grouped the numerous accident sequences beyond core degradation into categories, each category being treated separately in the calculation of core degradation/melt phenomena. This grouping allows the analysts to treat a tractable set of issues, with the limitation that errors and uncertainties are introduced because of the approximate nature of the categorization process.

Each group of accidents is characterized by a set of conditions: examples include high or low pressure within the reactor vessel; containment integrity intact or already failed before the onset of core degradation; time elapsed since SCRAM; various engineered safety systems either available or failed; and so on.

The challenge is to develop an acceptably accurate quantitative analysis of what happens to the core and its fission products, and how, and in what sequence, and under what phyhsical-chemical conditions. The result of the analysis should be a quantification of fission product releases from the plant in terms of species, quantities, times of release, physical-chemical conditions of the release (energy of the release, physical form, etc.), and some measure of the confidence of one's conclusions.

FRA's purpose in this review has been to develop an understanding of what the SAI analysis team has done in the course of their work, so as to ascertain whether their results form an acceptable basis for the calculation of offsite consequences: if the SAI calculations do form such an acceptable basis, then Suffolk County's emergency response plan can utilize them.

The approach taken by the SAI analysis team was to utilize a well-known, widely used, and well-documented set of computer-based models known as the MARCH-CORRAL code package. These codes, originally developed under AEC/NRC sponsorship for the 1973-74 WASH-1400 analysis, have been the main method of analysis of these phenomena ever since. However, it is widely known that these codes contain important approximations, omissions, and other limitations that make their modeling of physical reality less than precisely correct, and indeed the limitations of MARCH and CORRAL have been a continuing subject of research, analysis, and discussion in the reactor safety community for much of the last decade. In addition to the MARCH-CORRAL codes, the calculations of the accident phenomena require various types of input data, the most important of which are the fission-product release fractions or partition fractions in various stages of the postulated accidents. Examples include information about fission product releases from the fuel pins during fuel melting; fission product partitioning in a water/steam environment as a function of pressure, temperature, and other conditions; and fission product plateout, transport, and the like in the airborne state. The MARCH-CORRAL code requires data such as these essentially as parametric input to its modeled calculations.

The SAI analysis team made modifications to parts of MARCH and CORRAL in order to tailor the code to the Shoreham reactor, and to incorporate some recent insights about various phenomena. These modifications do not substantially modify the basic operating philosophy of the codes, nor their major limitations.

4.1 Grouping of Accidents into Classes

One important question that we have looked at is whether the grouping of accidents into five classes by the SAI team is reasonable. The five Classes can be briefly characterized as:

Class I	Loss of Coolant Makeup Accidents
Class II	Loss of Containment Heat Removal Accidents
Class III	Large LOCAs
Class IV	ATWS (Anticipated Transients Without Scram) without Poison Injection
Class V	LOCAs Outside Containment (including Interfacing Systems Accidents)

All accidents leading to and beyond core-vulnerable conditions are placed in one of these five Classes, and the five Classes are dealt with separately in the subsequent (MARCH-CORRAL type) analysis. We have studied the validity of this approximation, and find the grouping by SAI to be fully acceptable. We believe that some differences exist among accidents that are grouped by SAI into a common Class, an example being the variety of accidents grouped under Class II in which containment residual heat removal fails: the very long time that it takes for the Class II accidents to evolve is characteristic of the Class as a whole, but differences in detail emerge depending on which safety systems fail in which order. However, in this case (as generally), we believe that the grouping is a reasonable approach. Our reasoning is that the uncertainties in the phenomena are sufficiently great as to overwhelm any additional "information" that might have been gained by a grouping into more numerous Classes; and that the choice of specific sequences within each Class to serve as the model for each Class is also reasonable.

An important issue is the time duration between initiation of the accidents and the eventual release of fission products to the environment: this will differ from Class to Class substantially. We have studied the general grouping from this perspective, because of a concern that perhaps there could be mis-classification from the perspective of time evolution. We tend to affirm that the time evolution of the Classes is reasonably consistent with our experience: for example, the Class II accidents evolve very slowly compared to the other Classes, and we believe that all of the accidents grouped into Class II behave in this way. This conclusion allows us to deal with all of these accidents together in discussing their emergency-planning implications in the context of Suffolk County's needs.

4.2 Review of Phenomena After Meltdown

The approach we have taken for ascertaining whether the SAI meltdown analysis of each accident Class seemed reasonable was in part the use of subjective judgment based on the experience and knowledge of the reviewers, and in part some independent calculations of a few of the phenomena involved. These independent calculations, carried out by Dr. Fabic, tend to support the validity of the SAI conclusions for many of the important post-meltdown phenomena, with some exceptions that will be discussed next.

A general elevation-view drawing of the Shoreham reactor and containment building is shown on the next page, reproduced directly from the NRC Safety Evaluation Report for Shoreham (Ref. 6). The main important features to

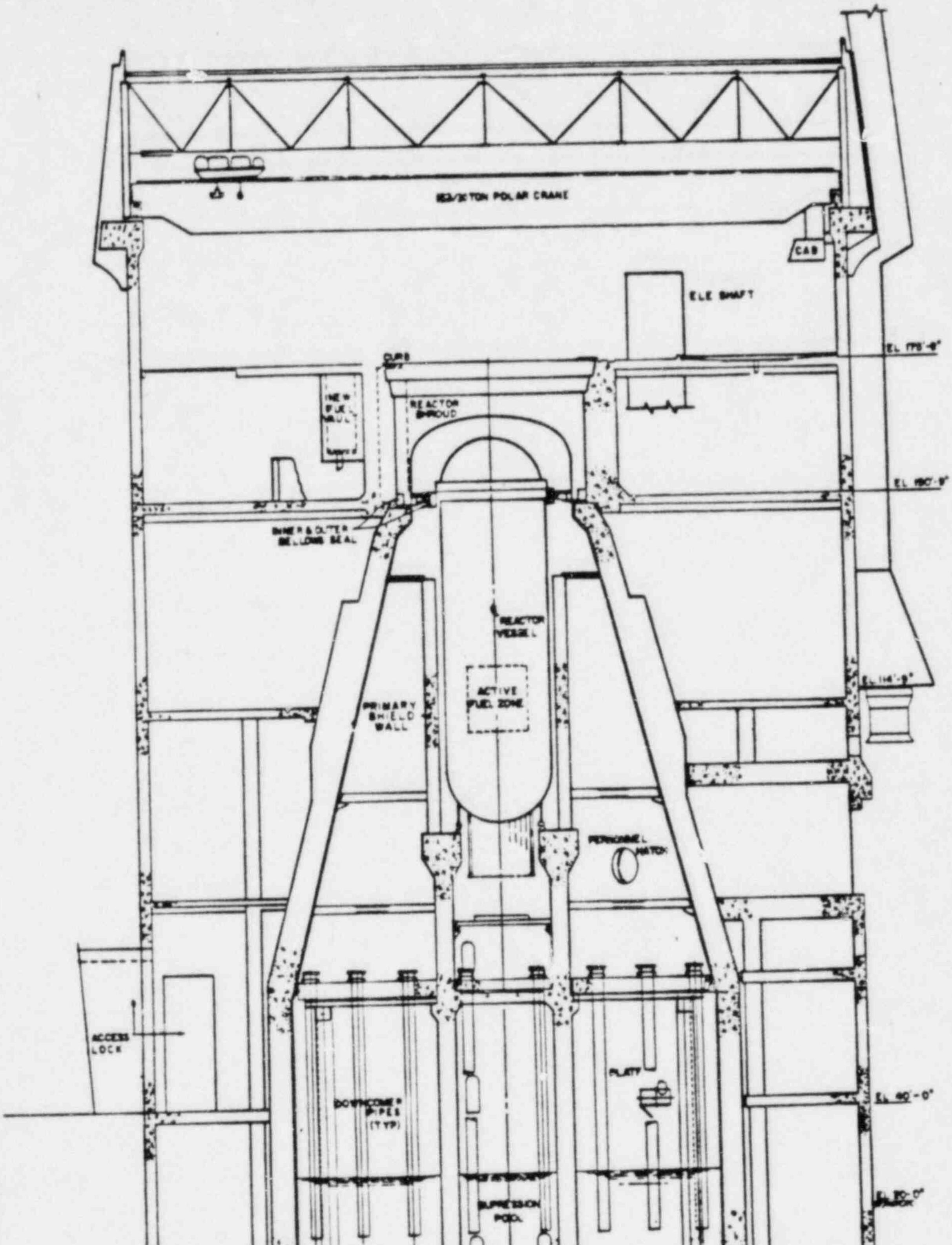

Figure 6-1

Shoreham Mark II Containment
(reproduced directly from the Shoreham "Safety Evaluation Report" by NRC, NUREG-0420)

concentrate on for our discussion below are the location of the suppression pool at the bottom of the building, and the downcomers penetrating the drywell floor below the reactor itself.

That the MARCH code (which models the highly complex processes involved in a meltdown) has deficiencies is, as mentioned above, well known. In Appendix B, some of these are discussed by Dr. Fabic, and his discussion will be excerpted here, as follows:

"In the course of the SASA (Severe Accident Sequence Analysis) calculations being performed by the Oak Ridge National Laboratory (ORNL) for the USNRC as applied to BWRs, the ORNL engineers have also come to the conclusion that MARCH has many <u>deficiencies</u>. Out of 21 listed inadequacies, we shall extract only a few:

* Modeling of heat transfer to upper and lower BWR vessel internals
* Modeling of core collapse
* Failure of bottom head via control rod drive tube penetration not considered
* Suppression pool and wetwell/drywell interaction
* Rod-to-rod radiation heat transfer not included
* Vessel water level calculation does not include variable flow areas
* Fuel pin melt/slump/freeze phenomena are not mechanistically modeled."

Dr. Fabic (Appendix B) also discussed the reasons why the MARCH code contains extensive simplifications:

> "The extensive simplifications in the code were introduced for two reasons: (1) to produce a tool useful for probabilistic risk assessment which requires many computer runs for exploration of consequences of various bounding assumptions. Hence, computational economy must have been one of the principal goals; (2) to produce a tool amenable to accepting some of the major uncertainties as input assumptions that could be changed from run to run. These stem from the relatively poor knowledge of various thermohydraulic processes that involve melt propagation and the attendant heat and mass transfer in complex geometries. The pertinent empirical base lags far behind the existing empirical base collected in the course of reactor safety research for situations that do not involve a degraded core. However, the word 'uncertainty' is also used here to imply simplifications one needs to make to intentionally bypass the detailed calculations that would cause the code to become long running."

Despite these limitations within MARCH, the conclusion of the FRA review team is that the code has been applied by the SAI team in a competent and conscientious manner. Our analysis of their description of their MARCH runs, the way they have thought about issues involved in MARCH, and their conclusions has left us with both a confident feeling about what the SAI analysts have done and a heightened appreciation of the approximations inherent in the MARCH modeling.

One of the important areas of disagreement concerns the mechanism of melt-through of the molten core as it penetrates the bottom head of the reactor pressure vessel (RPV). Again we quote from Appendix B, Dr. Fabic's report (the abbreviation "CRD" means "control rod drives"):

> "We do not believe that the RPV bottom head would fail in either of the two ways described in the Shoreham PRA: structural failure of the vessel wall due to elevated pressure and temperature or the wall melt-through when the RPV pressure is low.
>
> "Instead, we agree with the scenario described by R. Henry of Fauske Associates, Inc. (see item 8 in Section 1.1) wherein the relatively thin metal that seals the CRD tube, or the CRD tube itself, is much more likely to fail first upon contact with the melt.
>
> "The melt is very unlikely to reach the bottom head in a coherent fashion (gross slumping). It is more plausible to consider downward streaming of melt around the vessel axis. That melt will not attack the CRD tube as long as there is some water left in the lower plenum. If the amount of steam generation caused by quenching of melt is insufficient to stop further melting (a likely case),

increased amount of debris will accumulate on the bottom portion of the RPV and it would eventually remelt the fraction of debris that was frozen by the liquid--which by now has evaporated. This whole process could be delayed significantly if the CRD cooling water were continuously supplied.

"Eventually, one or more CRD tubes would fail and the debris discharge into the CRD room would commence."

Following the failure of the CRD tubes, another key question is how fast the melt will pour or stream through these CRD channels. Dr. Fabic calculates (Appendix B) that under the conditions likely to be present, only a very few seconds (perhaps less than 5 seconds) would elapse before as much as 50% of the melted metal would flow out of the lower vessel openings. In this scenario, there is no outright structural failure of the vessel wall, nor wall melt-through per se. However, as the melt penetrates the CRD tubes, significant enlargement of these penetrations will occur by ablation. If this is true, then only a small fraction of the melted core will remain in the bottom head, and that fraction will be the core material that has not melted, or has re-frozen at the bottom. Whether the flow through the CRD openings is pressure-driven or gravity-driven will make a difference, but should not make a very large difference to the phenomena in this process, which will be quite fast.

The next issue involves where the core debris streaming through the CRD openings will go. With a large amount of CRD hardware in the way (essentially hanging down in large massive metallic drive channels below the lower head), and with such a fast outflow, it seems unlikely that the CRD hardware will melt before the core material has passed through, which in turn indicates that the core material will reach the drywell floor only after being significantly broken up and "sprayed about" by the CRD hardware. However, an important insight (see Appendix B) is the conclusion that most of the core will go down into the drywell area just below the reactor pressure vessel (RPV) head, with rather little of it going into the annular outside region of the drywell floor. Once the level of core melt debris on the drywell floor exceeds the height above the floor (about 3.5 inches) of the downcomer top flanges, the molten debris will start pouring down the four downcomers that are located within the area of drywell floor just below the RPV: we find that it is unlikely that much of the melt will go down the numerous downcomer channels in the outer annular region, since the melt will go down these four inner downcomers promptly. Our analysis indicates that this will occur rather quickly, most likely within about 5 minutes after RPV breach.

Once the debris begins pouring down the downcomers into the deep wetwell, we studied whether steam produced by the molten material would produce enough countercurrent flow upward to impede further melt from penetrating downward. Our calculations (see Appendix B) seem to indicate that this phenomenon will not occur.

In the scenario described above, almost all of the melt will end up in the wetwell almost all of the time. Appendix B discusses why we are reasonably confident of this conclusion. It appears that there is only a limited range of conditions where melt will not mostly be quenched in the wetwell. The implications of this quenching are, of course, that significant removal of fission products in the wetwell water would occur, lowering by large factors the amount of fission products other than noble gases available in the gaseous phase for atmospheric release through an ultimate containment breach.

Perhaps the most important conclusions from Dr. Fabic's analysis in Appendix B are that vessel failure will probably take place through the CRD channels; that discharge from the vessel to the drywell will be very rapid (less than 1/2 minute), and discharge from the drywell to the suppression pool will also be fast (less than, say, 10 minutes); that core melt material will nearly always end up in the wetwell, where significant removal of fission products will occur; and that there is little time for debris-concrete interaction on the drywell floor, and too low a temperature for that interaction to occur on the wetwell floor.

All in all, the picture painted by our analysis is that the SAI results on core-melt behavior and fission-product release are likely to be conservative (that is, too high values for release, too much core-concrete interaction) compared to what we believe to be the real phenomena.

In particular, we expect that for two of the classes of accidents (Class I* and Class III**), there will be significant delay in the containment rupture time, and perhaps no containment rupture at all, depending on specific containment heat-leakage properties that we have not specifically addressed and that are both complicated and scenario-specific. Also, there will be quite large decontamination factors for fission products released in the wetwell pool, leaving the gaseous fission products released directly from the melted fuel pins as the principal airborne species available for release from the containment after breach.

Unfortunately, we are not able to quantify the extent of this conservatism. The phenomena that we believe will occur differ from those modeled in the SAI draft report in several areas, but almost always in the "conservative direction," meaning that if our analysis is a better representation, then the accidents will be less severe. We believe that much remains to be done before a reliable calculation of these phenomena can be carried out: some research on specific phenomena must be performed; some improved modeling of the sequence and character of the events must be accomplished; and some better data on specific coefficients must be obtained, for a variety of physical and chemical interactions.

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* Class I involves accidents where failure of core cooling with the RPV at high pressure after a transient or small-break LOCA results in core degradation, with an intact containment until after core melt.

** Class III involves accidents such as a large-break LOCA inside containment where RPV depressurization occurs prior to core degradation, with an intact containment until after core melt.

4.3 Some Specific Issues Involving Meltdown and Related Phenomena

In the course of our review of the meltdown part of the draft PRA on Shoreham, a number of specific issues arose that we believe are worth discussing within this report. These are technical issues that could affect the overall "bottom line" conclusions of the SAI study. For each we will indicate what our own conclusions are as to the impact of the issue vis-a-vis Suffolk County's application of the PRA results for emergency response planning.

The first issue is the magnitude of fission product releases from the fuel. In Table D-3 (page D-9) of the SAI draft report, a list is given of release fractions from the fuel to the primary reactor system during core meltdown. The table is reproduced on the next page. Shown in the table are two values for the release fraction for each isotope: one is the value used in the 1973-74 WASH-1400 analysis (Ref. 1), and the other is the more recent value found in NRC report NUREG-0772 (Ref. 5), which contains a discussion of recent understanding of fission product behavior in core-melt accidents. The SAI report states (p. D-7) that they have used an average of the two model estimates (WASH-1400 vs. NUREG-0772), in view of uncertainties in our knowledge. While this averaging does not make very much difference to most of the release values, the value for tellurium is quite different in the two models: WASH-1400 assumed that only 15% of the tellurium escapes from the fuel, while NUREG-0772 believes the correct value is 100%. While we do

:uced directly from the Shoreham PRA by SAI)

Table D.3

RELEASE FR TO REACTOR PRIMARY SYSTEM DURING CORE MELTDOWN

Fission Group	RSS	NUREG-0772
Xe, Kr	.9	1.0
I, Br	.9	1.0
Cs, Rb	.81	1.0
Te	.15	1.0
Sr	.10	.3
Ba	.10	.5
[illegible]**	.03	.02
Nt	.003	.03
L[illegible]	.003	.003

* Inc s Sb, Se

** Inc s Mo, Pd, Rh, Tc

† Inc s Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu

NOTE: :S" is "Reactor Safety Study", Report WASH-1400 not know which is correct (that awaits further experiments that have not yet been done), we believe after study of the NUREG-0772 arguments that the lower figure is more likely to be correct for tellurium, which would slightly decrease the offsite releases and doses. The effect is a small one generally, but not so small as to be negligible.

SAI's draft analysis also states (page D-7) that 90% of the iodine is released from the primary system as CsI (cesium iodide), which is a major departure from the WASH-1400 assumption that iodine was released 100% in elemental form. This makes a significant difference because the CsI is soluble while the elemental iodine, in gaseous form, was a key contributor to offsite doses in WASH-1400. We tend to agree with the recent arguments that elemental iodine is not a likely form for that element in these accidents, and believe that SAI's assumption is a reasonable one. However, there are still important uncertainties in our knowledge of what precisely happens to the iodine during the accidents under consideration. The reactor safety community will require some experiments, some modeling, and considerable discussion (all of which is underway) before this important issue is resolved and a consensus reached on it.

Another issue involves whether there would be zirconium-water reactions within the reactor vessel after the fuel has melted but before the melt penetrates the bottom head: SAI assumed (page C-23) that in a melted state the Zr-water reaction would not occur, and has explained their assumption (Ref. 2) as being due to the presence of a molten pool at the bottom of the lower vessel head, covered by a crust at the interface between molten and solid fuel. This crust would inhibit Zr-water interactions, according to the SAI opinion. While this scenario seems reasonable, the generation of zero Zr-water reaction seems too extreme, because surely some of the zirconium will be in contact with water or steam during parts of the meltdown process. This issue needs more analysis, in our view. We have no way of quantifying whether the Zr-water reaction is miniscule or only "small," but we do concur that it is unlikely to be "large" in the sense of comprising any reasonable fraction of the total zirconium in the fuel cladding. Thus we do not think this issue should have any important impact on the overall results of SAI's analysis.

Another issue that we have studied is whether there will be rapid degradation of the concrete walls of the drywell during core-melt accidents, which might lead to carbon-dioxide evolution, high pressures building up inside, and ultimate pressure failures of containment. First, we should point out that in our own analysis, almost all of the melt goes directly from the bottom vessel head to the CRD room floor to the suppression pool; almost none goes to the annular drywell floor (see Appendix B). However, for this discussion we will assume that we are considering melted core on the drywell floor. When our concern with the issue of drywell concrete degradation was raised with the SAI analysis team, their response (Ref. 2) was that only for the Class III sequences would a sufficiently high temperature occur in the drywell, and these sequences produced very short thermal transients. If this is true, the decomposition temperature would be reached only for a very thin layer near the surface of the concrete. We have not been able ourselves to calculate any sufficiently detailed numerical values for the temperature transients, nor can we find them in the draft SAI report. However, based on our experience with this issue, and our understanding of the phenomena that will occur during melting and core transport from vessel to drywell to wetwell, we believe that this concern is not likely to be a major one: specifically, we believe that even if significant core melt were to reach the drywell annulus, it is quite unlikely that this concrete-degradation effect will produce enough non-condensible gas to challenge containment earlier than it is challenged in the scenarios set down by SAI in their report.

We have also been concerned that containment integrity might be compromised by pre-existing containment leaks, that might not have been considered properly in the SAI analysis. Also, we were concerned with failures of penetrations at high temperatures. While SAI confirmed (Ref. 2) that this latter problem was not explicitly considered in the PRA, they believe that the former is adequately analyzed within the containment event trees. We are not convinced that the analysis incorporates either of these issues quantitatively; however, we believe that containment leakage is not as important an issue for the Shoreham BWR as it appears to be for many other reactors (in particular, for many of the PWRs), because the effective fission product removal action of the suppression pool provides such large reductions for the most likely accident classes. Thus the issue probably does not introduce major additional pathways to the environment even if it has not been thoroughly considered in the SAI analysis.

One of the key issues that has been an outstanding concern for many years in consideration of core-melt accidents is "debris-bed coolability," which is a shorthand phrase for the question of whether, if a bed of core rubble were to form from molten-then-solidified core material, that bed can be cooled adequately. Another question is whether steam, generated in and around the debris bed as it is cooled, can provide enough additional pressure to pose a challenge to containment. First, we must clarify that we are concerned here with cooling a debris bed on the drywell floor; we have no concern with cooling the later debris bed under many feet of suppression pool water, although ultimately after many days the pool water would boil away and the issue would arise again. The SAI response to our inquiry on this subject (see Appendix C) is reasonable, and is congruent with our own judgment on the matter. However, it must be emphasized that while our judgment and that of the SAI group agree, the entire subject of debris beds is still one where everybody's conclusions are highly speculative; in our view the safety community as a whole doesn't have enough experimental data, nor modeling talent, to put this issue to rest at this time. Fortunately, the issue is not believed by our group to pose significant additional risk potential, for the following reasons which are quoted here from Mr. Davis' discussion in Appendix C:

> "Upon further consideration of this issue, it is considered not to be significant in terms of the potential for risk increase for the following reasons:

"1.If containment failure has not occurred at the time debris bed cooling occurs, most of the fission products (which are released during the initial meltdown phase) will be securely trapped in the suppression pool water. Thus, the possibility of containment failure from a steam pressure surge upon debris bed cooling would not cause a large fission product release.

"2.If containment failure occurs before debris bed cooling, the major consequences of the accident would be underway, and the added fission product release would likely not be significant.

"3. A steam pressure surge sufficient to challenge containment integrity requires a large amount of water delivered to the bed and intimate mixing. It is not likely that a high volume water source would be available since most such sources within the containment must previously have been asssumed failed or degraded to cause the accident to progress through core meltdown."

4.4 Core Vulnerability vs. Core Melt: Containment Event Trees

In section 3.1 (above), we commented favorably on the methodological advance employed by SAI in the Shoreham study in which an accident that proceeds to a "core-vulnerable" state is differentiated from one that proceeds further to a "core melt" state. As discussed earlier, this differentiation was not made in the WASH-1400 analysis, and yet it is clearly important to recognize that only a fraction of "core-vulnerable" accidents proceed to "core melt," an example of one that stopped short being the Three Mile Island accident of 1979.

The SAI report finds, for the five Classes of accidents considered, the following conditional probability that a core melt will follow:

Class	Conditional Probability of Melt, Given Core Vulnerable State
I	8%
II	7%
III	84%
IV	43%
V	100%

The way these conditional probabilities were calculated in the Shoreham PRA analysis represents an advance over earlier probabilistic reactor analyses. The effort consisted of the development of very complex containment event trees (CETs) that considered the large variety of engineered safety features and phenomena that are brought into play during the time period after a core-vulnerable condition is reached. Of course, some assumptions are necessary to simplify the problem, and the SAI analysis team made several of them. We have studied them in the SAI draft report (see the list on p. 1-19, pages H-11 to H-13, H-48 to H-56, and Table H.7), and find them generally reasonable, although the analytical basis for the specific numbers is weak.

We believe that there are major uncertainties in the CET analysis, but our study of the SAI discussion leads us to believe that the SAI analysts were aware of these and handled them acceptably. For example, a key limitation is that there are essentially no data on the relative likelihoods of containment failure modes of different sizes and types that would result overpressure (see page H-50). Another example is that, although the analysis correctly differentiates between containment overpressure failures near the top as opposed to near the bottom of the containment (a failure near the bottom, although very unlikely, could compromise the suppression pool), their treatment is obviously an approximate one.

We believe that the point of departure of the SAI treatment is proper. They start by differentiating among three situations: those in which the core is vulnerable to melting in an intact containment; those in which containment may be vulnerable first while the core is still adequately cooled; and those involving containment bypass. These three topologically different states are then treated separately in the quantification process.

We also endorse the approach taken in the draft SAI report to consolidation of the release end-states. This consolidation has the effect of grouping numerous different accidents and treating each group singly, which inevitably implies loss of detail in the interest of calculational tractability. The SAI report acknowledges this issue (p. H-51), and states that the approximations made are conservative in nature. We affirm that this is probably correct but have not reviewed the details sufficiently to have an independent view of whether this is always true (see Table H.8, p. H-106 for details).

To mention other technical issues, we believe that the treatment of the steam explosion issue is a reasonable one. Also, the discussion on quantification of the CETs recognizes explicitly some concerns that clearly affect the quality of the results (see Table H.6, p. H-57 to H-60). The SAI analysts have explicitly differentiated among various qualities of their supporting data, have given not only their best point estimates but 10%-90% bounds for their results, and have documented their main assumptions. Again, however, as elsewhere in this review there was no that our review effort could examine the (literally, hundreds of) detailed numbers in the CET quantification.

Our failure to review these conditional probabilities quantitatively is not very troublesome to us, because we are of the opinion (a qualitative opinion, however) that the values quoted are reasonably within the range that we would expect. Furthermore, we believe that the inclusion or exclusion of the SAI conditional probability factors does not make any important difference to what Suffolk County will do in its emergency planning activities. Therefore, and because we are uncertain as to the quantitative validity of the results, we believe that it is prudent for the County to ignore these factors in its planning, and to take the core-vulnerable figures, as modified by our recommendations for Classes I and II, as the planning basis for emergency preparedness.

Our rationale for recommending the exclusion of these factors is as follows: for Classes I and II, we have already recommended that the County assume that the core-melt values lie in the range of about 10^{-4}/year, taking into account our improved analysis of internal flooding; for the internal-flooding sequences there does not exist a valid core-vulnerable/core-melt analysis, and we believe that our recommended numbers are in any event only a rough estimate. For Classes III and V, the SAI factors are so close to unity (84% and 100%, respectively) that there is no difference. For Class IV, the SAI factor is 43%, within about a factor of 2 of unity, but for this class we believe that the core-vulnerable number could be too high by a factor of 3 to 10 because of SAI's use of too high a value for failure of SCRAM on demand: so for Class IV the inclusion or exclusion of a 43% factor is practically like splitting hairs.

For all these reasons, we believe that omitting the core-vulnerable/core-melt factors from Suffolk County's planning basis is the prudent choice: the factors have no reasonable basis for the internal-flooding sequences and make essentially no difference in the other Classes.

4.5 Implications of the Review of In-Plant Phenomena

Our review of in-plant accident phenomena described in the SAI draft PRA report has resulted in a collection of specific comments and remarks that are covered in the earlier sections (4.1 - 4.4) of this chapter. It is important to describe the context in which these comments are to be understood. We believe that the present state-of-the-art of probabilistic analysis of in-plant phenomena is not very well advanced. In particular, our underlying understanding is inadequate for some phenomena (for example, core melting itself, core penetration of the vessel, debris bed formation and coolability, aerosol plateout, core-concrete interactions); for the performance of some key systems (for example, containment failure mechanisms, effectiveness of suppression pool heat removal mechanisms, efficacy of active aerosol removal systems); and for the time sequence and duration of some events during the accident (for example, duration of meltdown itself, pressure buildup).

Given the paucity of experimental information, only limited applicability of the data that do exist, and calculational intractability of models complex enough to contain detailed differential effects, it is no surprise that differences of opinion exist within the professional community. Some of these differences are reflected in our comments above. What is important to leave with the reader of this report is that we do not believe that there are important differences between what SAI has accomplished and documented and our own view about release magnitudes: differences of interpretation, differences in level of detail, differences in selecting experimental data or modeling approximations seem in every case to produce effects on the final PRA results that are within the quoted uncertainties of the SAI analysis.

If there is one overriding impression that our review team is left with in the aftermath of this in-plant phenomena review effort, it is that the magnitudes of the radioactive releases are likely, in actual accidents, to be smaller than are calculated in the SAI draft report. This is due to the inherent introduction of numerous conservatisms in the analysis. Paramount among these are conservatisms in the description of removal mechanisms that will keep important fission products within the reactor building, for which incomplete credit has been taken in the analysis. Examples include the likelihood that bottom-head penetration may not even occur at all for some scenarios, either because melting will be incomplete, or heat transfer larger, or CRD cooling water available (see our Appendix B); the analysis of suppression pool decontamination factors (see our Appendix C); and the various assumptions that SAI has made on containment failure mechanisms themselves.

We are unable to quantify the extent of these conservatisms in the in-plant phenomena analysis; indeed, we believe that it will be several years from now before enough research has been carried out to enable a consensus to be reached on these issues. For this reason, we believe it imprudent to take account of them for the purposes of advising Suffolk County's emergency preparedness effort. However, it is important that the reader understand our view that the SAI radiation release results, taken at face value and considering their large quoted uncertainties, are more likely to be too high (that is, to represent accidents more severe than actual) than too low.

5.0 Summary and Conclusions: Are the Probabilities and Magnitudes of the Calculated Radiation Releases Correct?

In earlier sections we have already given our summary conclusions as to the correctness of the probabilities of the accident sequences (Section 3.4) and the magnitudes of the calculated releases (Section 4.5). Here we will summarize our findings, and discuss their implications when applied to Suffolk County's emergency planning activities.

Concerning the SAI calculation of accident probabilities, we have found two important differences between our analysis and the SAI analysis. They were noted in Section 3.4, as follows: First, we believe that due to an error in the SAI treatment of internal flooding, the contribution of internal flooding to Class I and II accidents has been underestimated. While we are unable to provide our own analysis of the internal flooding accidents, we recommend that Suffolk County use, as a basis for their emergency planning effort, values of about 10^{-4} per reactor year for the frequency of core melt for both Classes I and II. Second, we believe that Class IV accidents will occur less frequently than the SAI report claims, because we believe the SAI report has used too high a value for the failure on demand of the SCRAM system. However, we recommend that the County should use the SAI results for Class IV as their planning basis. We recommend that the County use all other values for core-vulnerable and core-melt frequencies directly as found in the SAI draft report.

Concerning the SAI calculation of magnitudes of radiation releases from Shoreham, our review has left us with the strong impression, which is supported by the discussion in our Chapter 4, that the releases calculated in SAI's study are more likely to be too high than too low (that is, we believe that their calculation has conservatisms within it). We have been unable to quantify the magnitude of these conservatisms, and in fact we do not believe that it is possible to do so at the present time, because of insufficient information about the phenomena that characterize accidents of the type under discussion. However, we believe that the conservatisms could amount to large factors of reduction for some accident types, particularly those important accidents (albeit, in an absolute sense quite rare accidents) where the melted core would pass into the Shoreham pressure suppression pool. In our opinion the suppression pool, and the way the Shoreham design provides for prompt passage of melted core material to it in these unlikely accidents, will be very effective in fission-product removal. The substance of our remarks here is our opinion that the removal effectiveness of the pool may be even greater than the SAI analysts have used in their calculations.

Thus we arrive at the following overview of our conclusions as to the probabilities and magnitudes of SAI's calculated radiation releases from accidents at Shoreham. We believe that the Class I and II accident groups have higher probabilities than are found in the SAI report, because of internal flooding accidents; and we believe that the magnitudes of the releases are likely to be lower than SAI has calculated.

The implications for Suffolk County's emergency planning effort will depend upon what the County planners decide to do with our findings. We believe that the implications for the County of higher core-melt probabilities in Classes I and II would be minor because these classes already dominate the composite offsite doses from Shoreham, even if SAI's lower numbers are used. If the County planners use the composite dose-distance curves as their planning basis, increasing the absolute probabilities of these curves changes nothing that the County will do or decide. Concerning the magnitudes of the releases, use of substantially smaller releases would have a major effect on the County's planning effort; however, we recommend that the County utilize the SAI draft results as their planning basis because we cannot quantify the degree of conservatism in them.

6.0 ACKNOWLEDGMENTS

The authors wish to acknowledge the advice and assistance of Fred C. Finlayson, our collaborator on this project without whose help the work could not have been accomplished. Fred has also served in a liaison capacity with Suffolk County throughout this work. We also acknowledge Christopher McMurray, counsel to Suffolk County, for much assistance. We have benefitted from important interactions with Robert Erdmann and Edward Burns of Science Applications, Inc. and Voyin Joksimovich of NUS Corporation, who have explained parts of the SAI report to us. Finally, our interactions with Suffolk County itself, through Frank Jones and Patricia Dempsey, have been smooth and successful throughout.

7.0 References

The primary reference used in this report is the probabilistic risk assessment carried out by Science Applications, Inc. on the Shoreham Plant. The reference is:

Science Applications, Inc. (San Jose, California), "Probabilistic Risk Assessment, Shoreham Nuclear Power Station, Long Island Lighting Company," Preliminary Draft Report SAI-001-83-SJ, March 1982

The following are referred to by number in the text:

1. U. S. Nuclear Regulatory Commission, "Reactor Safety Study," Report WASH-1400, October 1975

2. Letter, E. T. Burns (of Science Applications) to R. J. Budnitz (of Future Resources Associates), dated 2 August 1982, on the subject "Responses to questions generated during July 16, 1982 meeting"

3. "PRA Procedures Guide, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," prepared by the American Nuclear Society and the Institute of Electrical and Electronic Engineers, Report NUREG/CR-2300, Review Draft, dated April 5, 1982

4. A. D. Swain and H. E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," Report NUREG/CR-1278, April 1980

5. U. S. Nuclear Regulatory Commission, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," Report NUREG-0772, June 1981

6. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Shoreham Nuclear Power Station, Unit No. 1," Report NUREG-0420, April 1981

DRAFT

REVIEW AND CRITIQUE OF PREVIOUS PROBABILISTIC ACCIDENT ASSESSMENTS FOR THE SHOREHAM NUCLEAR POWER STATION

Report on Task 1 of the Project

"Consequence Assessment for Suffolk County Radiological Emergency Response Plan"

VOLUME II: APPENDICES

Prepared for

Radiological Emergency Response Plan
Steering Committee

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EXCERPTS FROM THE CONTRACT BETWEEN
SUFFOLK COUNTY AND
FUTURE RESOURCES ASSOCIATES

APPENDIX A

EXCERPTS FROM CONTRACT BETWEEEN
SUFFOLK COUNTY
and
FUTURE RESOURCES ASSOCIATES, INC.

3. SCOPE OF WORK

A meaningful assessment of consequences of potential accidents at the Shoreham nuclear power plant requires: (1) assessment of the probability of severe accidents for the facility; (2) determination of potential scenarios, time sequences, and magnitudes of releases for the associated accident categories of the plant; (3) analysis of the characteristics of accident generated radioactive clouds and the local meteorological impacts resulting from their transport beyond the facility boundaries; (4) assessment of public health consequences resulting from the accident-generated cloud transport depending upon the level of protective actions taken by the affected population. The program outlined below identifies a proposed approach to assessing the potential consequences in accordance with the above outlined requirements. A detailed outline of the Tasks to be performed, as described below, is presented in Appendix A to this proposed scope of work.

3.1 Task 1. Review and Critique of Previous Studies of Probabilistic Accident Assessments for the Shoreham Facility

The review will include a critical assessment of accident sequences defined in the earlier, LILCO-supported SAI study of the probability of accidents at Shoreham. The projected probabilities for accident sequences defined above will be reassessed. The phenomena associated with fission product releases within the reactor containment structure will be reviewed. The factors used by SAI to estimate potential reductions of fission product quantities released from the molten core as a result of the effects of engineered safety features and other physical phenomena taking place within the containment structure will be assessed. Realistic estimates (from an emergency planners point of view) will be made of fission product behavior within the containment structure prior to releases to the environment. Containment failure mechanism modeling will be examined and reevaluated and quantitative descriptions of fission product release characteristics to the environment will be established as part of the requisite input parameters for Task 2.

3.2 Task 2. Review and Critique of Previous Radioactive Plume Transport Modeling and Radiological Consequence Assessments

This review will include an analysis of earlier LILCO-supported PL&G studies within this topical area and a reevaluation of the critical parameters associated with the transport of radioactivity and consequent public health effects. An assessment will be made of the results of plume transport modeling and resultant radioactivity concentration studies. Critical factors used in the studies for the evaluation of health effects from the radioactive transport will be assessed. Estimates of population distribution for both current demographic features and projections of future distributions will be evaluated. The meteorological models used in the PL&G analysis will be reassessed and the potential applicability of the models to provide insight into peak public consequence conditions that influence the shape of probabilistic consequence curves will be analyzed. Protective action procedures and models will be analyzed and Long Island's unique terrain related influence on public health effects assessed.

3.3 Task 3. Performance of Site Specific Consequence Analyses and Assessment of Protective Action Effectiveness

Independent calculations of public health consequences will be made with the CRAC-2 code to assess the validity of LILCO-supported analyses. Preliminary calculations will be made with input parameters similar to those used in the utility-supported studies in order to develop an understanding of differences in results induced by differences in numerical methods used in the proposed studies and those used in the earlier analyses. Following the preliminary review of utility-supplied results, site and emergency response plan-specific analyses will be made to determine the consequences of reactor accidents if only minimal protective actions are taken. Other calculations will be made to estimate the plan-specific effectiveness and support the development of potential protective actions proposed for utilization by planners developing the County's Radiological Emergency Response Plan. An assessment will be made of the ranges of uncertainty associated with the results based upon the uncertainties of selected critical variables that may strongly influence public health impacts of nuclear power plant accidents.

3.4 Task 4. Project Integration, Documentation, and Technology Transfer

Support will be provided to assure integration of the consequence analyses with the activities of PRC-Vorhees, the County's contractors responsible for developing its radiological emergency response plan. Local issues and concerns will be assessed and integrated into the consequence analysis through support provided to the Radiological Emergency Executive Steering Committee and Suffolk County officials. Support will also be provided to the Steering Committee and County officials in presentations explaining the methodology and results of the Consequence Analysis studies. Internal integration of the program will be provided to assure that portions of the study conducted under the direction of Dr. R.J. Budnitz and those conducted specifically under Dr. F.C. Finlayson's direction are properly coordinated. Overall supervision of the project will be provided by Dr. F.C. Finlayson. Documentation for the Consequence Analyses will be provided in a final report. The report will contain a presentation of the significant data derived in the study in a format useful for planners. A review and analysis of the data will be presented in the report. Critical issues associated with the results will be identified and assessed in terms of the results that have significant impact on the County's emergency response planning for public risk reduction from potential accidents at the Shoreham nuclear power plant.

APPENDIX A. DETAILED OUTLINE OF PROPOSED SCOPE OF WORK

Task 1. Review and Critique of SAI's PRA Treatment of Accident Sequences, Fission Product In-Plant Release and Interaction, Containment Challenge Analysis, and Analysis of Fission Product Releases from the Plant.

1.1 Critical Review of Accident Sequence Definition

1.1.1 Selection of Accident Initiators

- o justification for those omitted
- o criteria for inclusion
- o effect of these judgments on the final results

1.1.2 Event-Tree Definition

- o system description
- o justification for level of detail
- o treatment of common-cause issues
- o human factors issues in event-tree logic

1.1.3 Fault-Tree Development

- o selection of which faults requires fault-tree analysis
- o level of detail
- o human factors issues, including operating procedures
- o special attention to control systems issues and support systems issues

1.2 Quantification of Accident Sequences

1.2.1 System Failure Probabilities

- o component failure analysis and data
- o human factors issues
- o dependent failure analysis

1.2.2 Event-Tree Quantification

1.2.3 Fault-Tree Quantification

1.2.4 Uncertainty Analysis

- o systems interactions
- o dependencies across systems

1.2.5 Critical Review of Conclusions

- o special treatment of possible major issues
- o uncertainty of emergency planning development to these results

1.3 Within-Containment Phenomena

For each "important" accident sequence,

1.3.1 Description of Phenomena Leading to Core Degradation

- quantitative description, including time-sequence issues
- quantitative analysis

1.3.2 Meltdown and Release of Fission Products

1.3.3 In-Vessel and In-Containment Fission Product Behavior

- effect of engineered safety features
- time progression
- uncertainty analysis

1.3.4 Containment Challenge Issues

1.4 Containment Challenge Analysis

1.4.1 Time-Sequence Analysis of Containment Challenge

- overpressure
- bottom containment penetration
- steam explosions
- etc.

1.4.2 Containment Failure Modeling

- categorization of accident sequences into groups
- failure analysis
- uncertainty analysis

1.5 Quantification of Releases

1.5.1 Quantitative Description of Releases

- time sequences
- duration of release
- thermal character of release
- elevation or site of release
- fission product characterization

1.5.2 Quantification

- Emphasis on fission product release fractions
  - time sequences
  - physical properties of the release
- Uncertainty analysis

Task 2. Review and Critique of Previous Radioactive Plume Transport Modeling and Radiological Consequence Assessments

2.1 Plume Transport Modeling and Radioactivity Concentration

- o plume modeling parameters
- o dose-radioisotope concentration relationships

2.2 Health Effects Analysis

- o dose conversion relationships
  - o early deaths and injuries
  - o latent effects
  - o long-term exposure mechanisms (out of scope??) (ingestion pathway EPZ to be defined??)

2.3 Population Distribution

- o current census (1980)
- o projected population distribution (2000(?))

2.4 Meteorological Models

- o long-term historical data from site
- o single year hourly data source model from site
- o regional meteorological models (CRACIT satellite stations)
- o data preprocessing routine impacts (CRAC-2 & CRACIT)
- o sampling frequency requirements and probability/confidence results

2.5 Protective Action Procedures

- o nominal exposure relationships without protective actions
- o evacuation models and site characteristics
- o sheltering models for local residential/commercial structures

2.6 Modeling of Island-Specific Terrain Influences on Consequences

Task 3. Performance of Site Specific Consequence Analyses and Assessment of Protective Action Effectiveness

3.1 Assessment of Utility-Supported Analyses

- o CRAC-2 simulation of PL&G input parameters
- o comparision of CRAC-2 and CRACIT results

3.2 Site Specific Analyses

- o assessment of nominal results without protective actions
- o specialized protective action effectiveness assessments

3.3 Uncertainty Assessment for Critical Variables

- assessment of variations in results associated with uncertainties in critical variables identified in Tasks 1 & 2

Task 4. Project Integration, Documentation, and Technology Transfer

4.1 PRC Voorhees/Consequence Analysis Interface

4.2 Suffolk County and Radiological Emergency Executive Steering Committee Support

- Presentations to Steering Committee/County of Methodology and Results of Consequence Assessments
  - review of technological features of consequence analysis and implications to planners
  - reviews of final results of consequence assessment study
- Additional Support as Required

4.3 Internal Integration of Parts 1, 2, & 3

4.4 Preparation of Reports

- data presentation, review & analysis
- issue identification and assessment
  - presentation of critical results of consequence analysis affecting emergency planning potential for public risk reduction

APPENDIX B

APPENDIX B

REVIEW OF WITHIN-CONTAINMENT PHENOMENA
AND CONTAINMENT CHALLENGES

Reported in
Probabilistic Risk Assessment,
Shoreham Nuclear Power Plant Station

This Review Was Performed by
Dynatrek, Inc.
Under Subcontract from Future Resources Associates

July 1982

CONTENTS

1.0 INTRODUCTION

Future Resources Associates, Inc., of Berkeley, California, was contracted by Suffolk County, New York, to carry out Task 1 of the "Consequence Assessment for Suffolk County Radiological Emergency Response Plan" related to the Shoreham Nuclear Power Station.

The Dynatrek, Inc., was, in turn, subcontracted by Future Resources Associates, Inc., to review and study Sections 1.3 and 1.4 of Task 1, titled "Within-Containment Phenomena" and "Containment Challenge Analysis," respectively. Ten man-days (80 man-hours) were allocated for this study, plus the additional 5 man-days for preparation of the report.

This report represents the Dynatrek, Inc., contribution to the above-mentioned Consequence Assessment.

1.1 Listing of Documents Reviewed for This Study

Probabilistic Risk Assessment, Shoreham Nuclear Power Station, Long Island Lighting Company, by Science Applications, Inc., San Jose, California.

1. Volume I: pp. 1-1 through 3-12 and 3-138 through 4-6.
2. Appendix C: Containment Response Analysis for Degraded Core Accidents, pp. C-1 through C-90.
3. Appendix H: Containment Event Trees, pp. H-1 through H-112.
4. "MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual," NUREG/CR-1711, October 1980.
5. "Interim Technical Assessment of the MARCH Code," J. B. Rivard, Project Leader, Sandia Report, NUREG/CR-2285, November 1981.
6. "Core-Meltdown Experimental Review," prepared for USNRC by Sandia National Laboratories, Nuclear Fuel Cycle Programs, NUREG-0205, SAND 74-0382, March 1977.
7. Reactor Safety Study: Section III, Appendix A, NUREG-75/014 (WASH-1400).
8. "Phenomenological Assessment of Hypothetical Severe Accident Conditions of the Limerick Generating Station," by R. E. Henry of Fauske Associates, Inc., in Volume II of Probabilistic Risk Assessment, Limerick Generating Station, Philadelphia Electric Company, Docket Nos. 50-352, 353, March 1981.

9. Memorandum: L. S. Tong, Chief Scientist, USNRC/RES, to R. B. Minogue, Director, USNRC/RES, "Technical Review Meeting on Steam Explosions," with enclosures, May 24, 1982, USNRC Public Document Room.

1.2 Brief Summary of SAI (Science Applications, Inc.) Computed Accident Scenarios Considered in This Review

For the Shoreham PRA (Probabilistic Risk Assessment), five representative accident sequences have been developed for detailed deterministic containment response analysis. SAI has determined that these sequences represent the range of core melt events described in the probabilistic event trees. It was outside the Dynatrek, Inc., scope of work to either examine or challenge the choice of these limiting scenarios. Instead, our main emphasis is on the review of the methodology and of the results of SAI studies pertaining to (a) in-vessel (core/damage and meltdown progression), (b) transport of core debris from the reactor pressure vessel (RPV) to the control rod drive (CRD) room, (c) transport of core debris to containment drywell and wetwell, and (d) the resulting challenge to the containment integrity for the five scenarios selected by SAI.

It should be noted that the Fission Products transport within the RPV, within containment, and to the outside environment was also outside the scope of Dynatrek's work assignment.

In this Section, the salient results of the SAI deterministic analyses, performed with the MARCH computer code (Reference 4) are summarized to allow for more orderly, subsequent description of the impact of Dynatrek's assessment.

1.2.1 Class I Scenario: Loss-of-Coolant Makeup to Reactor

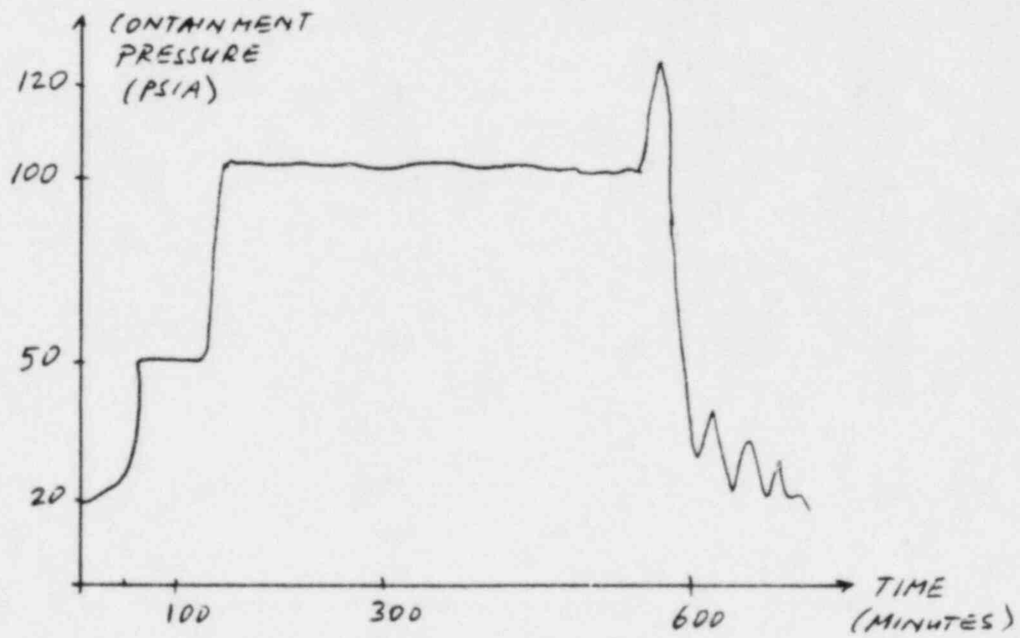
a) Main Assumptions:

- Some transient causes a reactor scram from 100% power
- Feedwater to RPV is unavailable
- Emergency Core Coolant is unavailable
- Main Steam Isolation Valves (MSIV) are closed
- The specified operator action to depressurize the RPV is not taken. Hence, the reactor pressure reaches the pressure relief valves set point and the steam passing through these valves (SRVs) is dumped to the containment drywell.

b) SAI Computed Events:

Time	Events
0-0.68 hrs.	Core heatup at constant pressure, core gradually uncovered, with attendant clad oxidation.
0.68-1.0 hrs.	Partial core meltdown. Core slumping into the RPV bottom head, at t=56 min. when more than 70% of core is melted and 22% of cladding has undergone chemical reaction (oxidized).
	Increased steam generation due to boiloff of lower plenum (LP) water; core debris quenched (may have become partially or totally solid).
115 min.	Reheat of core debris and some heatup of RPV bottom head. The latter fails due to high fluid pressure, combined with the elevated metal temperature, rather than due to melt-through.

Time	Events
	Temperature of the molten debris exiting the RPV is low. May be quenched in the drywell to below the 2500 F criterion for initiation of concrete interaction.
	"Dead water" associated with recirculation piping and other RPV fluid (steam and H_2) are discharged into containment.
	During boiloff of drywell water, the core debris is quenched from 2600°F to 732°F.
	Subsequently, core debris interacts with the concrete, causing generation of large amounts of noncondensible gas (mainly CO_2) and further oxidation of the debris metal (H_2 evolution).
595 min.	Containment pressure failure (120 psia) reached.
	Concrete penetration continues until metal solidifies, at which time gas generation is greatly reduced.



1.2.2 Class II Scenario: Loss-of-Containment Heat Removal

a) Assumptions:

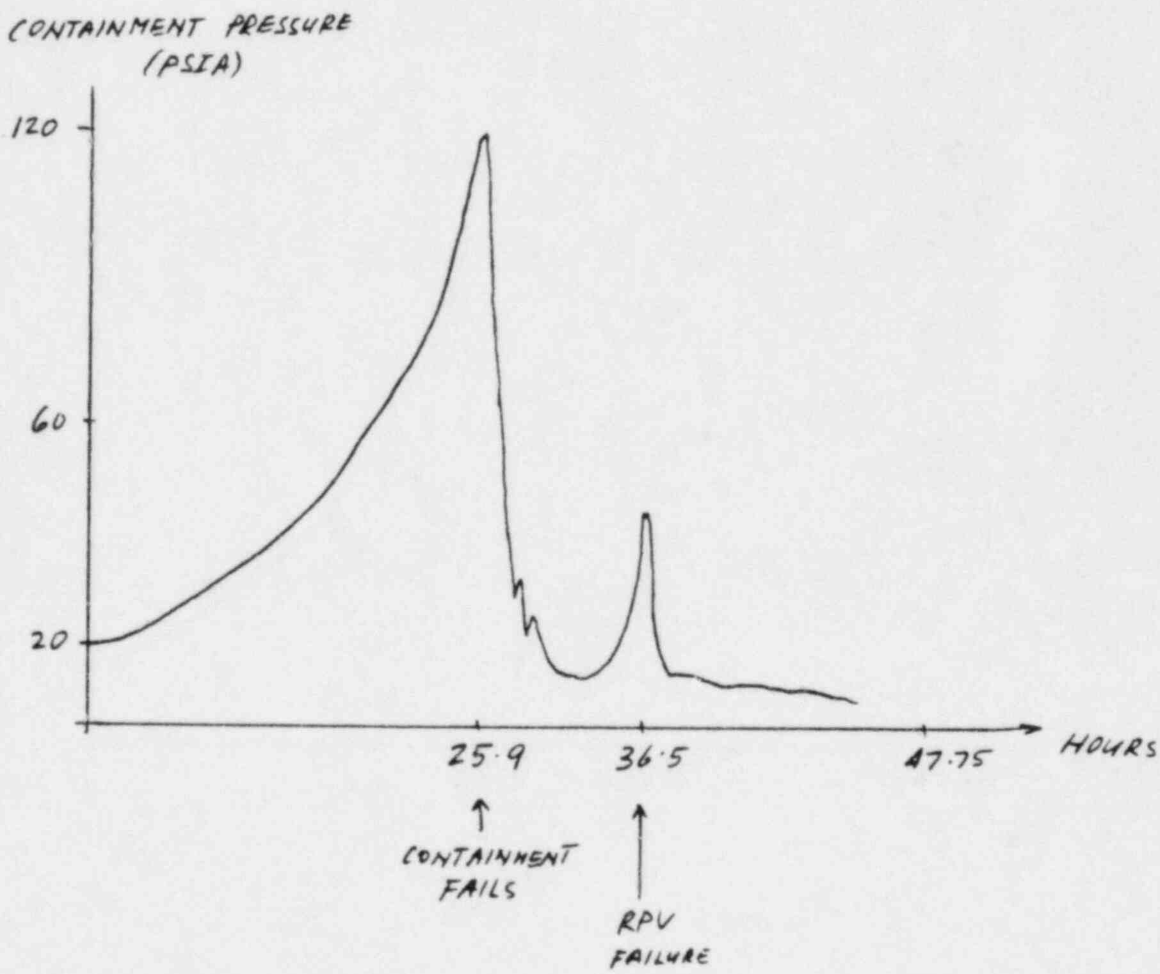
- Some transient causes a reactor scram from 100% power
- Feedwater to RPV is unavailable
- MSIVs are closed
- Loss-of-containment heat removal due to failure of the power conversion system and of the residual heat removal system
- RPV is at the SRV set point pressure with steam being discharged into containment wetwell.

b) SAI Computed Consequences:

Time	Events
	RPV water level decreasing. The reactor core isolation cooling system (RCIC) (using water from the condensate storage tank at 100°F) is initiated when low level mark is reached in RPV.
88 min.	Containment suppression pool temperature rises. When T_{pool}=120°F, the Automatic Depressurization System (ADS) is activated and RCIC injection is terminated. It should be noted that the reactor manufacturer specifications call for gradual rather than sudden depressurization as pool water heats up, thus prolonging RCIC injection.
	RPV depressurizes from 1146 psia to 50 psia in 3 minutes.

Time	Events
	Water in the RPV flashes and the core is totally uncovered for less than 10 minutes because the Low Pressure Core Spray System (LPCS) is activated upon the drop of the RPV pressure.
	Adequate core cooling by LPCS. Water level increases and covers the core. The RPV is maintained at low pressure (~150 psia) by relieving steam through SRVs to wetwell pool. Containment pressure increases.
25.9 hrs.	Containment failure pressure (120 psia) reached.
	ECCS assumed lost due to loss suction head caused by containment pressure drop to 1 atm.
	Water level in RPV begins to decrease due to loss of ECCS, and the core uncovers. Low level of decay heat.
27.8 hrs.	Core meltdown initiated.
28.5 hrs.	75% of core molten when core slumps into RPV bottom head. 21% of Z_r oxidized.
30.0 hrs.	Dryout of water in RPV bottom head.
	RPV head attack initiated when debris reaches 1840°F.
36.5 hrs.	The RPV bottom head fails when 27-53% of the bottom head is molten. Core debris reaches 3600°F.
	Debris-coolant interaction on drywell floor and in wetwell produces a 30 psi pressure spike, despite failed containment.
36.7 hrs.	Core debris totally quenched to saturation temperature.

Time	Events
43.75 hrs.	Concrete-debris interaction starts when debris reaches 2500°F.
46.17 hrs.	Vertical penetration rate into concrete increases significantly when oxide and metal layers exchange position.
	Decomposition gases blow through melt and react with metal. Heat of reaction increases metal layer heat source by a factor of 60 over the decay heat.
47.75 hrs.	Metal layer frozen, vertical penetration of concrete terminated.



1.2.3 Class III Scenario: Large Break LOCA

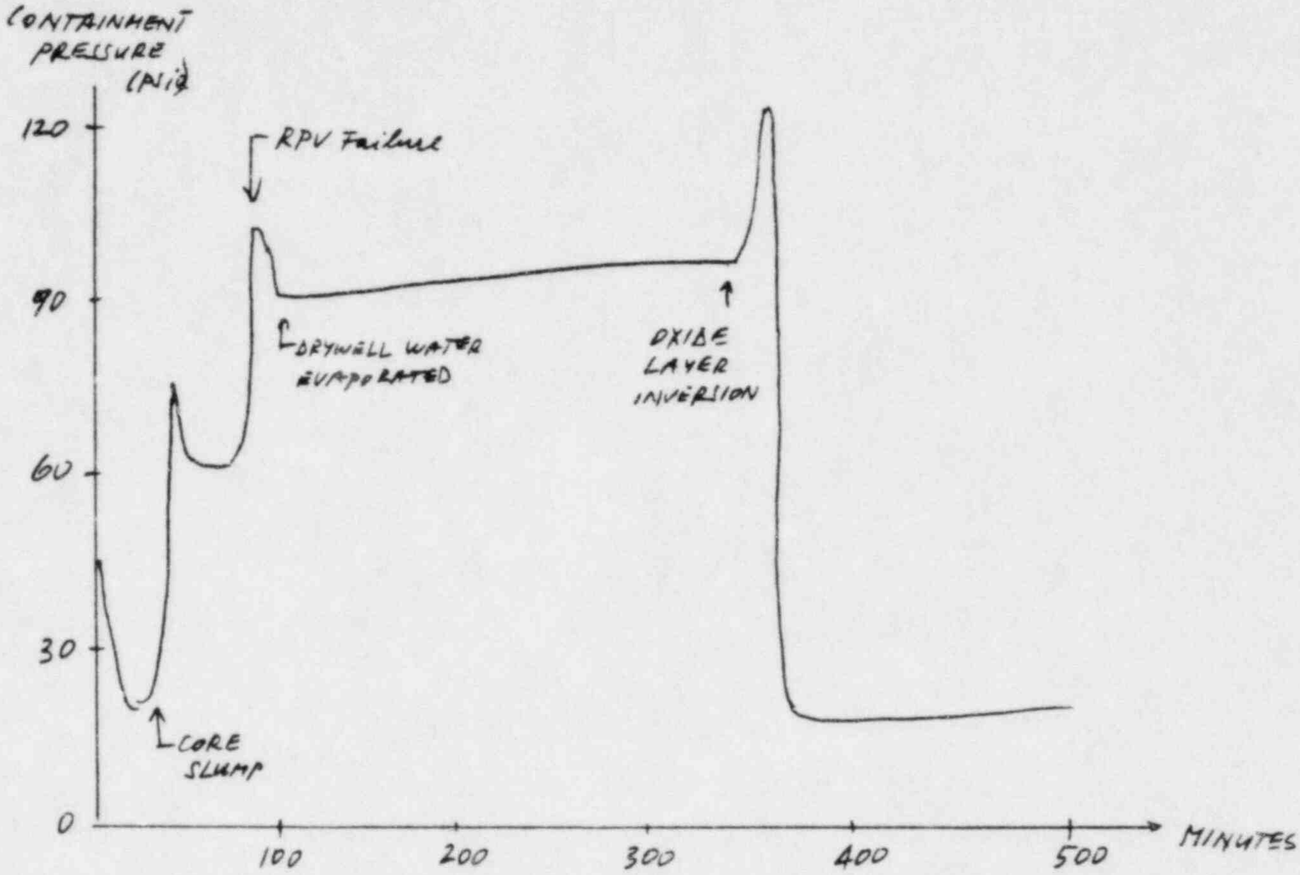
a) Assumptions:

- Double-ended (guillotine) break of the main coolant recirculation pipe, upstream of pump
- All ECCS lost
- Pressure suppression pool cooling is unavailable
- Containment sprays not operational

b) SAI Computed Consequences:

Time	Events
1 minute	RPV pressure decreased to the containment pressure. Core is uncovered before the fuel rods can heat up significantly.
18 minutes	The lowest core layer does not remain solid due to lack of cooling. Core starts to slump into RPV bottom head when 51% molten.
0.35 hrs.	Bottom head water rapidly evaporates (containment pressure reaches 80 psia) and RPV bottom head attack is initiated.
	High energy gases produced during core slump cause very high drywell temperature (3000°F).
	Very high metal-water reaction rates computed for the intact core region after slumping due to assumption of no flow blockage.
	Steam starvation could increase the meltdown time period of the (initially) intact core region by about a factor of 10.
	Rapid heatup rate of RPV bottom head. Debris/wall interface climbs from 370°F to 2600°F in 2 minutes.

Time	Events
48 minutes	RPV bottom head fails when 52.5% of wall thickness penetrated by the melt.
	Vessel inventory plus 32,800 lbs of water (from recore piping) discharged into drywell.
	Hot core materials interact with 126,100 lbs of water on drywell floor and containment pressure increases to 95 psia.
	Fragmented core debris quenches to 1175°F on drywell floor. Water limited interaction.
53 minutes	Drywell water evaporated. Total metal-water reaction of 37% produces 2050 lbs of H_2. However, less than 2% produced during core-water interaction outside RPV.
	Subsequent steam generation rate very low (10-150 lb/min), causing containment pressure to drop.
274 minutes	Core debris on drywell floor heats up to 25°F, starting interaction with concrete.
324 minutes	Strong interaction with concrete, due to metal vs. oxide layer reversal, causing large evolution of noncondensible gas.
354 minutes	Containment pressure fails (120 psia).
474 minutes	Melt propagated 11 inches into concrete then stopped. Event terminated.



1.2.4 Class IV Scenario: ATWS

a) Assumptions:

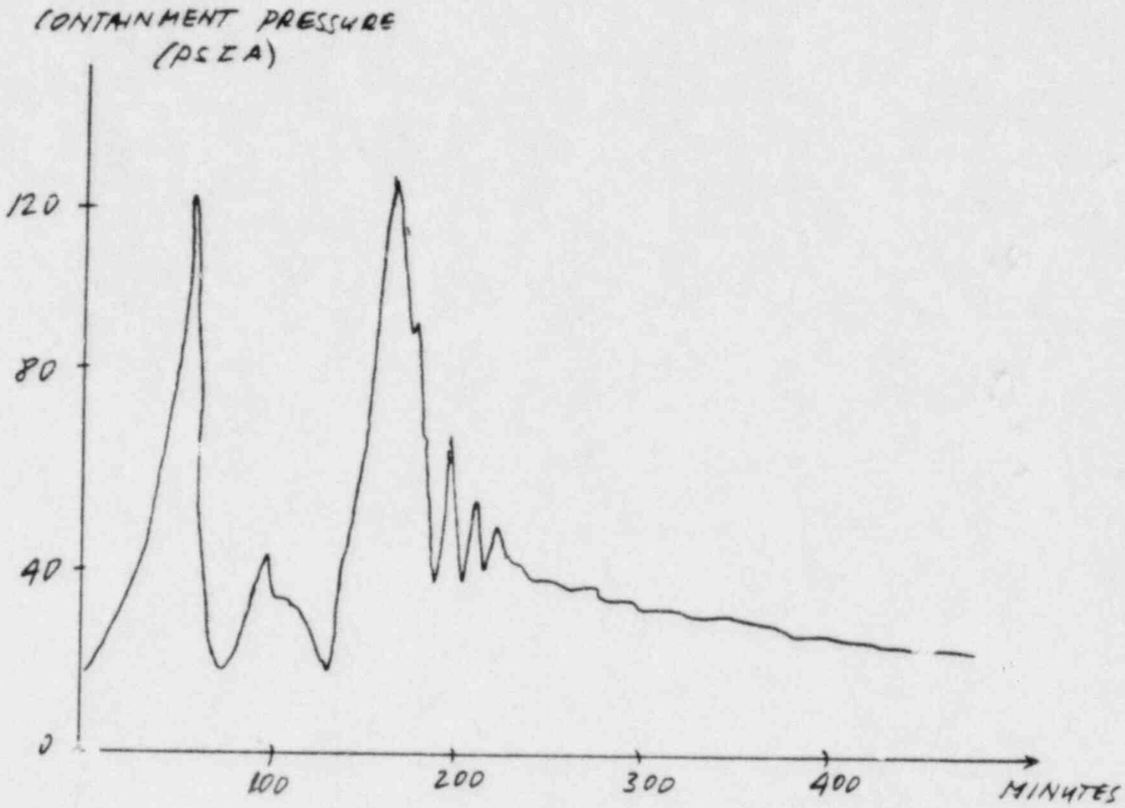
Some transient, at 100% reactor power

- MSIVs assumed to close
- Feedwater to RPV is not available
- One recirculation pump trips. Due to 50% reduction in core flow, neutronics feedback reduces power to 30%, as computed by the manufacturer.
- Failure of control rods to insert
- Failure of the liquid poison injection system

b) SAI Computed Consequences:

Time	Events
	HPCI turns on at low RPV water level. Throttles back when high level regained. RPV at high pressure relieving steam to wetwell pool via SRVs.
	Capacity of the Residual Heat Removal (RHR) system is insufficient to remove the additional heat load.
32.4 minutes	Containment fails upon reaching 120 psia.
	HPCI pumps cavitate due to low P_{cont}.
	Core starts to uncover; power starts to decrease due to void feedback.
	Core uncovers fast, hence, rods do not heat up significantly, causing low amount of clad oxidation.
	Water level drops by 14 ft/min. after ECCS terminated.
	Water level is at 2 ft above bottom support plate when core heatup occurs.
46 minutes	Core meltdown starts. Steam generation low since core nearly totally uncovered (water level less than 0.5 ft).
	7% of clad oxidized and 55% of core melted when core starts to slump into bottom head. Noncoherent slumping since no water present.
68 minutes	Total core collapse. Prior to that, rapid evergy transfer due to partial slumping, vaporizes all water in the RPV lower plenum. 47% of Z_r oxidized since no flow blockage assumed.

Time	Events
	Quenched core debris and support structure in lower plenum come to thermal equilibrium at about 3100°F.
73 minutes	Core debris starts to react with RPV bottom head.
105 minutes	Bottom head failure without melt-through caused by 1075 psi pressure differential and elevated wall temperature.
	Following RPV failure, containment pressure increases in spite of ruptured containment.
	Energy sources: 51,300 lbs of saturated water in recirculation piping plus 34,000 lbs of steam at elevated pressure in RPV vapor space.
110 minutes	Further debris-coolant interaction outside vessel creates additional energy source causing second pressure spike of about 120 psia.
118 minutes	Drywell floor dryout. Core debris quenched during dryout from 2600°F to 434°F.
534 minutes	Total period of about 400 minutes during which no significant mass and energy generation from debris-coolant or debris-concrete interaction, except for vaporization of wetwell water due to fallen debris.
	Concrete penetration starts when oxide layer becomes molten, followed by rapid degassing. The latter becomes the main driving force for fission product leakage to atmosphere.



1.2.5 <u>Class V: Main Steam Line Break Outside Primary Containment: Containment Bypass</u>

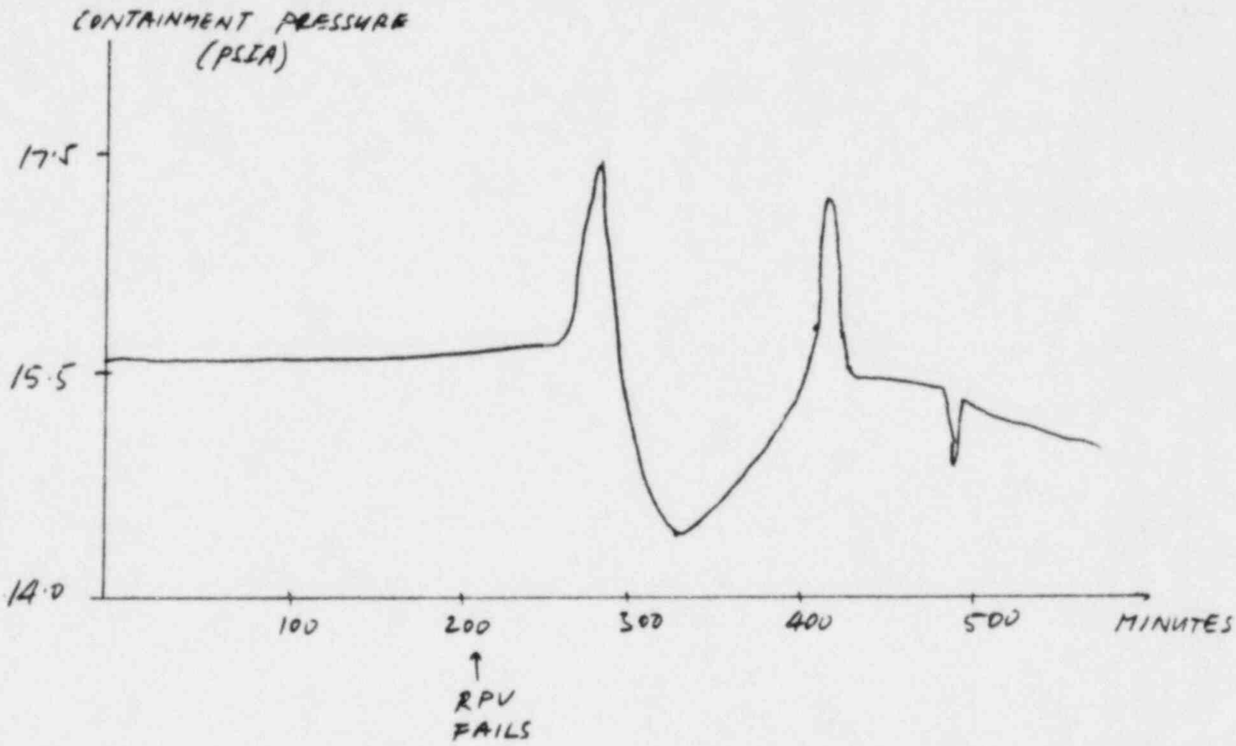
a) Assumptions:

- Main steam line pipe break (0.678 ft^2 break flow area due to flow restrictor) outside primary containment when reactor at full power.
- Isolation valves (MSIVs) fail to close
- Coolant makeup system failure
- Extended exposure to high temperature steam within secondary containment causes ECCS equipment failure
- Secondary containment blowout panels open when 3 psia reached.

b) SAI Computed Consequences:

Time	Events
	Amount of clad oxidation as per Class I.
	Primary system boiloff bypasses containment from beginning of transient.
5 minutes	RPV pressure dropped from 1026 psig to 273 psig.
12 minutes	Core starts to melt 10 minutes after uncovery.
16 minutes	Core slump initiated when 60% of core is molten.
	Core debris is held up in the core support structures and no vaporization predicted after initial energy exchange until after the second support grid reaches its failure temperature. Hence, metal/water reaction in the core region is limited.
50 minutes	Bottom head attack begins.
209 minutes	Bottom head fails. Core debris is at about 3780°F (2082°C). Debris drops onto dry drywell floor. Hence, no debris-coolant interaction in drywell.
	Core debris interacts with structure in the reactor pedestal area before interaction with concrete.
300 minutes	Debris-concrete interaction. Degassing releases energy into containment which is initially absorbed by wall and other internal structures since these were cold.

Time	Events
400 minutes	Rapid penetration of concrete starts. Energy released into drywell is about 2.8×10^5 Btu/min of which 80% is absorbed by containment walls.
	Containment drywell fluid (gas) temperature reaches a peak of 370°F.



2.0 ISSUES CONSIDERED BY DYNATREK, INC.

2.1 Meltdown Process Within RPV

It is our impression that SAI engineers have used competently and conscientiously, the only tool available at the time; i.e., the MARCH code, designed to address, in a highly simplified manner, the extremely complex physical processes involved in the meltdown scenario.

The extensive simplifications in the code were introduced for two reasons: (1) to produce a tool useful for probabilistic risk assessment which requires many computer runs for exploration consequences of various bounding assumptions. Hence, computational economy must have been one of the principal goals; (2) to produce a tool amenable to accepting some of the major uncertainties as input assumptions that could be changed from run to run. These stem from the relatively poor knowledge of various thermohydraulic processes that involve melt propagation and the attendant heat and mass transfer in complex geometries. The pertinent empirical base lags far behind the existing empirical base collected in the course of reactor safety research for situations that do not involve a degraded core. However, the word "uncertainty" is also used here to imply simplifications one needs to make to intentionally bypass the detailed calculations that would cause the code to become long running.

Sandia engineers have performed a thorough review of the MARCH code (see item 5 in Section 1.1) and found many items that could be improved upon, most of which were known to the Battelle Columbus Laboratories staff that developed MARCH for USNRC. As a result of these and other reviews, the USNRC has sponsored the development of a significantly improved and extended code, now named MELCOR, the development of yet another code at the Idaho National Engineering Laboratory (INEL) named SCDAP, and procurement of more pertinent empirical base at INEL (in the PBF test facility) and elsewhere.

When reviewing the major assumptions in the MARCH code, one gets the impression that the code was primarily oriented towards pressurized water reactors (PWRs) and their containments that do not employ pressure suppression systems, such as in Shoreham.

The RPV structure, its flow patterns and cooling modes, and, especially, the structural components within the RPV bottom head, are quite dissimilar from PWRs. None of these have entered in any significant way to the MARCH code application used for the Shoreham PRA.

In the course of the SASA (Severe Accident Sequence Analysis) calculations being performed by the Oak Ridge National Laboratory (ORNL) for the USNRC as applied to BWRs, the ORNL engineers have also come to the conclusion that MARCH has many <u>deficiencies</u>. Out of 21 listed inadequacies, we shall extract only a few:

* Modeling of core heat transfer to upper and lower BWR vessel internals
* Modeling of core collapse
* Failure of bottom head via CRD tube penetration not considered
* Suppression pool and wetwell/drywell interaction

* Rod-to-rod radiation heat transfer not included
* Vessel water level calculation does not include variable flow areas
* Fuel pin melt/slump/freeze phenomena are not mechanistically modeled.

The work on removal of the last item mentioned above was subcontracted to the Renssalaer Polytechnic Institute (Professor R. T. Lahey, Jr., as Principal Investigator).

Some of the listed inadequacies have already been removed by ORNL and/or Battelle engineers. Others may take years to replace.

Dynatrek, Inc., staff could not undertake any serious effort in the 10 day-assignment to address consequences of these code weaknesses and have, therefore, decided to accept the SAI computed in-vessel events at their face values. We would have felt more at ease, however, if SAI had also considered the inflow of the CRD (control rod drive) cooling water at 185 gpm, since it could significantly delay the in-vessel meltdown and debris collapse onto the RPV bottom head.

2.2 Vessel Melt-Through and Discharge of Molten Debris into CRD Room

2.2.1 Properties of the Molten Debris, Pertinent to Shoreham

UO_2: 456.7 lbs/assembly x 560 assemblies = 255,752 lbs

Z_r: 224.3 lbs/assembly x 560 assemblies = 125,608 lbs

Steel: If all internals were to melt: ~500,000 lbs

Total weight of "corium" ≈400,000 kg

Weight % $UO_2 = 29$

$Z_r = 14$

$SS = 57$

Corium properties:

Density

$$\rho_m = 10^3 (0.29 \times 10.96 + 0.14 \times 6.5 + 0.57 \times 7.95) \approx 8.6 \times 10^3 \ kg/m^3$$

Spec. heat

$$c_{p_m} = (0.29 \times 628 + 0.14 \times 356 + 0.57 \times 670) \approx 613.9 \ J/kg\,^\circ C$$

Conductivity (thermal)

$$k_m = (0.29 \times 0.035 + 0.71 \times 0.3) \simeq 0.22 \ W/cm\,^\circ K = 22 \ W/m\,^\circ K$$

Heat of fusion, $\lambda_m \approx 2.713 \times 10^5 \ J/kg$

2.2.2 Melt-Through Mechanism

We do not believe that the RPV bottom head would fail in either of the two ways described in the Shoreham PRA: structural failure of the vessel wall due to elevated pressure and temperature or the wall melt-through when the RPV pressure is low.

Instead, we agree with the scenario described by R. Henry of Fauske Associates, Inc. (see item 8 in Section 1.1) wherein the relatively thin metal that seals the CRD tube, or the CRD tube itself, is much more likely to fail first upon contact with the melt.

The melt is very unlikely to reach the bottom head in a coherent fashion (gross slumping). It is more plausible to consider downward streaming of melt around the vessel axis. That melt will not attack the CRD tube as long as there is some water left in the lower plenum. If the amount of steam generation caused by quenching of melt is insufficient to stop further melting (a likely case), increased amount of debris will accumulate on the bottom portion of the RPV and it would eventually remelt the fraction of debris that was frozen by the liquid--which by now has evaporated. This whole process could be delayed significantly if the CRD cooling water were continuously supplied.

Eventually, one or more CRD tubes would fail and the debris discharge into the CRD room would commence.

2.2.3 Melt Discharge Rate at High RPV Pressure

Since it is anticipated that the total discharge of the melt would last a short time and the melt volume is small compared to the total system volume under pressure, it is reasonable to assume that the fluid pressure inside the vessel will not change until all melt has discharged.

As the hot debris flows through the initial opening (of assumed equivalent radius r_o), the RPV wall would ablate, thus, enlarging the break size.

Let: h = heat transfer coefficient between the melt stream and the vessel wall

T_m = debris temperature (~2000°C)

$T_{s,m}$ = molten steel temperature (~1500°C)

ρ_s = vessel wall density

T_s = vessel steel (initial) temperature (~300°C)

$p_{RPV} \cong 7.0$ MPa

p_{CON} = 0.2 MPa

The rate of ablation/melting of the vessel wall could be found, approximately, from

$$\rho_s \cdot [c_{p_s} (T_{sm} - T_s) + \lambda_s] 2\pi r \frac{dr}{dt} = 2\pi r h (T_m - T_{ms})$$

$$\frac{dr}{dt} \approx \frac{h (T_m - T_{ms})}{\rho_s [c_{p_s} (T_{sm} - T_s) + \lambda_s]} \qquad \ldots (1)$$

R. Henry utilized the Reynolds analogy to determine the heat transfer coefficient

$$h = f \cdot \rho_m \cdot c_{pm} \cdot \frac{u_m}{2}$$

where f = friction factor ~0.005

$$u_m = \sqrt{2(p_{RPV} - p_{CONT})/\rho_m} \simeq 40 \text{ m/sec} \quad \cdots (2)$$

$$h = 528 \times 10^3 \ \frac{W}{m^2 \, ^\circ C}$$

$$\lambda_s \simeq 2.76 \times 10^5 \text{ J/kg}$$

and

$$\frac{dr}{dt} = \frac{528 \times 10^3 (2000 - 1500)}{7.45 \times 10^3 [670 (1500 - 300) + 2.76 \times 10^5]}$$

$$\simeq 0.0307 \quad \cdots (3)$$

$$r(t) = r_0 + 0.0307 \cdot t \quad \cdots (4)$$

Let n_{CRD} define the number of CRD tubes that have failed, more or less simultaneously.

Then, the total break area

$$A_{BRK}(t) = \pi \cdot n_{CRD} \, (r_0 + 0.0307 \cdot t)^2 \quad \cdots (5)$$

Let $B \equiv 0.0307$ (using Henry's nomenclature), then, since $u_m = 40$ m/sec (from equation 2), volumetric flow of melt through the total break becomes

$$Q_{RPV}(t) = 40\, n_{CRD}\, \pi\, (r_o + Bt)^2 \cdot C_D \quad \ldots (6)$$

where C_D = discharge coefficient ~0.6.

Mass of metal/debris discharged

$$M_m(t) = \rho_m \int_0^t Q_{RPV}(t)\, dt$$

$$= 1.08 \times 10^6\, C_D\, n_{CRD} \left(r_o^2 t + r_o B t^2 + \frac{B^2}{3} t^3\right) \quad \ldots (7)$$

Let $n_{CRD} = 5$, $C_D = 0.6$

$B = 0.0307$

$r_o = \frac{1}{2} \times 0.1 = 0.05$ m

$$M_m(t) = 3.24 \times 10^6 \cdot t\,(2.5 \times 10^{-3} + 1.535 \times 10^{-3} \cdot t + 3.1416 \times 10^{-4} t^2)$$

t(secs)	M_m(kg)	A_{BRK}(m^2)
1	1.41×10^4	0.1023
2	4.4×10^4	0.1949
3	9.65×10^4	0.317
4	1.77×10^5	0.469
5	2.92×10^5	0.65

Not all of the available metal within the RPV will melt and flow out. Assuming that about 50% will flow out (i.e., ~200,000 kg), it could discharge in less than 5 seconds.

2.2.4 Gravity Driven Discharge of Melt

To simplify the matter, we will assume that all of the debris that could flow out has been accumulated in the RPV bottom head just prior to failure of n_{CRD} tubes.

The height of the molten debris within the hemispherical bottom head, as shown by Henry, can be found from *debris volume*

$$V_m = \frac{M_m}{\rho_m} = \int_0^H \pi r^2 dh \qquad \ldots \quad (8)$$

where $R \sin \theta = 4$

$R \cos \theta = R - h$

$r^2 \quad = 2Rh - h^2$

where R = radius of the RPV

$$\therefore \quad V_m = \pi \left(RH^2 - H^3/3\right) \qquad \ldots \quad (9)$$

The rate of change of that volume will equal the volumetric outflow

$$Q_{RPV}(t) = \frac{dV_m}{dt} = -\pi H(2R - H)\frac{dH}{dt} \qquad \ldots \quad (10)$$

Velocity of discharge through the break

$$u_m = \sqrt{2gH} \qquad \ldots (11)$$

Therefore,

$$Q_{RPV}(t) = n_{CRD}\, \pi\, r^2(t)\, C_D \sqrt{2g\,H(t)} \qquad \ldots (12)$$

As before,

$$\frac{dr}{dt} = \frac{h\,(T_m - T_{s,m})}{\rho_s\,[c_{p_s}\,(T_{sm} - T_s) + \lambda_s]}$$

where now

$$h = \frac{f\,\rho_m\, c_{p_m} \sqrt{2gH}}{2} \qquad \ldots (13)$$

and the Reynolds analogy was again utilized, together with equation (11), for the jet velocity.

Let $$K \equiv \frac{f \cdot \rho_m \cdot c_{p_m}\,(T_m - T_{sm})\sqrt{2g}}{2\rho_s\,[c_{p_s}\,(T_{sm} - T_s) + \lambda_s]} \qquad \ldots (14)$$

$$\frac{dr}{dt} = K\sqrt{H} \qquad \ldots (14a)$$

$$r(t) = r_o + K \int_0^t \sqrt{H(t)}\, dt \qquad \ldots (15)$$

Equating (10) and (11) and utilizing (15)

$$H(2R-H)\frac{dH}{dt} = -C_D n_{CRD}\sqrt{2gH}\left\{r_0 + K\int_0^t \sqrt{H}\,dt\right\}^2 \quad \ldots \quad (16)$$

$$\sqrt{H}(2R-H)\frac{dH}{dt} = -C_D n_{CRD}\sqrt{2g}\left\{r_0 + K\int_0^t \sqrt{H}\,dt\right\}^2$$

This equation can be solved numerically for H(t), utilizing forward finite differencing and small Δt

$$H_N = H_0 - \frac{(C_D n_{CRD}\sqrt{2g})\Delta t}{(2R-H_0)\sqrt{H_0}}\left\{r_0 + K\int_0^t \sqrt{H_0}\,dt\right\}^2 \quad \ldots \quad (17)$$

where, using trapezoidal (simplest) numerical integration, and

$t = m\Delta t$

$$\int_0^t \sqrt{H}\,dt = \frac{\Delta t}{2}\left\{(\sqrt{H_*}+\sqrt{H_1}) + (\sqrt{H_1}+\sqrt{H_2}) + \cdots (\sqrt{H_{m-1}}+\sqrt{H_m})\right\}$$

and $\sqrt{H_*}$ = initial height of melt in RPV.

Another way would be to solve for r using (14a) by forward finite differences

$$r_N = r_{N-1} + \Delta t \cdot k\sqrt{H_{N-1}} \qquad \ldots (18)$$

Substitute on the RHS of (16a)

$$\sqrt{H_{N-1}}\,(2R - H_{n-1})\,\frac{H_N - H_{n-1}}{\Delta t} = -\,(C_0\, n_{CRD}\sqrt{2g})(r_{n-1} + \Delta t\, k\sqrt{H_{n-1}})^2$$

$$H_n = H_{n-1} - \frac{\Delta t\,(C_0\, n_{CRD}\sqrt{2g})}{(2R - H_{n-1})\sqrt{H_{n-1}}}\,(r_{n-1} + \Delta t\, K\sqrt{H_{n-1}})^2 \qquad \ldots (19)$$

With r and H known from (18) and (19), respectively, Q_{RPV} is found from (12); providing fine enough time increments are used.

2.3 Debris Transport Within CRD Room

2.3.1 Debris Outflow from CRD Room

The floor of the CRD room (pedestal room?) is provided with four downcomers that connect it to the wetwell pool below. The remaining 84 downcomers (six of them capped) are located on the drywell floor area outside the pedestal.

Within the CRD room, the upper edge (entrance) of the downcomers is less than 3½ inches above the floor. In addition, the CRD room is provided with a 2-ft wide x 6-ft high manway that provides a passage to the drywell region outside the pedestal. That manway is blocked over its lower 2 ft by a 2½-inch-thick steel "sill" that, presumably, would hinder transport of any fluid across the manway, unless airborne.

The CRD room contains drive housings for 137 control rods, various pipes, and massive support beams, all fairly tightly clustered below the reactor bottom. Any jet (be it pressure or gravity fed) of molten metal would be first intercepted by this structure. It is unlikely that the structure would melt before all the melt within the RPV bottom head has been discharged. While some dispersal of the melt onto the drywell floor outside the pedestal area may take place (through upper support structure), it is judged that this will be a very small fraction of the amount discharged from the RPV, in the case gravity discharge, and a somewhat larger fraction (still less than 5-10%) in the case of pressurized discharge.

As the melt discharged from the RPV accumulates on the CRD room floor, once its level exceeds the height of the downcomer top flange (about 3½ inches = 0.0889 m), the molten debris will start pouring into the four downcomers.

A simple "weir" type relationship could be used to evaluate the volumetric flow rate per unit length of downcomer circumference

$$Q_d''(t) = \int_0^{\Delta H(t)} \sqrt{2gy}\, dy = \frac{2}{3}\sqrt{2g}\left[\Delta H(t)\right]^{3/2} \quad \cdots (20)$$

where $\Delta H(t) = H_{CRD}(t) - H_{TD}(t)$

$H_{CRD}(t)$ = Molten debris level height in the CRD room

$H_{TD}(t)$ = Height of top edge of downcomer above floor in CRD room. It is shown as a function of time because it could eventually be removed by melting.

That relationship is valid only if the flow of melt through the downcomer does not encounter unusual resistance as formed, for example, by the countercurrent flow limitation (CCFL) which will be examined in Section 2.3.3.

Consequently, barring any further obstacles to downflow of melt, the total volumetric flux of melt leaving the CRD room through four downcomers becomes

$$Q_d(t) = \frac{2}{3}\sqrt{2g} \cdot 4 \cdot 2\pi R_d \cdot \Delta H^{3/2}$$

and, with $R_d = 1\ ft = 0.3048\ m$

$$Q_d(t) = 22.62\left[H_{CRD}(t) - H_{TD}(t)\right]^{3/2} \quad \cdots (22)$$

In a similar fashion, outflow of melt from CRD room through the manway becomes

$$Q_{SILL}(t) = b\,\frac{2}{3}\sqrt{2g}\left[H_{CRD}(t) - H_{SILL}(t)\right]^{3/2} \qquad \cdots \quad (23)$$

where H_{SILL} is the manway sill height which can also disappear if the sill plate should melt when in contact with hot debris.

With b = 2 ft = 0.6096 m

$$Q_{SILL}(t) = 1.8\left[H_{CRD}(t) - H_{SILL}(t)\right]^{3/2} \qquad \cdots \quad (23a)$$

Finally, the height of molten debris in the CRD room is obtained from the following mass balance:

$$A_{CRD} \cdot \frac{dH_{CRD}}{dt} = Q_{RPV}(t) - Q_{\alpha}(t) - Q_{SILL}(t) \qquad \cdots \quad (24)$$

where A_{CRD} = net floor area in the downcomer room.

From Shoreham PRA specifications, the gross CRD room (cavity) floor area = 261.1 ft^2.

The area of four downcomers = $4 \times \frac{2^2\pi}{4} = 12.57\ ft^2$

$A_{CRD} = 249\ ft^2 = 23.13\ m^2$

Equation (24) will be solved numerically for $H_{CRD}(t)$ for the cases of pressurized discharge and the gravity discharge.

To summarize, the source and sink terms:

Pressurized Discharge:

$$Q_{RPV} = 75.4\ n_{CRD}(0.05 + 0.0307 \cdot t)^2$$

as long as $M_m(t) < 200{,}000$ kg

where, by equation (7)

$$M_m(t) = 6.484\text{x}\ 10^5 \cdot n_{CRD} t(2.5\text{x}10^{-3} + 1.535\text{x}10^{-3} t + 3.1416\text{x}10^{-4} t^2)$$

Thereafter, $Q_{RPV} = 0$.

Gravity Discharge:

$$Q_{RPV}(t) = 8.35\ n_{CRD}\ r^2 \sqrt{H}$$

$$r = 0.05 + K \int_0^t \sqrt{H}\, dt$$

$$K = 0.0307\sqrt{2g}/40 = 3.44\text{x}10^{-3}$$

and H is found from numerical integration of

$$\sqrt{H}(5.54 - H)\frac{dH}{dt} = -2.658\ n_{CRD}\ r^2$$

where 5.54 = 2R = 18'-2" RPV diameter and the initial melt height in RPV, H_o, is found by trial and error from equation (9):

$$2.27\ H_o^2 - \frac{H_o^3}{3} = \frac{M_{m,o}}{\pi \rho_m} = 7.4$$

$$\therefore H_o \approx 1.855\ \text{m}$$

$Q_{RPV} = 0$ when $H = 0$

$$Q_d = 22.62\left[H_{CRD}(t) - H_{TD}(t)\right]^{1.5}$$

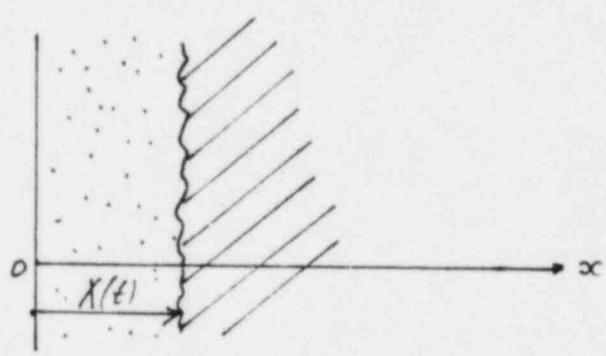
$$Q_{SILL} = 1.8\left[H_{CRD}(t) - H_{SILL}(t)\right]^{1.5}$$

Before proceeding with examination of downcomer countercurrent flow limitations, it is necessary to determine the time behavior of $H_{TD}(t)$ and $H_{SILL}(t)$.

2.3.2 Melting of Steel Barriers

Two cases will be addressed: The first involving an analytic solution for melt propagation into a semi-infinite region of steel; the second based on Hesson's model (see NUREG-0205, p. 5-30).

a) Analytic Solution for Semi-Infinite Steel Region



Consider a very thick steel slab with one face being kept in intimate contact with the liquid debris. Let

$T_S(x,t)$ denote the local temperature of steel

$T_m(x,t)$ = local temperature of molten debris

$T_{m,S}$ = melting point of steel

Governing equations:

$$\frac{\partial T_s}{\partial t} = \varkappa_s \frac{\partial^2 T_s}{\partial x^2} \qquad \ldots \quad (25)$$

$$\frac{\partial T_m}{\partial t} = \varkappa_m \frac{\partial^2 T_m}{\partial x^2} \qquad \ldots \quad (26)$$

$$T_s(x,t) \rightarrow T_{s0} \quad \text{as } x \rightarrow \infty \qquad \ldots \quad (27)$$

$$T_m(x,t) = T_{m0} \quad \text{at } x=0,\ t>0 \qquad \ldots \quad (28)$$

$$T_m = T_s = T_{m,s} \quad \text{at } x = X(t) \qquad \ldots \quad (29)$$

X(t) = position of the steel melt boundary

$$k_s \left.\frac{\partial T_s}{\partial x}\right|_{x=X} - k_m \left.\frac{\partial T_m}{\partial x}\right|_{x=X} = \rho_s \lambda_s \frac{dX}{dt} \qquad \ldots \quad (30)$$

where k = thermal conductivity

λ_s = heat of fusion

ρ_s = steel density

$\varkappa$ = thermal diffusivity

$$T_m(x,t) = T_{m0} + A \operatorname{erf} \frac{x}{2\sqrt{\varkappa_m t}} \qquad \ldots \quad (31)$$

satisfies (26) and (28)

$$T_s(x,t) = T_{s0} + B \operatorname{erfc} \frac{x}{2\sqrt{\varkappa_s t}} \qquad \ldots \quad (32)$$

satisfies (25) and (27)

Upon substitution into (30)

$$-\frac{k_s B}{\sqrt{\pi \varkappa_s t}} e^{-\frac{X^2}{4\varkappa_s t}} - \frac{k_m A}{\sqrt{\pi \varkappa_m t}} e^{-\frac{X^2}{4\varkappa_m t}} = \rho_s \lambda_s \frac{dX}{dt} \quad \ldots (33)$$

This will be satisfied, for all t, if

$$X(t) = 2\beta\sqrt{\varkappa_m t} \quad \ldots (34)$$

where β is a constant that needs to be determined. (34) into (33):

$$-\frac{k_s B\, e^{-\beta^2 \frac{\varkappa_m}{\varkappa_s}}}{\sqrt{\pi \varkappa_s t}} - \frac{k_m A\, e^{-\beta^2}}{\sqrt{\pi \varkappa_m t}} = 2\beta \rho_s \lambda_s \frac{\sqrt{\varkappa_m}}{2\sqrt{t}} \quad \ldots (35)$$

From (29) and (31)

$$A = -\frac{T_{m0} - T_{m,s}}{\operatorname{erf} \beta} \quad \ldots (36)$$

From (29) and (32)

$$B = \frac{T_{m,s} - T_{s0}}{\operatorname{erfc}\left(\beta\sqrt{\frac{\varkappa_m}{\varkappa_s}}\right)} \quad \ldots (37)$$

Then, since $\varkappa_m = \frac{k_m}{\rho_m c_{p_m}}$, substituting (36) and (37) into (33)

$$\frac{e^{-\beta^2}}{\mathrm{erf}\,\beta} - \left(\frac{T_{m,s}-T_{so}}{T_{m_o}-T_{m,s}}\right)\frac{k_s}{k_m}\sqrt{\frac{\varkappa_m}{\varkappa_s}}\;\frac{e^{-\beta^2\frac{\varkappa_m}{\varkappa_s}}}{\mathrm{erfc}\left(\beta\sqrt{\frac{\varkappa_m}{\varkappa_s}}\right)} = \frac{\beta\,\lambda_s\sqrt{\pi}}{c_{p_m}\left(T_{m_o}-T_{m,s}\right)} \quad \cdots (38)$$

This equation is solved for β by trial and error. Then, equation (34) can be used to approximate a time when a thickness δ_s (=X) will be melted. Hence, time to melt

$$\Delta t_s \leq \frac{1}{\varkappa_m}\left(\frac{\delta_s}{2\beta}\right)^2 \quad \cdots\cdots (39)$$

Let

$\rho_s \approx 8000(\mathrm{kg/m^3})$

$c_{p_m} \approx 500(\mathrm{J/kg\ °C})$

$\lambda_s \approx 2.76 \times 10^5(\mathrm{J/kg})$

$k_m \approx k_s \approx 46(\mathrm{W/m\ °C})$

$\varkappa_m \approx \varkappa_s \approx 1.164 \times 10^{-5}(\mathrm{m^2/sec})$

$T_{mo} \approx 2200°C$

$T_{ms} \approx 1500°C$

$T_{so} \approx 100°C$

$$\frac{e^{-\beta^2}}{\mathrm{erf}\,\beta} - 2.0\,\frac{e^{-\beta^2}}{\mathrm{erfc}\,\beta} = 1.4\beta \quad \cdots (40)$$

By trial and error, $\beta \approx 0.275$

δ_s (inch)	δ_s (m)	Δt_s (sec)	Δt_s (min)
3/8	0.009525	25.8	0.43
1.0	0.0254	183.0	3.0
2.0	0.0508	733.0	12.2
3.0	0.0762	1649.0	27.5

b) The Second Model, Lumped Parameter Concept

According to Hesson's experiments (see NUREG-0205, p. 5-30), the horizontal heat flux from molten UO_2, to steel is about $30.0\ cal/(cm^2\text{-}sec) = 1.26 \times 10^6\ W/m^2$. Then the lumped parameter heat balance yields

$$\rho_s c_{p_s} \delta_w (T_{m,s} - T_{so}) + \rho_s \lambda_s \delta_w = 1.26 \times 10^6 \Delta t \qquad \ldots (4)$$

With
$c_{p_s} \simeq 500 (J/kg\ °C)$

$T_{ms} \simeq 1500°C$

$T_{so} \simeq 100°C$

$\lambda_s \simeq 2.76 \times 10^5\ (J/kg)$

$\rho_s \simeq 8000\ (kg/m^3)$

$\Delta t \simeq 6188 \cdot \delta_w$ (sec)

δ_w (inch)	δ_w (m)	t (sec)	t (min)
3/8		58.9	1
1		157.18	2.6
2		314.3	5.24
3		471.5	7.9

This model gives, as expected, significantly shorter melt times for steel thicknesses greater than 1 inch.

While the above melt times may be appropriate for addressing the manway sill melt, the downcomer top flanges would require consideration of the downward-directed heat flux which Hesson found to be smaller by about a factor of 2.5.

Consequently, a 1½-inch-thick flange, if covered by molten debris, would melt in approximately 4 x 2.5 = 10 minutes. The 3 inch stub of 3/8-inch-thick downcomer pipe protruding above the flange would be removed in less than 1 minute by ablation, as soon as the melt starts to pour into the downcomers.

During our visit to the Shoreham plant, we have noticed that each downcomer top carries a (missile shield?) steel cover plate about 1½ inch thick, supported by three or four columns (about 1 inch in diameter).

While these steel columns would melt quickly, the cover plates could either fall down and temporarily block the downcomer, or because the molten debris is heavier than steel, the covers could be pushed away by the jet or by debris before they are also melted.

2.4 Study of Countercurrent Flow Limitations Within Downcomers

2.4.1 Scenario

As the molten debris pours over the downcomer entrance, it is likely to break up into chunks or globules. Initially, these globules encounter only the negligible resistance offered by more/less stagnant steam. They would reach a velocity

$$u_1 = \sqrt{2 g \ell_1}$$

by the time they hit
water within the downcomer.

If ℓ_1 + h = downcomer length and h is the submerged depth of the downcomer (= 8 ft per Shoreham PRA), then, for the total downcomer length of about 56 ft,

$$\ell_1 \simeq 48 \text{ ft} = 14.6 \text{ m}$$

$$u_1 \approx 17 \text{ m/sec}$$

As soon as the melt hits the water, it will break up into much smaller particles.

The final size of these particles, if they had many feet (>>8) of water to fall through, could be obtained from the force balance on particles and the Weber criterion:

$$C_D \pi r_m^2 \rho_\ell \frac{V_r^2}{2} = (\rho_m - \rho_\ell) g \frac{4}{3} \pi r_m^3 \qquad \ldots (42)$$

$$\text{and} \qquad We = \rho_\ell V_r^2 \frac{2 r_m}{\sigma_m} \qquad \ldots (43)$$

where r_m = radius of the representative particle,

C_D = drag coefficient for spherical particles

ρ_L = density of water

V_r = relative velocity between particles and water

ρ_m = particle density

σ_m = particle surface tension

W_e = Weber number

Combining (42) and (43)

$$V_r = \left(\frac{8 W_e}{3 C_D}\right)^{\frac{1}{4}} \cdot \rho_L^{-\frac{1}{2}} \cdot \left[\sigma_m \, g \, (\rho_m - \rho_L)\right]^{\frac{1}{4}} \quad \ldots \quad (44)$$

back substitution into (43) gives

$$r_m = \sqrt{\frac{3 W_e C_D}{32}} \cdot \sqrt{\frac{\sigma_m}{g(\rho_m - \rho_L)}} \quad \ldots \quad (45)$$

With σ_m = 441 dynes/cm = 0.441 N/m

$\rho_m = 8.6 \times 10^3$

$\rho_L = 950$

$W_e = (W_e)_{CRIT} \simeq 12$

$C_D \simeq 0.5$

$$\left(\frac{8W_e}{3C_D}\right)^{\frac{1}{4}} \simeq 2.8 \quad , \quad \left(\frac{3\ C_D\ W_e}{32}\right)^{\frac{1}{2}} \simeq 1.5$$

V_r = 1.22 m/sec, $r_m = 3.714 \times 10^{-3}$m

Due to the fact that the length of travel through water is much shorter than needed to obtain the terminal size, we assume that the effective r_m is about 50% larger; i.e.,

$$r_m \approx 0.005571 \text{ m}$$

(particle diameter of 11.5 mm)

As these particles traverse the water space, the heat transfer between particles and water will produce steam that would flow upwards against the descenting particles in the steam space above the water surface. That steam flow, under certain circumstances, may be large enough to generate the drag on particles that equals or exceeds the particle weight, thus preventing the particles from proceeding downwards or expelling them out through the downcomer entrance. That situation is called "flooding."

Presence of metal droplets within the downcomer liquid region and their deceleration to terminal velocity, if attained, will cause depression of that liquid region. The extent of that liquid region needs to be calculated since it affects the amount of vapor produced.

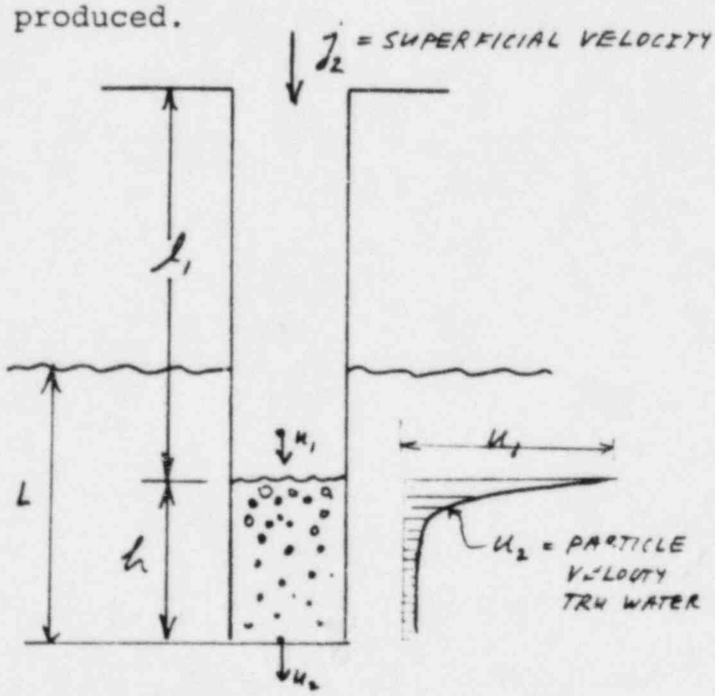


FIGURE 1.

Static force balance on column "h"

Let force due to deceleration of particles, per unit area, $\equiv F_1$

$$F_1 = j_2\rho_m(U_1-U_2)$$

F_2 = gravity force of metal/water mixture

$$= \bar{\rho}hg$$

$$F_3 = \rho_L Lg$$

Force balance: $F_1 + F_2 = F_3$

Metal volume in h is $j_2 \cdot A_p \cdot \frac{h}{\bar{U}_2}$

Liquid volume in h is $A_p h - j_2 A_p \frac{h}{\bar{U}_2}$

Hence
$$\bar{\rho} = \rho_m \frac{j_2}{\bar{u}_2} + \rho_L\left(1 - \frac{j_2}{\bar{u}_2}\right)$$

$$= \rho_L + (\rho_m - \rho_L)\frac{j_2}{\bar{u}_2}$$

where we have ignored the fact that some vapor will also be present in h. $\bar{U}_2$ is the mean particle velocity in h. Therefore,

$$j_2 \rho_m (u_1 - u_2) + (\rho_m - \rho_L) h g \frac{j_2}{\bar{u}_2} + \rho_L h g = \rho_L L g$$

$$h = \frac{\rho_L L - \rho_m (u_1 - u_2) j_2/g}{\rho_L + (\rho_m - \rho_L)\frac{j_2}{\bar{u}_2}} \qquad \ldots (46)$$

Time to traverse h, $\Delta t = \frac{h}{\bar{U}_2}$ $\qquad \ldots (47)$

Particle velocity, U_2, could be found from the force balance

$$\frac{4\pi}{3} r_m^3 \rho_m \frac{du_2}{dt} = - C_D \pi r_m^2 \frac{u_2^2}{2} \rho_L + \frac{4\pi}{3} r_m^3 (\rho_m - \rho_L) g$$

$$\frac{du_2}{dt} = - \frac{3}{8} C_D \frac{u_2^2}{r_m} \frac{\rho_L}{\rho_m} + \left(\frac{\rho_m - \rho_L}{\rho_m}\right) g \qquad \ldots \quad (48)$$

where r_m will be fixed to $r_m \simeq 0.00557$ m

$C_D = 0.5$

$\rho_m = 8600$ kg/m^3

$\rho_L = 950$ kg/m^3

$$\frac{du_2}{dt} = -3.72\, u_2^2 + 8.73 = 8.73\,(1 - 0.43\, u_2^2)$$

or

$$\frac{du_2}{0.43\, u_2^2 - 1} = -8.73\, dt \qquad \ldots \quad (49)$$

Let $\sqrt{0.45} \equiv \alpha = 0.6526$

Integration of (46) results in

$$\frac{1}{2\alpha} \log \frac{\alpha u_2 - 1}{\alpha u_2 + 1} = -8.7\, t + C$$

when $t = 0$, $U_2 = U_1$

$$\therefore \quad \log \left\{ \left(\frac{\alpha u_2 - 1}{\alpha u_2 + 1}\right) / \left(\frac{\alpha u_1 + 1}{\alpha u_1 - 1}\right) \right\} = -17.46\, \alpha\, t$$

$$\frac{\alpha u_2 - 1}{\alpha u_2 + 1} = \left(\frac{\alpha u_1 - 1}{\alpha u_1 + 1}\right) e^{-17.46\, \alpha\, t} \equiv \beta(t)$$

Hence

$$\alpha u_2 - 1 = (\alpha u_2 + 1) \cdot \beta(t)$$

$$u_2(t) = \frac{1}{\alpha} \left\{ \frac{1 + \beta(t)}{1 - \beta(t)} \right\} \qquad \ldots (50)$$

When $t = 0$, $\beta = \frac{\alpha u_1 - 1}{\alpha u_1 + 1}$

$$1 + \beta = \frac{\alpha u_1 + 1 + \alpha u_1 - 1}{\alpha u_1 + 1} = \frac{2 \alpha u_1}{\alpha u_1 + 1}$$

$$1 - \beta = \frac{2}{\alpha u_1 + 1}$$

$$\frac{1+\beta}{1-\beta} = \alpha u_1$$ and, $u_2 = u_1$, as desired

When $t \to \infty$, $\beta \to 0$

$$u_2 = \frac{1}{\alpha} = 1.53 \text{ m/sec}$$, for the assumed particle size

With $U_1 = 17$ m/sec, $\alpha = 0.6526$, $\alpha U_1 = 10.64$

$$\beta(t) = 0.83\, e^{-11.39t}$$

$$u_2(t) = 1.53 \left(\frac{1 + 0.83\, e^{-11.39t}}{1 - 0.83\, e^{-11.39t}} \right) \qquad \ldots (51)$$

t(secs)	$\beta(t)$	U_2(m/sec)	z(m)
0.01	0.741	10.27	0.14
0.02	0.66	7.48	0.33
0.04	0.53	4.98	0.45
0.06	0.42	3.74	0.54
0.08	0.33	3.03	0.61
0.10	0.27	2.65	0.67
0.20	0.09	1.83	0.89
0.40	0.0087	1.55	1.23
0.60	$8.9\text{x}10^{-4}$	1.53	1.54
0.80	0	1.53	1.85
1.00	0	1.53	2.15

where z = distance travelled through water (by numerical integration).

As can be seen, the particle would reach its terminal velocity of about 1.53 m/sec in about half a second, at which time it would have travelled a distance of about 1.35 meters. If none of the liquid were expelled from the downcomer (through bottom exit), it would take the particle about 1 second to traverse the initial liquid region within the downcomer.

To simplify the matter, let $\bar{U}_2 = U_2$ = terminal velocity of particles = 1.53 m/sec.

Since L = 8 ft = 2.44 m, equation (46) becomes

$$h = \frac{2318 - 13562\, j_2}{950 + 5000\, j_2} = 2.44\left(\frac{1 - 5.85\, j_2}{1 + 5.26\, j_2}\right) \qquad \ldots (52)$$

and $\Delta t = 0.65h$ $\ldots$ (53)

Equation (52) also implies that if $j_2 \geq \frac{1}{5.85}$ (m/sec), the liquid region h would disappear; i.e., metal droplets would not create steam within, but outside the downcomer. That steam would vent into the wetwell vapor space rather than through the downcomers because of lower resistance, thus eliminating flooding.

2.4.2 Flooding Criterion

Let j_1 denote the superficial velocity of upward flowing steam

$$j_1 = \frac{W_v}{\rho_v A_D}$$

W_v = mass rate of steam flow

A_D = downcomer flow area

ρ_v = steam density

Let j_2 = superficial velocity of particles

$$j_2 = \frac{Q_d}{4 A_D}$$

where Q_d was given by equation (22) for all four downcomers.

At the flooding point, the following relation applies

$$v_o^{1/2} = |j_1|^{\frac{1}{2}} + |j_2|^{\frac{1}{2}} \qquad \ldots (54)$$

where, according to Wallis (p. 383)

$$v_o \simeq v_\infty$$

and v_∞ is the same as V_r given by equation (44) except that ρ_v replaces ρ_L:

$$v_\infty \cong 2.8\,\rho_v^{-\frac{1}{2}}\left[\sigma_m g(\rho_m - \rho_v)\right]^{\frac{1}{4}} \qquad \ldots \quad (55)$$

Hence, from (44), setting $K_{FL} = \sqrt{v_\infty}$ $\qquad \ldots \quad (56)$

$$K_{FL} = \sqrt{j_1} + \sqrt{j_2} \qquad \ldots \quad (57)$$

Next, we turn our attention to the evaluation of j_1 from consideration of heat transfer.

2.4.3 Particle/Water Heat Transfer

Ignoring chemical reaction heat and decay heat generation due to very short times of interest, the transient heat conduction equation for the metal sphere, transferring heat by convection to the ambient at T_{sat}, becomes

$$\frac{\partial T_m}{\partial t} = \alpha_m\left(\frac{\partial^2 T_m}{\partial r^2} + \frac{2}{r}\frac{\partial T_m}{\partial r}\right), \quad 0 \le r < a \qquad \ldots \quad (58)$$

where $\quad$ a = radius of the sphere

Initial condition

$$T_m(r, 0) = T_{m0}$$

Boundary condition

$$\text{at } r=a: \quad k_m \frac{\partial T_m}{\partial r} + h(T_m - T_{sat}) = 0 \qquad \ldots (59)$$

where $\quad r = a$

$\quad h$ = (constant) convective heat transfer coefficient

Let $\quad v \equiv T_m - T_{sat}$

$$V \equiv T_{mo} - T_{sat} \qquad \ldots (60)$$

$$h' = h/k_m$$

Then

$$\frac{\partial v}{\partial t} = \kappa_m \left(\frac{\partial^2 v}{\partial r^2} + \frac{2}{r}\frac{\partial v}{\partial r} \right) \qquad \ldots (58a)$$

$$v(r,0) = V$$

$$\frac{\partial v}{\partial r} + h'v = 0 \quad \text{where } r=a \qquad \ldots (59a)$$

Analytic solution for this problem can be found in Carslaw and Jaeger, p. 238:

$$v(r,t) = \frac{2h'V}{r}\sum_{n=1}^{\infty} e^{-\kappa_m \alpha_n^2 t} \frac{a^2\alpha_n^2 + (ah'-1)^2}{\alpha_n^2[a^2\alpha_n^2 + ah'(ah'-1)]}(\sin a\alpha_n)\sin r\alpha_n \qquad \ldots (61)$$

where

α_n, n=1,2,..., are the roots of $a\alpha_n \cot(a\alpha_n) + (ah'-1) = 0$ $\qquad \ldots (62)$

The droplet surface superheat, $v(a,t)$, can be found from (51) by setting r = a, and the instantaneous heat flux to water is

$$q(t) = -k_m \left(\frac{\partial v}{\partial r}\right)_{r=a} \qquad \ldots (63)$$

Let

$$B_n \equiv (2h'V)e^{-\kappa_m \alpha_n^2 t} \frac{[a^2\alpha_n^2 + (ah'-1)^2]\sin(a\alpha_n)}{\alpha_n^2[a^2\alpha_n^2 + ah'(ah'-1)]} \quad \ldots \quad (64)$$

then

$$v(a,t) = \frac{1}{a}\sum_{n=1}^{\infty} B_n \sin(a\alpha_n) \quad \ldots \quad (65)$$

and, since

$$\frac{\partial v}{\partial r} = -\frac{1}{r^2}\sum_{n=1}^{\infty} B_n \sin r\alpha_n + \frac{1}{r}\sum_{n=1}^{\infty}\alpha_n B_n \cos r\alpha_n$$

from (53), we obtain

$$q(t) = -\frac{k_m}{a}\left\{\sum_{n=1}^{\infty}\alpha_n B_n \cos a\alpha_n - \frac{1}{a}\sum_{n=1}^{\infty} B_n \sin a\alpha_n\right\} \quad \ldots \quad (66)$$

$$\bar{q} = \frac{1}{\Delta t}\int_0^{\Delta t} q(t)\,dt \quad \ldots \quad (67)$$

In order to evaluate the heat transfer coefficient, h, we consider the fact that, at least initially, the particle surface superheat will be very high and, therefore, the heat transfer coefficient is a combination of the film boiling and radiation components. According to Collier, p. 218,

$$\text{if } \frac{u}{\sqrt{gD_m}} > 2.0$$

$$h = h_{FB,C} + 0.75\, h_r \quad \ldots \quad (68)$$

where

$$h_{FB,C} = 2.7\left(\frac{u\, k_v\, \rho_v\, i'_{fg}}{D_m\, \Delta T_{sat}}\right)^{\frac{1}{2}} \quad \ldots (69)$$

with $$i'_{fg} = i_{fg}\left[1 + 0.68\left(\frac{C_{pv}\,\Delta T_{sat}}{i_{fg}}\right)\right]$$

$$h_r = \sigma_r\, \varepsilon \left(\frac{T_m^4 - T_{sat}^4}{T_m - T_{sat}}\right) \quad \ldots (70)$$

With r_m = 0.00557 m, and U = 1.53 m/sec (assumed mean velocity $\bar{U}_2$)

$$\frac{u}{\sqrt{g D_m}} > 2$$ and the relations do apply.

With T_m = 1500°C, $T_{sat} \approx$ 120°C

V = 1380°C = ΔT_{sat} (Vapor at 0.2MPa)

$k_v \approx$ 0.6 W/m°C

ρ_v = 1.15 kg/m^3

i_{fg} = 2.1842x10^6 J/kg

C_{pv} = 2000 J/kg°C

σ_r = 5.727x10^{-8} $\frac{W}{m^2 k^4}$

ε = 0.7

$$h_{FB,C} = 1273.2 \quad W/(m^2\text{-}°C)$$

$$h_r = 110.2$$

$$\therefore \quad h = 1383.4 \left(\frac{W}{m^2 \cdot °C}\right) \quad \ldots (71)$$

This is a very large heat transfer coefficient. For comparison, the prequench h in fuel reflood tests is only about 50 W/m^2°C. If with this large h the particle surface superheat remains high enough to preclude nucleate boiling, the constant h assumption made in this analysis remains valid. The eigenvalues α_n, the terms β_n, the sums, the surface superheat V(a,t), the instantaneous heat flux q(t), and $\Delta t \cdot \bar{q}$ have been numerically evaluated on a computer (see Appendix I for listing). Results are tabulated below as functions of time.

Time (sec)	Surface Superheat (°C)	$q\left(\frac{W}{m^2}\right)$	$\int_0^t q\,dt$
0.1000000	1312.314	1815455.	181545.5
0.2000000	1281.450	1772758.	358821.3
0.3000000	1256.513	1738259.	532647.2
0.4000000	1234.596	1707940.	703441.2
0.5000000	1214.577	1680245.	871465.7
0.6000000	1195.879	1654378.	1036904.
0.7000000	1178.163	1629869.	1199891.
0.8000001	1161.208	1606415.	1360532.
0.9000001	1144.864	1583805.	1518913.
1.000000	1129.022	1561889.	1675101.
1.100000	1113.602	1540557.	1829157.
1.200000	1098.546	1519729.	1981130.
1.300000	1083.808	1499340.	2131064.
1.400000	1069.355	1479346.	2278999.
1.500000	1055.160	1459708.	2424969.
1.600000	1041.203	1440399.	2569009.
1.700000	1027.467	1421397.	2711149.
1.800000	1013.941	1402686.	2851418.
1.900000	1000.615	1384250.	2989843.
2.000000	987.4789	1366078.	3126450.
2.100000	974.5281	1348162.	3261266.
2.200000	961.7563	1330493.	3394316.
2.300000	949.1588	1313066.	3525622.
2.400000	936.7314	1295874.	3655210.
2.500000	924.4709	1278913.	3783101.
2.600000	912.3738	1262178.	3909319.
2.700000	900.4371	1245664.	4033885.
2.799999	888.6584	1229370.	4156822.
2.899999	877.0350	1213290.	4278151.
2.999999	865.5648	1197422.	4397893.
3.099999	854.2451	1181762.	4516069.
3.199999	843.0742	1166309.	4632700.
3.299999	832.0494	1151057.	4747806.
3.399999	821.1696	1136006.	4861406.
3.499999	810.4319	1121151.	4973521.
3.599999	799.8349	1106491.	5084170.
3.699999	789.3767	1092024.	5193373.
3.799999	779.0555	1077745.	5301147.
3.899998	768.8689	1063653.	5407513.
3.999998	758.8158	1049745.	5512487.
4.099998	748.8943	1036020.	5616089.

Other results, shown in Appendix I, pertain to (a) larger size (1 inch diameter) particles, same heat transfer coefficient; (b) same (small) particles, and smaller heat transfer coefficient; $h = 150\ W/m^2\,°C$.

The larger particles result in larger integrated heat flux. However, j_1 would now be smaller since the particle size appears in the denominator. Effects of smaller heat transfer coefficient (which is still 3 times larger than that observed in reflood experiments ahead of the quench front) are as expected .

2.4.4 Steam Generation Rate

Let n_m = number of particles contained within the submerged portion of the downcomer. If Δt is the time spent by the debris in the liquid region, see equations (52) and (53), then $A_d j_2 \Delta t$ is the volume of debris at any time in contact with water. That volume would contain a maximum of

$$\frac{A_d\, j_2\, \Delta t}{\frac{4\pi}{3} r_m^3} \quad \text{particles} \qquad \cdots (72)$$

and the total heat transfer surface A_{HT} becomes

$$A_{HT} = 4\pi r_m^2 \frac{A_d\, j_2\, \Delta t}{\frac{4\pi}{3} r_m^3} = \frac{3 A_d\, \Delta t}{r_m} j_2 \quad (m^2) \qquad \cdots (73)$$

Rate of total heat transferred to liquid

$$H_{HT} = A_{HT} \cdot \bar{q} = \frac{3 A_d\, \bar{q}\, \Delta t}{r_m} j_2 \quad (W) \qquad \cdots (74)$$

Heat balance for steam production

$$H_{HT} = W_v \left[C_{p_L} (T_{sat} - T_L) + h_{fg} \right]$$

Therefore, mass rate of steam produced

$$W_v = \frac{3 A_d \, \bar{q} \, \Delta t}{r_m \left[C_{p_L} (T_{sat} - T_L) + h_{fg} \right]} \, j_2 \qquad \ldots \quad (75)$$

and, since

$$j_1 = \frac{W_v}{\rho_v A_d} \qquad \ldots \quad (76)$$

$$\therefore \quad j_1 = K_{HT} \, j_2 \qquad \ldots \quad (77)$$

where utilizing eq. (67)

$$K_{HT} = \frac{3 \int_0^{\Delta t} q \, dt}{r_m \rho_v \left[C_{p_L} (T_{sat} - T_L) + h_{fg} \right]} \qquad \ldots \quad (78)$$

With $r_m = 0.00557$ m

$\rho_v = 1.15$ kg/m^3

$C_{\rho L} = 4185$ (J/kg°C)

$T_{sat} - T_L \approx 40$°C

$h_{fg} = 2.1842 \times 10^6$ J/kg

$$K_{HT} = 1.99 \times 10^{-4} \int_0^{\Delta t} q \, dt \qquad \ldots \quad (79)$$

Substituting equation (52) into (53)

$$\Delta t = 1.586 \left(\frac{1 - 5.85\, j_2}{1 + 5.26\, j_2} \right) \qquad \ldots \quad (80)$$

hence, $\Delta t_{max} = 1.586$ sec.

j_2 (m/sec)	Δt (sec)	$\int_0^{\Delta t} q\, dt \times 10^{-6}$ $(h = 1383\ \frac{W}{m^2\,°C})$	K_{HT} (eq. 78)	j_1 (eq. 77)
0				0
0.02	1.27	2.08608	415.13	8.3026
0.04	1.0062	1.6755	333.4	13.337
0.06	0.7844	1.310	260.7	15.64
0.08	0.5954	1.0293	204.8	16.386
0.10	0.4324	0.7579	150.8	15.08
0.12	0.2905	0.5161	102.7	12.325
0.14	0.1657	0.2980	59.3	8.303
0.15	0.1089	0.1973	39.3	5.889
0.1709	0	0	0	0

This heat transfer determined relationship is plotted in Figure 2. Note that the assumption of no subcooling would increase j_1 by only 7.7%.

We shall next plot the hydrodynamically determined relationship between j_1 and j_2 coming from the flooding correlation

$$\sqrt{v_\infty} = \sqrt{j_1} + \sqrt{j_2} \qquad \ldots \quad (57)$$

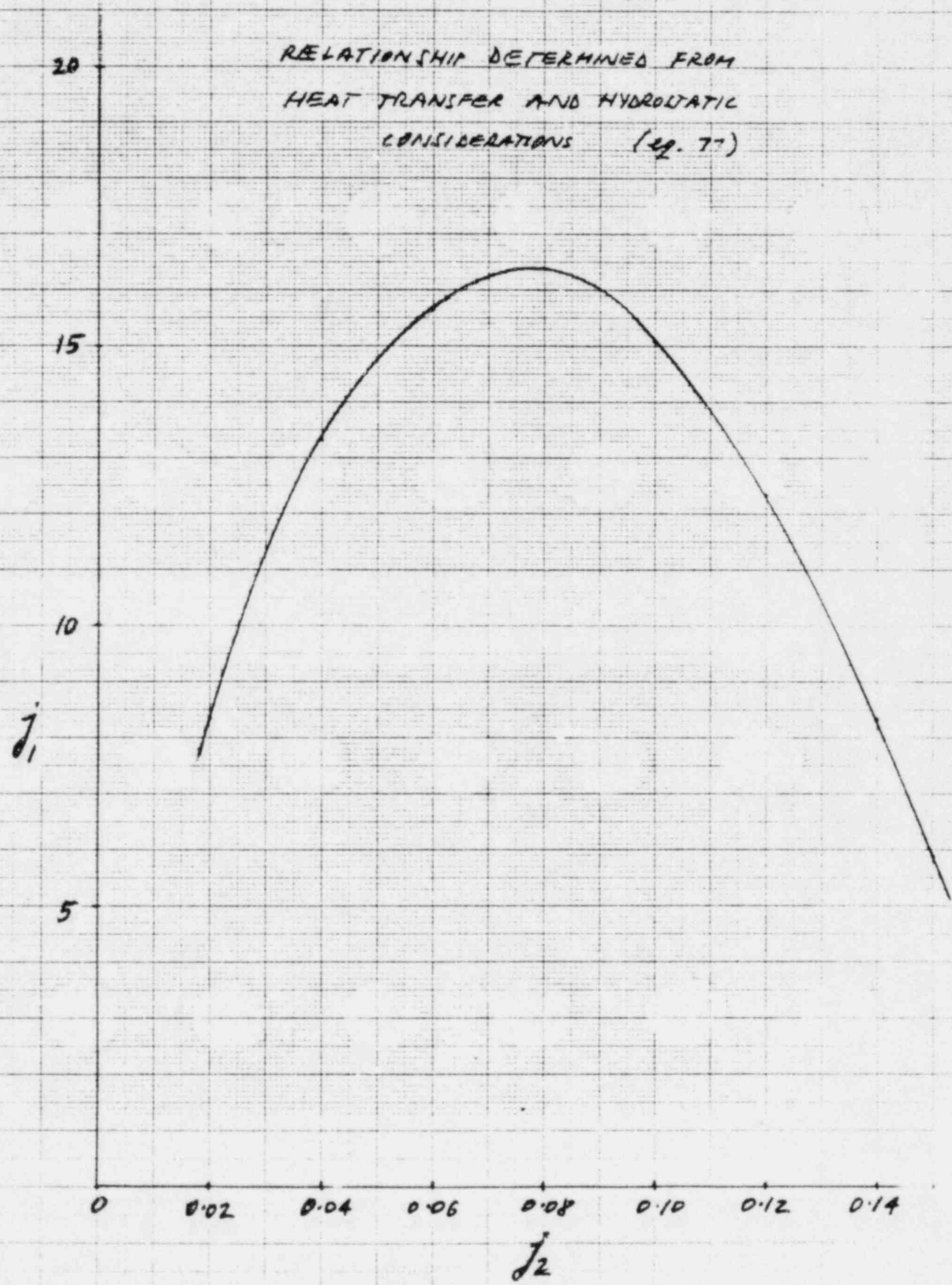
where from equation (55), $v_\infty = 36.26$, $\sqrt{v_\infty} = 6.02$. Hence,

$$j_1 = (6.06 - \sqrt{j_2})^2 \qquad \ldots \quad (81)$$

That relationship is plotted in Figure 3.

FIGURE 2

RELATIONSHIP DETERMINED FROM HEAT TRANSFER AND HYDROSTATIC CONSIDERATIONS (eq. 7?)



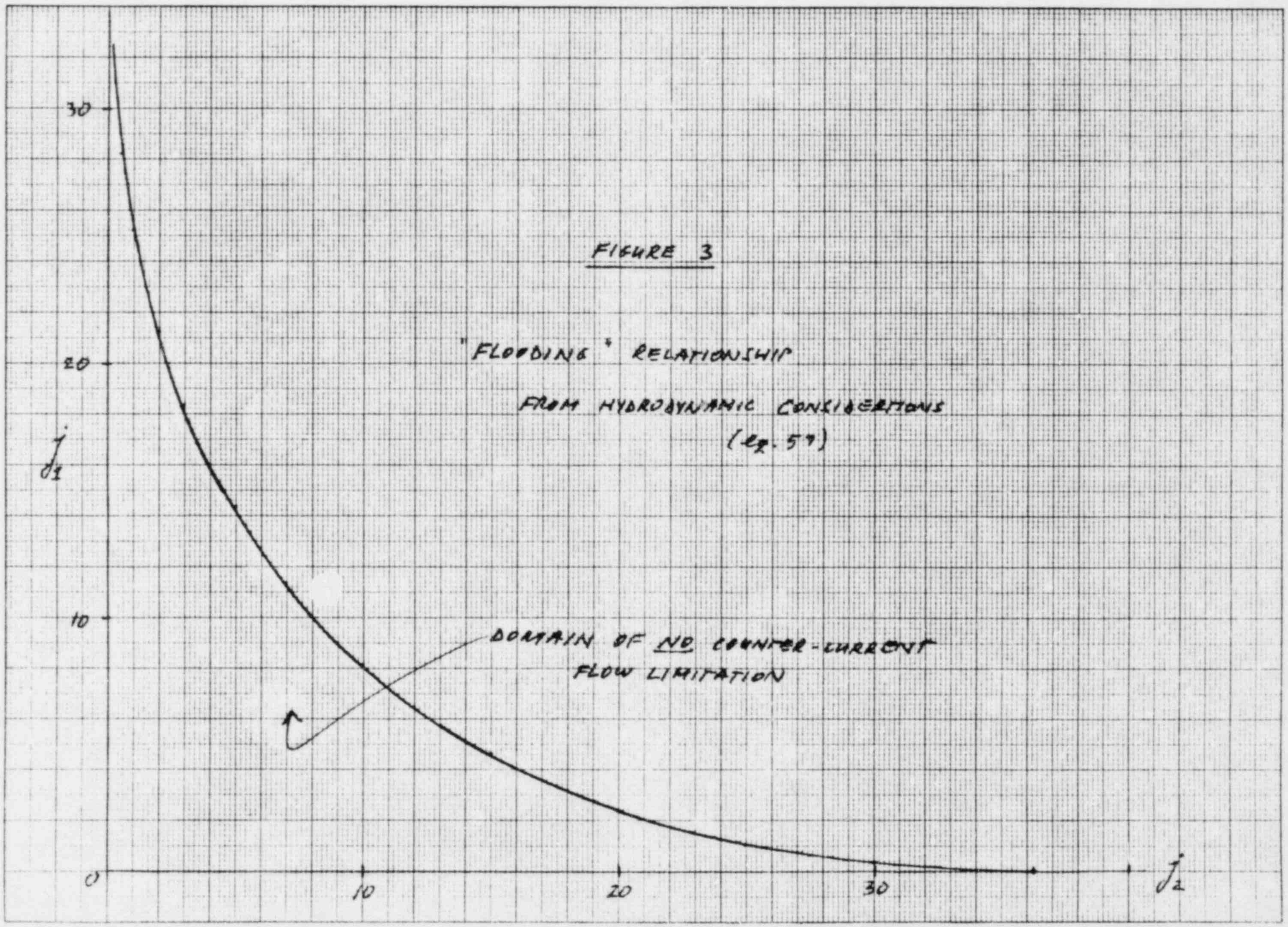


FIGURE 3

"FLOODING" RELATIONSHIP FROM HYDRODYNAMIC CONSIDERATIONS (eq. 57)

Inspection of Figures 2 and 3 indicates that the heat transfer induced vapor flow lies below the vapor flow that would cause countercurrent flow limitation, even with the high heat transfer coefficient assumed in this analysis. This indicates that no flooding restriction will take place.

If, for whatever reasons, one would postulate some limitation to the magnitude of metal downflow--note that complete absence of j_2 is not possible since, in that case, there would be no vapor generation to inhibit downflow--one could easily visualize that heat transfer between the downflowing hot debris and the relatively thin downcomer pipe wall (3/8 inches) would quickly impair the pipe wall strength and the downcomer would collapse into the wetwell pool. Any vapor generation within the pool, due to the falling debris, would then vent into the wetwell's vapor space, thus, again precluding flooding conduction from occurring.

2.5 Numerical Evaluation of Debris Transport in CRD Room

The following ten cases were numerically evaluated.

Case No.	Melt Mass in RPV(kg)	Pressurized Discharge	Gravity Discharge	Number of Ruptured CRD Tubes	Downcomer Cover Blocking Entry Until Melted
1	200,000	Yes	---	5	No
2	200,000	---	Yes	5	No
3	200,000	Yes	---	5	Yes (390 sec delay)
4	200,000	Yes	---	5	Yes (300 sec delay)
5	250,000	Yes	---	5	No
6	250,000	---	Yes	5	No
7	250,000	Yes	---	10	No
8	250,000	---	Yes	10	No
9	250,000	Yes	---	1	No
10	250,000	---	Yes	1	No

Detailed listing of results and of the computer program, CMELT, that incorporated the equations developed in this report are given in Appendix II.

The most interesting result shared by all of these cases is that nearly all of the discharged debris ends up in the wetwell. The only cases showing minute fractions of the discharged debris entering the drywell, through the manway, are Case Nos. 4, 5, and 7, with the computed fractions equal to 0.007, 0.005, and 0.001, respectively. These fractions are much smaller than those expected through the entrainment process discussed by Henry. He concluded that in the case of the plant that does not feature downcomers in the reactor cavity region, sufficient depth of molten debris could accumulate on the cavity room floor so that a significant fraction, up to 15%, could be entrained by hot gases discharged through the RPV ruptures following the melt discharge.

Due to the four downcomers in Shoreham, the depth of debris in the CRD room is never very high. Therefore, we expect less than 10% melt entrainment, reaching the drywell floor, and only for the pressurized discharge cases.

All cases were terminated when less than ½-inch-thickness of molten layer remained on the CRD room floor. That accounts for the remainder of the discharged material.

Other results of interest:

a) Time to discharge the melt from RPV vessel

Time to Complete Discharge (sec)	Pressurized Discharge	Gravity Discharge	Assumed Number of CRD Tube Ruptured (each 10 cm diameter vertically)	Assumed Initial Mass of Melt (kg)	Final (Total) Break Flow Area (m^2)
5	X	-	5	2×10^5	0.556
28	-	X	5	2×10^5	0.449
5	X	-	5	2.5×10^5	0.651
29	-	X	5	2.5×10^5	0.526
4	X	-	10	2.5×10^5	0.779
21	-	X	10	2.5×10^5	0.681
10	X	-	1	2.5×10^5	0.367
60	-	X	1	2.5×10^5	0.302

b) Maximum depth of debris on CRD room floor

Case No.	Approximate Maximum Depth (cm)	Approximate Time of Maximum Depth (sec)
1	59	4
2	16	22
3	59	4
4	59	4
5	59	4
6	18	23
7	71	3
8	21	17
9	51	9
10	13	30

It will be noticed that only in Case Nos. 3, 4, and 5 does the maximum depth approach the height of the manway sill (60.95 cm). Actually, the maximum debris depth must have briefly exceeded the sill height because minute amount of discharge over the sill was computed for these cases.

That fraction clearly depends on the assumptions regarding (a) when would the sill plate melt and/or (b) whether the sill plate could become dislodged prior to its melting. We have no information from which to judge the probability of the latter.

2.6 Brief Review of the In-Vessel and Ex-Vessel Explosion Probabilities

Those interested in an indepth analysis of this topic should refer to Henry's report cited earlier. We have noticed that SAI also dismissed consideration of such so-called "steam explosions" in their PRA report. Our reasons for agreeing with their position are as follows:

(a) In-vessel explosion:

Not likely because the RPV bottom head/lower plenum region is crowded with the control rods, their guide tubes, and other supporting structure. Coherent slump of core onto the bottom head does not appear possible and noncoherent immersions of molten chunks could not produce any serious explosions. Furthermore, there appears to be no clear path for producing accelerating water slugs that could impact the RPV upper head wall and create missiles.

(b) Explosion in the CRD room:

The four downcomers located in that room guarantee that if any water were to collect on the CRD room floor prior to the vessel wall failure, its depth could not exceed more than 3½ inches. This depth is too shallow to create steam explosions that may endanger containment integrity.

(c) Explosion in wetwell pool:

The ratio: water volume to volume of melt reaching the wetwell floor, far exceeds those ratios which could not sustain steam explosions under laboratory conditions. In addition, the noncoherent "dump," as calculated for the downcomer flow, is also not conducive to energetic explosions that may breach the containment, either due to missiles or shock waves.

2.7 Overall Conclusions and Recommendations

(1) It is our impression that the CRD cooling water injection was not accounted for in the SAI analysis. Its impact would be to delay the time of the RPV breach.

(2) Our analysis indicates that most of the molten debris discharged from the reactor vessel will end up, within about 5 minutes after the RPV rupture, in the wetwell pool.

(3) Consequently, the amount of the debris reaching the drywell floor, outside the pedestal area, would be so small that, when spread over the floor, it would be coolable by convective and radiative heat transfer to the containment atmosphere, and by conduction to concrete (neglecting presence of water). In other words, it appears likely that the temperature of that melt fraction will quickly fall below that required for initiation of concrete attack. If that is true, the amount of noncondensible gases generated by the concrete interaction on the drywell may be very small.

(4) During the 5 minutes, or less, that it takes for the discharged melt to drain into the CRD room downcomers, the melt/concrete interaction can indeed take place, however, over the much smaller floor area. The CRD room wetwell floor area is nearly 20 times smaller than the drywell floor area outside the pedestal room.

(5) During its transport to the wetwell floor, the molten debris surface temperature should be cooled to below that required for significant metal/water reaction and for the concrete

interaction. We could not address the subsequent heat balance for the wetwell water, steam production and the containment pressure rise because the conclusions are strongly influenced by the magnitude of heat leakage to and through the containment walls.

We recommend that SAI be asked to repeat the MARCH calculation, except, this time, to assume that about 90% of the released melt be transported directly to the wetwell pool, within 5 minutes after vessel rupture, to examine the effect on containment pressure.

We believe, however, that there will be no debris/concrete interaction on the wetwell floor due to low debris temperature.

(6) In our opinion, the RPV failure would not come about by the failure of the vessel wall but, rather, through rupture/melt-through of one or more control rod guide tubes. That, however, plays a minor role in the overall scenario since all of the debris would be discharged in less than ½ minute.

3.0 IMPACT OF RESULTS OF THIS STUDY ON SAI RESULTS CONCERNING ACCIDENT CONSEQUENCES

Class I Scenario

Expect very significant delay in the containment rupture time. If heat leakage to containment walls is significant, the containment may not rupture. It will certainly not rupture if containment sprays are operational. Need containment code analyses for quantification. Large decontamination factor is appropriate for fission products (FP) released within the wetwell pool.

Class II Scenario

No significant impact, except for scrubbing action of FP within the pool.

Class III Scenario

Same impact as for the Class I scenario.

Class IV Scenario

No impact on containment failure (which occurs before vessel melt-through). However, the subsequent high pressure peak, around 120 minutes, may not occur because of very much smaller production of noncondensible gases. This would significantly affect the transport of fission products through the ruptured containment. Containment code calculations are needed to quantify the pressure history. Wetwell pool scrubbing of FP.

Class V Scenario

Some details would change after the vessel melt-through. However, no significant impact since the containment pressure remains low. FP decontamination within the wetwell pool and much smaller production of noncondensible gases, would cause lower venting rates from containment through ruptured vessel (if flow passages not blocked) to the secondary containment.

4.0 LIST OF OTHER REFERENCES PERUSED DURING THIS STUDY

J. G. Collier, Convective Boiling and Condensation, McGraw-Hill, 1972.

G. B. Wallis, One-Dimensional Two-Phase Flow, McGraw-Hill, 1969.

Carslaw and Jaeger, Conduction of Heat in Solids, Second Edition, Oxford Press, 1959.

APPENDIX I

Listing of Computer Program HTRANS Developed for this Study, and Results

(a) HTRANS Listing

```
          PROGRAM HTRANS
          DIMENSION ALFN(6),BB(9),CC(9)
          DATA CC/-1.0, -0.95, -0.9, -0.5, 0.0, 0.5, 1.0, 10., 100./
          DATA BB/0., 0.3854, 0.5423, 1.1656, 1.5708, 1.8366, 2.20288,
         * 2.8628, 3.1105/
C
 10       CONTINUE
          QTOT=0.0
          NITER=0
          TYPE 300
 300      FORMAT(1H1,//'   ENTER H(W/M**2,DEG C) AND PARTICLE RADIUS A(M)'/)
          ACCEPT*,H,A
          IF(H.EQ.0)STOP
          DIF=4.167E-6
          COND=22.
          HP=H/COND
          AHP=A*HP
          C=AHP-1.
          DO 100 I=1,9
          IF(CC(I).GE.C)GO TO 102
 100      CONTINUE
 102      CONTINUE
          BO=BB(I-1)+(BB(I)-BB(I-1))*(C-CC(I-1))/(CC(I)-CC(I-1))
          J=1
 105      CONTINUE
          NITER=NITER+1
          COTB=COS(BO)/SIN(BO)
          BN=BO-(BO*COTB+C)/(COTB*(1.-BO*COTB)-BO)
          IF(ABS((BN-BO)/BO).LE.1.0E-5)GO TO 110
          IF(NITER.GE.20)GO TO 400
          BO=BN
          GO TO 105
```

```
  400   TYPE 410
  410   FORMAT('  NUMBER OF ITERATIONS MORE THAN 20')
        STOP
  110   ALFN(J)=BN/A
        NITER=0
        IF(J.EQ.6)GO TO 120
        J=J+1
        BO=BN+3.1416
        GO TO 105
  120   CONTINUE
        TYPE 200,(ALFN(I),I=1,6)
  200   FORMAT(' EIGENVALUES ALFN=',6F10.4/)
        TMO=1773.
        TSAT=393.
        V=TMO-TSAT
        T=0.0
        DT=0.10
C
  1     T=T+DT
        SUMVS=0.
        SUMQ1=0.
        SUMQ2=0.
C
        DO 2 J=1,6
        ALF=ALFN(J)
        ALFS=ALF*ALF
        AN=ALF*A
        ANS=AN*AN
        E=-DIF*ALFS*T
        EX=0.
        IF(-E.LE.50.)EX=EXP(E)
        DEN=ALFS*(ANS+AHP*C)
        XK=2.*HP*V*EX*(ANS+C*C)*SIN(AN)/DEN
        SUMVS=SUMVS+XK*SIN(AN)
        SUMQ1=SUMVS
        SUMQ2=SUMQ2+ALF*XK*COS(AN)
  2     CONTINUE
C
        VS=SUMVS/A
        Q=-COND/A*(SUMQ2-SUMQ1/A)
        QTOT=QTOT+Q*DT
        TYPE *,T,VS,Q,QTOT
        IF(T.GE.4.0)GO TO 10
        GO TO 1
        END
```

(b) Additional Results

Case 2: Heat transfer coefficient, h= 1383.4(W/sq.m -C)
Particle radius = 0.0127 m (1 inch diam.)

First column: Elapsed time (secs)
Second column: Spherical particle surface temperature, deg C
3rd and 4th col.: q (W/m^2), and $\int q\,dt$, respectively

```
ENTER H(W/M**2,DEG C) AND PARTICLE RADIUS A(M)

1383., 0.0127
EIGENVALUES ALFN=  112.6610  367.6561  616.3961  864.3471 1112.0386
```

t(sec)	V(°C)	q	$\int q\,dt$
0.1000000	1312.516	1815209.	181520.9
0.2000000	1288.523	1782027.	359723.6
0.3000000	1268.529	1754375.	535161.1
0.4000000	1251.203	1730414.	708202.4
0.5000000	1235.757	1709051.	879107.6
0.6000000	1221.699	1689610.	1048069.
0.7000000	1208.712	1671649.	1215234.
0.8000001	1196.579	1654868.	1380720.
0.9000001	1185.144	1639054.	1544626.
1.000000	1174.295	1624049.	1707031.
1.100000	1163.946	1609737.	1868004.
1.200000	1154.030	1596023.	2027607.
1.300000	1144.493	1582834.	2185890.
1.400000	1135.294	1570111.	2342901.
1.500000	1126.395	1557804.	2498682.
1.600000	1117.768	1545873.	2653269.
1.700000	1109.387	1534282.	2806697.
1.800000	1101.231	1523002.	2958997.
1.900000	1093.281	1512007.	3110198.
2.000000	1085.521	1501276.	3260326.
2.100000	1077.937	1490787.	3409404.
2.200000	1070.517	1480525.	3557457.
2.300000	1063.249	1470473.	3704504.
2.400000	1056.123	1460619.	3850566.
2.500000	1049.132	1450949.	3995661.
2.600000	1042.266	1441453.	4139806.
2.700000	1035.518	1432121.	4283018.
2.799999	1028.882	1422944.	4425313.
2.899999	1022.353	1413914.	4566704.
2.999999	1015.924	1405022.	4707206.
3.099999	1009.590	1396263.	4846833.
3.199999	1003.347	1387630.	4985596.
3.299999	997.1914	1379116.	5123507.
3.399999	991.1183	1370716.	5260579.
3.499999	985.1243	1362427.	5396821.
3.599999	979.2057	1354242.	5532245.
3.699999	973.3600	1346157.	5666861.
3.799999	967.5840	1338169.	5800678.
3.899998	961.8746	1330273.	5933705.
3.999998	956.2301	1322466.	6065952.
4.099998	950.6473	1314745.	6197426.

Case 3: Heat transfer coefficient, h=150 W/m^2-deg C
Particle radius = 0.00557 m

t (sec)	V (°C)	q (W/m^2)	$\int_0^t q\,dt$
0.1000000	1372.410	205861.1	20586.11
0.2000000	1368.777	205315.9	41117.70
0.3000000	1365.763	204864.7	61604.17
0.4000000	1363.056	204457.3	82049.90
0.5000000	1360.534	204079.9	102457.9
0.6000000	1358.138	203720.3	122829.9
0.7000000	1355.829	203374.6	143167.4
0.8000001	1353.586	203037.7	163471.2
0.9000001	1351.392	202708.9	183742.0
1.000000	1349.235	202384.5	203980.5
1.100000	1347.107	202065.8	224187.1
1.200000	1345.001	201750.6	244362.1
1.300000	1342.914	201437.2	264505.9
1.400000	1340.840	201125.8	284618.5
1.500000	1338.779	200816.8	304700.2
1.600000	1336.727	200508.8	324751.0
1.700000	1334.683	200201.7	344771.[illegible]
1.800000	1332.646	199896.1	364760.8
1.900000	1330.614	199591.4	384720.0
2.000000	1328.589	199287.7	404648.8
2.100000	1326.568	198985.0	424547.3
2.200000	1324.551	198682.3	444415.5
2.300000	1322.538	198380.5	464253.6
2.400000	1320.529	198078.7	484061.4
2.500000	1318.524	197778.9	503839.3
2.600000	1316.522	197477.6	523587.1
2.700000	1314.524	197177.8	543304.8
2.799999	1312.528	196879.4	562992.8
2.899999	1310.536	196580.5	582650.8
2.999999	1308.547	196281.6	602279.0
3.099999	1306.562	195983.7	621877.4
3.199999	1304.579	195686.3	641446.0
3.299999	1302.600	195390.3	660985.0
3.399999	1300.623	195093.3	680494.3
3.499999	1298.649	194797.4	699974.1
3.599999	1296.679	194501.4	719424.2
3.699999	1294.712	194206.4	738844.8
3.799999	1292.747	193912.3	758236.1
3.899998	1290.786	193617.3	777597.8
3.999998	1288.828	193324.2	796930.3
4.099998	1286.872	193030.6	816233.3

APPENDIX II

Listing and Results, of Computer Program CMELT, to Compute Transport of Molten Debris

```
      PROGRAM CMELT
C
1     CONTINUE
      II=0
      IF(T.EQ.0.) GO TO 304
      SUMSIL=SUMSIL*8600./EMMO
      SUMWW=SUMWW*8600./EMMO
      TYPE 305,SUMSIL,SUMWW
304   CONTINUE
305   FORMAT(/' ACCUMULATED FRACTION OF  DEBRIS IN:'/
     *'  DRYWELL(OUTSIDE CRD ROOM)=',E15.5/
     *'  WETWELL POOL=',E15.5)
      TYPE 1000
1000  FORMAT(1H1//,' ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS,
     * RBRKO')
      ACCEPT*,IPRESS,DT,TMSUP,TMCOVR,TMSILL,NBRKS,RBRKO
      IF(IPRESS.LT.0)STOP
      TYPE 1100
1100  FORMAT(' ENTER INITIAL MOLTEN MASS IN RPV(KG)')
      ACCEPT*,EMMO
      IF(IPRESS.EQ.1)TYPE 100
      IF(IPRESS.EQ.0)TYPE 110
      TYPE 200
      T=0.0
      DTPRNT=1.0
      TPRINT=1.0
      HRPV=1.855*EMMO/2.E+5
      EMM=EMMO
      HCRD=0.0
      SUMSIL=0.
      SUMWW=0.
      JITER=0
2000  CONTINUE
      JITER=JITER+1
      HRPVS=HRPV*HRPV
      F=2.27*HRPVS-0.333*HRPVS*HRPV-3.7013E-5*EMM
      DF=5.54*HRPV-HRPVS


      HRPVN=HRPV-F/DF
      IF(ABS((HRPVN-HRPV)/HRPV).LE.1.E-5)GO TO 2100
      IF(JITER.LT.50)GO TO 2050
      IF(IPRESS.EQ.0)STOP
      GO TO 20
2050  CONTINUE
      HRPV=HRPVN
      GO TO 2000
2100  CONTINUE
```

```
      IF(IPRESS.EQ.1)GO TO 20
      SUMO=SQRT(HRPV)
      TMLTMN=TMCOVR
      IF(TMSILL.LT.TMCOVR)TMLTMN=TMSILL
10    CONTINUE
C
      T=T+DT
      IF(IPRESS.EQ.0)GO TO 12
C----PRESSURIZED DISCHARGE
C
      QBRK=0.
      IF(HRPV.LT.1.0E-2)GO TO 20
      RBRK=RBRKO+0.0307*T
      R2=RBRK*RBRK
      ABRK=NBRKS*3.1416*R2
      QBRK=75.4*NBRKS*R2
      EMM=6.484E+5*NBRKS*T*(2.5E-3+1.535E-3*T+3.1416E-4*T*T)
      IF(EMM.GE.EMMO)QBRK=0.
      IF(EMM.GE.EMMO)EMM=EMMO
      JITER=0
      EMM=EMMO-EMM
      GO TO 2000
12    CONTINUE
C
C----GRAVITY DISCHARGE
C
      QBRK=0.
      IF(HRPV.LE.0.)GO TO 14
      Y=SQRT(HRPV)
      SUMM=SUMO+0.5*DT*(Y+YO)
      RBRK=RBRKO+3.443E-3*SUMM
      R2=RBRK*RBRK
      ABRK=NBRKS*3.1416*R2
      TERM=4.43*NBRKS*DT/(Y*(5.54-HRPV))
      HRPV=HRPV-TERM*R2
      IF(HRPV.LT.0.)HRPV=0.0
      YO=Y
      SUMO=SUMM
      QBRK=8.35*NBRKS*R2*Y
14    CONTINUE



20    CONTINUE
C
C---MELT FLOW THROUGH DOWNCOMERS
C
      QDWNC=0.
      IF(HCRD.LE.0.114.AND.T.LE.5.)GO TO 30
      IF(II.EQ.1)GO TO 23
      TMSUP=T
      II=1
23    CONTINUE
      IF(T.GT.TMSUP)HTD=0.038*(1.-T/300.)
      IF(HTD.LT.0.)HTD=0.
      IF(TMCOVR.GT.0.)GO TO 21
      GO TO 22
```

```
21    CONTINUE
      IF(T.GT.TMSUP+10..AND.T.LT.TMSUP+TMCOVR)GO TO 30
      IF(T.GE.TMSUP+TMCOVR)HTD=0.
22    CONTINUE
      IF(HCRD-HTD.LE.0.)GO TO 30
      QDWNC=22.62*(HCRD-HTD)**1.5
30    CONTINUE
C
C---MELT FLOW PAST MANWAY SILL
C
      QSILL=0.
      IF(HCRD.LT.0.6096.AND.T.LT.TMSILL)GO TO 40
      IF(T.LT.TMSILL)QSILL=1.8*(HCRD-0.6096)**1.5
      IF(T.GE.TMSILL)QSILL=1.8*HCRD**1.5
40    CONTINUE
      SUMSIL=SUMSIL+QSILL*DT
      SUMWW=SUMWW+QDWNC*DT
C
C---NOW FIND THE MELT POOL LEVEL IN THE CRD ROOM
C
      HCRD=HCRD+DT/23.13*(QBRK-QDWNC-QSILL)
      IF(T.LT.TPRINT)GO TO 42
      TYPE 300,T,RBRK,ABRK,HRPV,HCRD,QBRK,QDWNC,QSILL
      IF(T.GE.30.)DTPRNT=10.
      IF(HCRD.LE.0.0127.AND.T.GT.20.)GO TO 1
      TPRINT=T+DTPRNT
42    CONTINUE
      GO TO 10
C
100   FORMAT(/'     PRESSURIZED DISCHARGE CASE'/)
110   FORMAT(/'     GRAVITY DISCHARGE CASE'/)
105   FORMAT('  DT=',F5.3,' TMSUP, TMCOVR, TMSILL=',3F10.3,
     *'  NBRKS=',I5,' RBRK0=',F10.5/)
200   FORMAT(' T(SEC)    RBRK(M)  ABRK(M*M)    HRPV(M)    HCRD(M)
     *  QBRK(M**3/SEC) QDWNC        QSILL'/)
300   FORMAT(F10.1,4F10.3,3E12.5)

      END
```

Nomenclature for CMELT Input

IPRESS= Indicator; =1 for pressurized discharge, =0 for gravity discharge
DT = Integration time increments (secs)
TMSUP = Time to melt cover support columns. Not utilized from input.
TMCOVR= Time to melt/remove downcomer cover plates in CRD room
TMSILL= Time to melt the manway sill
NBRKS = Assumed number of ruptured CRD tubes
RBRKO = Initial radius of each break (m)

Nomenclature for CMELT Output

T = Elapsed time, secs, from start of RPV rupture
RBRK = Instantaneous radius (m) of each "hole" in RPV bottom head
ABRK = Instantaneous, total break flow area (sq.m)
HRPV = Height of molten debris pool within reactor vessel (m)
HCRD = Height of molten debris pool on CRD room floor (m)
QBRK = Volumetric flow of melt through vessel rupture (m^3/sec)
QDWNC = Volumetric flow of melt through 4 downcomers in CRD room
QSILL = Volumetric flow of melt through CRD room manway

Case 1. See page 60 in text for case description

```
ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
1, 0.5, 50., 0., 390., 5, 0.05
ENTER INITIAL MOLTEN MASS IN RPV(KG)
2.0E+5

    PRESSURIZED DISCHARGE CASE
```

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.081	0.102	2.091	0.088	0.24552E+01	0.00000E+00	0.00000E+00
2.0	0.111	0.195	1.871	0.232	0.46786E+01	0.14894E+01	0.00000E+00
3.0	0.142	0.317	1.465	0.412	0.76125E+01	0.34131E+01	0.00000E+00
4.0	0.173	0.469	0.640	0.590	0.11257E+02	0.71587E+01	0.00000E+00
5.0	0.188	0.556	0.000	0.287	0.00000E+00	0.47241E+01	0.00000E+00
6.0	0.188	0.556	0.000	0.186	0.00000E+00	0.18573E+01	0.00000E+00
7.0	0.188	0.556	0.000	0.137	0.00000E+00	0.94989E+00	0.00000E+00
8.0	0.188	0.556	0.000	0.110	0.00000E+00	0.55922E+00	0.00000E+00
9.0	0.188	0.556	0.000	0.092	0.00000E+00	0.36016E+00	0.00000E+00
10.0	0.188	0.556	0.000	0.081	0.00000E+00	0.24705E+00	0.00000E+00
11.0	0.188	0.556	0.000	0.072	0.00000E+00	0.17763E+00	0.00000E+00
12.0	0.188	0.556	0.000	0.066	0.00000E+00	0.13248E+00	0.00000E+00
13.0	0.188	0.556	0.000	0.061	0.00000E+00	0.10177E+00	0.00000E+00
14.0	0.188	0.556	0.000	0.058	0.00000E+00	0.80105E-01	0.00000E+00
15.0	0.188	0.556	0.000	0.055	0.00000E+00	0.64362E-01	0.00000E+00
16.0	0.188	0.556	0.000	0.052	0.00000E+00	0.52634E-01	0.00000E+00
17.0	0.188	0.556	0.000	0.050	0.00000E+00	0.43708E-01	0.00000E+00
18.0	0.188	0.556	0.000	0.049	0.00000E+00	0.36789E-01	0.00000E+00
19.0	0.188	0.556	0.000	0.047	0.00000E+00	0.31341E-01	0.00000E+00

t	P_{BRK}	A_{BRK}	H_{RPV}	H_{CRD}	Q_{BRK}	Q_{DOWNC}	Q_{SILL}
20.0	0.188	0.556	0.000	0.046	0.00000E+00	0.26990E-01	0.00000E+00
21.0	0.188	0.556	0.000	0.045	0.00000E+00	0.23471E-01	0.00000E+00
22.0	0.188	0.556	0.000	0.044	0.00000E+00	0.20595E-01	0.00000E+00
23.0	0.188	0.556	0.000	0.043	0.00000E+00	0.18220E-01	0.00000E+00
24.0	0.188	0.556	0.000	0.043	0.00000E+00	0.16241E-01	0.00000E+00
25.0	0.188	0.556	0.000	0.042	0.00000E+00	0.14579E-01	0.00000E+00
26.0	0.188	0.556	0.000	0.041	0.00000E+00	0.13173E-01	0.00000E+00
27.0	0.188	0.556	0.000	0.041	0.00000E+00	0.11975E-01	0.00000E+00
28.0	0.188	0.556	0.000	0.040	0.00000E+00	0.10949E-01	0.00000E+00
29.0	0.188	0.556	0.000	0.040	0.00000E+00	0.10064E-01	0.00000E+00
30.0	0.188	0.556	0.000	0.040	0.00000E+00	0.92973E-02	0.00000E+00
40.0	0.188	0.556	0.000	0.037	0.00000E+00	0.52526E-02	0.00000E+00
50.0	0.188	0.556	0.000	0.035	0.00000E+00	0.39006E-02	0.00000E+00
60.0	0.188	0.556	0.000	0.033	0.00000E+00	0.33615E-02	0.00000E+00
70.0	0.188	0.556	0.000	0.032	0.00000E+00	0.31274E-02	0.00000E+00
80.0	0.188	0.556	0.000	0.030	0.00000E+00	0.30214E-02	0.00000E+00
90.0	0.188	0.556	0.000	0.029	0.00000E+00	0.29726E-02	0.00000E+00
100.0	0.188	0.556	0.000	0.028	0.00000E+00	0.29498E-02	0.00000E+00
110.0	0.188	0.556	0.000	0.027	0.00000E+00	0.29392E-02	0.00000E+00
120.0	0.188	0.556	0.000	0.025	0.00000E+00	0.29342E-02	0.00000E+00
130.0	0.188	0.556	0.000	0.024	0.00000E+00	0.29319E-02	0.00000E+00
140.0	0.188	0.556	0.000	0.023	0.00000E+00	0.29308E-02	0.00000E+00
150.0	0.188	0.556	0.000	0.021	0.00000E+00	0.29303E-02	0.00000E+00
160.0	0.188	0.556	0.000	0.020	0.00000E+00	0.29300E-02	0.00000E+00
170.0	0.188	0.556	0.000	0.019	0.00000E+00	0.29299E-02	0.00000E+00
180.0	0.188	0.556	0.000	0.018	0.00000E+00	0.29299E-02	0.00000E+00
190.0	0.188	0.556	0.000	0.016	0.00000E+00	0.29298E-02	0.00000E+00
200.0	0.188	0.556	0.000	0.015	0.00000E+00	0.29298E-02	0.00000E+00
210.0	0.188	0.556	0.000	0.014	0.00000E+00	0.29298E-02	0.00000E+00
220.0	0.188	0.556	0.000	0.013	0.00000E+00	0.29298E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.00000E+00
WETWELL POOL= 0.98706E+00

Case 2

```
ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
0,,,,,,,,,,,,,,,
ENTER INITIAL MOLTEN MASS IN RPV(KG)
,,
```

GRAVITY DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.059	0.055	2.178	0.009	0.21426E+00	0.00000E+00	0.00000E+00
2.0	0.064	0.064	2.160	0.019	0.25187E+00	0.00000E+00	0.00000E+00
3.0	0.069	0.075	2.140	0.032	0.29198E+00	0.00000E+00	0.00000E+00
4.0	0.074	0.086	2.116	0.046	0.33439E+00	0.00000E+00	0.00000E+00
5.0	0.079	0.098	2.089	0.061	0.37885E+00	0.00000E+00	0.00000E+00
6.0	0.084	0.111	2.059	0.071	0.42509E+00	0.10141E+00	0.00000E+00
7.0	0.089	0.124	2.024	0.084	0.47277E+00	0.18720E+00	0.00000E+00
8.0	0.094	0.139	1.986	0.095	0.52157E+00	0.27563E+00	0.00000E+00
9.0	0.099	0.153	1.944	0.105	0.57106E+00	0.36037E+00	0.00000E+00
10.0	0.104	0.168	1.898	0.113	0.62083E+00	0.43902E+00	0.00000E+00
11.0	0.108	0.184	1.847	0.120	0.67037E+00	0.51132E+00	0.00000E+00
12.0	0.113	0.201	1.792	0.126	0.71917E+00	0.57792E+00	0.00000E+00
13.0	0.118	0.217	1.732	0.132	0.76662E+00	0.63964E+00	0.00000E+00
14.0	0.122	0.234	1.667	0.137	0.81209E+00	0.69714E+00	0.00000E+00
15.0	0.127	0.252	1.598	0.142	0.85486E+00	0.75075E+00	0.00000E+00

t	R_{BRK}	A_{BRK}	H_{RAV}	H_{CRD}	Q_{BRK}	Q_{DWNC}	Q_{SILL}
16.0	0.131	0.269	1.523	0.146	0.89414E+00	0.80051E+00	0.00000E+00
17.0	0.135	0.287	1.442	0.149	0.92906E+00	0.84616E+00	0.00000E+00
18.0	0.139	0.305	1.356	0.153	0.95862E+00	0.88721E+00	0.00000E+00
19.0	0.143	0.323	1.263	0.155	0.98168E+00	0.92296E+00	0.00000E+00
20.0	0.147	0.340	1.164	0.157	0.99691E+00	0.95250E+00	0.00000E+00
21.0	0.151	0.358	1.058	0.159	0.10027E+01	0.97473E+00	0.00000E+00
22.0	0.154	0.375	0.944	0.159	0.99701E+00	0.98828E+00	0.00000E+00
23.0	0.158	0.391	0.821	0.159	0.97714E+00	0.99148E+00	0.00000E+00
24.0	0.161	0.407	0.686	0.158	0.93917E+00	0.98211E+00	0.00000E+00
25.0	0.164	0.421	0.536	0.155	0.87676E+00	0.95713E+00	0.00000E+00
26.0	0.166	0.434	0.365	0.149	0.77771E+00	0.91177E+00	0.00000E+00
27.0	0.168	0.445	0.150	0.141	0.60950E+00	0.83715E+00	0.00000E+00
28.0	0.169	0.449	0.000	0.118	0.00000E+00	0.70817E+00	0.00000E+00
29.0	0.169	0.449	0.000	0.097	0.00000E+00	0.43909E+00	0.00000E+00
30.0	0.169	0.449	0.000	0.083	0.00000E+00	0.29311E+00	0.00000E+00
40.0	0.169	0.449	0.000	0.044	0.00000E+00	0.28922E-01	0.00000E+00
50.0	0.169	0.449	0.000	0.037	0.00000E+00	0.96451E-02	0.00000E+00
60.0	0.169	0.449	0.000	0.034	0.00000E+00	0.53561E-02	0.00000E+00
70.0	0.169	0.449	0.000	0.032	0.00000E+00	0.39395E-02	0.00000E+00
80.0	0.169	0.449	0.000	0.031	0.00000E+00	0.33779E-02	0.00000E+00
90.0	0.169	0.449	0.000	0.029	0.00000E+00	0.31347E-02	0.00000E+00
100.0	0.169	0.449	0.000	0.028	0.00000E+00	0.30248E-02	0.00000E+00
110.0	0.169	0.449	0.000	0.027	0.00000E+00	0.29741E-02	0.00000E+00
120.0	0.169	0.449	0.000	0.025	0.00000E+00	0.29506E-02	0.00000E+00
130.0	0.169	0.449	0.000	0.024	0.00000E+00	0.29395E-02	0.00000E+00
140.0	0.169	0.449	0.000	0.023	0.00000E+00	0.29344E-02	0.00000E+00
150.0	0.169	0.449	0.000	0.021	0.00000E+00	0.29319E-02	0.00000E+00
160.0	0.169	0.449	0.000	0.020	0.00000E+00	0.29308E-02	0.00000E+00
170.0	0.169	0.449	0.000	0.019	0.00000E+00	0.29303E-02	0.00000E+00
180.0	0.169	0.449	0.000	0.018	0.00000E+00	0.29300E-02	0.00000E+00
190.0	0.169	0.449	0.000	0.016	0.00000E+00	0.29299E-02	0.00000E+00
200.0	0.169	0.449	0.000	0.015	0.00000E+00	0.29298E-02	0.00000E+00
210.0	0.169	0.449	0.000	0.014	0.00000E+00	0.29298E-02	0.00000E+00
220.0	0.169	0.449	0.000	0.013	0.00000E+00	0.29298E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.00000E+00
WETWELL POOL= 0.80383E+00

ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
1, 0.5, 50., 300., 390., 5, 0.05
ENTER INITIAL MOLTEN MASS IN RPV(KG)
2.0E+5

Case 3

PRESSURIZED DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.081	0.102	2.091	0.088	0.24552E+01	0.00000E+00	0.00000E+00
2.0	0.111	0.195	1.871	0.235	0.46786E+01	0.13528E+01	0.00000E+00
3.0	0.142	0.317	1.465	0.413	0.76125E+01	0.34492E+01	0.00000E+00
4.0	0.173	0.469	0.640	0.591	0.11257E+02	0.71742E+01	0.00000E+00
5.0	0.188	0.556	0.000	0.287	0.00000E+00	0.47272E+01	0.00000E+00
6.0	0.188	0.556	0.000	0.186	0.00000E+00	0.18581E+01	0.00000E+00
7.0	0.188	0.556	0.000	0.137	0.00000E+00	0.95020E+00	0.00000E+00
8.0	0.188	0.556	0.000	0.110	0.00000E+00	0.55937E+00	0.00000E+00
9.0	0.188	0.556	0.000	0.092	0.00000E+00	0.36024E+00	0.00000E+00
10.0	0.188	0.556	0.000	0.081	0.00000E+00	0.24710E+00	0.00000E+00
11.0	0.188	0.556	0.000	0.072	0.00000E+00	0.17766E+00	0.00000E+00
12.0	0.188	0.556	0.000	0.066	0.00000E+00	0.13250E+00	0.00000E+00
13.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
14.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
15.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
16.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
17.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
18.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
19.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
20.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
21.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
22.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
23.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
24.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
25.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
26.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
27.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
28.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
29.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
30.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
40.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
50.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
60.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
70.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
80.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
90.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00

t	R_{BRK}	A_{BRK}	H_{RS}	H_{CRD}	Q_{BRK}	Q_{DWNC}	Q_{SILL}
100.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
110.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
120.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
130.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
140.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
150.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
160.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
170.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
180.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
190.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
200.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
210.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
220.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
230.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
240.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
250.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
260.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
270.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
280.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
290.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
300.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
310.0	0.188	0.556	0.000	0.014	0.00000E+00	0.42915E-01	0.00000E+00
320.0	0.188	0.556	0.000	0.006	0.00000E+00	0.99492E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.00000E+00
WETWELL POOL= 0.99409E+00

Case 4

```
ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
1, 0.5, 50., 300., 300., 5, 0.05
ENTER INITIAL MOLTEN MASS IN RPV(KG)
2.0E+5
```

PRESSURIZED DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.081	0.102	2.091	0.088	0.24552E+01	0.00000E+00	0.00000E+00
2.0	0.111	0.195	1.871	0.232	0.46786E+01	0.14894E+01	0.00000E+00
3.0	0.142	0.317	1.465	0.412	0.76125E+01	0.34131E+01	0.00000E+00
4.0	0.173	0.469	0.640	0.590	0.11257E+02	0.71587E+01	0.00000E+00
5.0	0.188	0.556	0.000	0.287	0.00000E+00	0.47241E+01	0.00000E+00
6.0	0.188	0.556	0.000	0.186	0.00000E+00	0.18573E+01	0.00000E+00
7.0	0.188	0.556	0.000	0.137	0.00000E+00	0.94989E+00	0.00000E+00
8.0	0.188	0.556	0.000	0.110	0.00000E+00	0.55922E+00	0.00000E+00
9.0	0.188	0.556	0.000	0.092	0.00000E+00	0.36016E+00	0.00000E+00
10.0	0.188	0.556	0.000	0.081	0.00000E+00	0.24705E+00	0.00000E+00
11.0	0.188	0.556	0.000	0.072	0.00000E+00	0.17763E+00	0.00000E+00
12.0	0.188	0.556	0.000	0.066	0.00000E+00	0.13248E+00	0.00000E+00
13.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
14.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
15.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
16.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
17.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
18.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
19.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
20.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
21.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
22.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
23.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
24.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
25.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00

t (secs)	R_{BRK} (m)	A_{BRK} (m^2)	[illegible]	H_{CRD} (m)	Q_{BRK}	Q_{DWC}	Q_{SILL}
26.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
27.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
28.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
29.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
30.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
40.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
50.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
60.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
70.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
80.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
90.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
100.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
110.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
120.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
130.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
140.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
150.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
160.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
170.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
180.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
190.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
200.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
210.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
220.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
230.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
240.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
250.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
260.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
270.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
280.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
290.0	0.188	0.556	0.000	0.066	0.00000E+00	0.00000E+00	0.00000E+00
300.0	0.188	0.556	0.000	0.065	0.00000E+00	0.00000E+00	0.30595E-01
310.0	0.188	0.556	0.000	0.013	0.00000E+00	0.36795E-01	0.29280E-02
320.0	0.188	0.556	0.000	0.005	0.00000E+00	0.82752E-02	0.65850E-03

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.68688E-02
WETWELL POOL= 0.98786E+00

Case 5

ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
1, 0.5, 50., 0., 300., 5, 0.05
ENTER INITIAL MOLTEN MASS IN RPV(KG)
2.5E+5

PRESSURIZED DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.081	0.102	2.451	0.088	0.24552E+01	0.00000E+00	0.00000E+00
2.0	0.111	0.195	2.234	0.232	0.46786E+01	0.14894E+01	0.00000E+00
3.0	0.142	0.317	1.853	0.412	0.76125E+01	0.34131E+01	0.00000E+00
4.0	0.173	0.469	1.200	0.590	0.11257E+02	0.71587E+01	0.00000E+00
5.0	0.204	0.651	0.000	0.427	0.00000E+00	0.11595E+02	0.32112E-01
6.0	0.204	0.651	0.000	0.239	0.00000E+00	0.31832E+01	0.00000E+00
7.0	0.204	0.651	0.000	0.164	0.00000E+00	0.14144E+01	0.00000E+00
8.0	0.204	0.651	0.000	0.125	0.00000E+00	0.76909E+00	0.00000E+00
9.0	0.204	0.651	0.000	0.102	0.00000E+00	0.47023E+00	0.00000E+00
10.0	0.204	0.651	0.000	0.087	0.00000E+00	0.31081E+00	0.00000E+00
11.0	0.204	0.651	0.000	0.077	0.00000E+00	0.21731E+00	0.00000E+00
12.0	0.204	0.651	0.000	0.070	0.00000E+00	0.15856E+00	0.00000E+00
13.0	0.204	0.651	0.000	0.064	0.00000E+00	0.11966E+00	0.00000E+00
14.0	0.204	0.651	0.000	0.060	0.00000E+00	0.92808E-01	0.00000E+00
15.0	0.204	0.651	0.000	0.056	0.00000E+00	0.73645E-01	0.00000E+00
16.0	0.204	0.651	0.000	0.054	0.00000E+00	0.59582E-01	0.00000E+00
17.0	0.204	0.651	0.000	0.052	0.00000E+00	0.49016E-01	0.00000E+00
18.0	0.204	0.651	0.000	0.050	0.00000E+00	0.40918E-01	0.00000E+00
19.0	0.204	0.651	0.000	0.048	0.00000E+00	0.34602E-01	0.00000E+00
20.0	0.204	0.651	0.000	0.047	0.00000E+00	0.29601E-01	0.00000E+00
21.0	0.204	0.651	0.000	0.046	0.00000E+00	0.25588E-01	0.00000E+00
22.0	0.204	0.651	0.000	0.045	0.00000E+00	0.22329E-01	0.00000E+00
23.0	0.204	0.651	0.000	0.044	0.00000E+00	0.19654E-01	0.00000E+00

t	R_{BRK}	A_{BRK}	H_{RDW}	H_{CRD}	Q_{BRK}	Q_{DWNC}	Q_{SILL}
24.0	0.204	0.651	0.000	0.043	0.00000E+00	0.17438E-01	0.00000E+00
25.0	0.204	0.651	0.000	0.042	0.00000E+00	0.15586E-01	0.00000E+00
26.0	0.204	0.651	0.000	0.042	0.00000E+00	0.14026E-01	0.00000E+00
27.0	0.204	0.651	0.000	0.041	0.00000E+00	0.12703E-01	0.00000E+00
28.0	0.204	0.651	0.000	0.041	0.00000E+00	0.11573E-01	0.00000E+00
29.0	0.204	0.651	0.000	0.040	0.00000E+00	0.10603E-01	0.00000E+00
30.0	0.204	0.651	0.000	0.040	0.00000E+00	0.97644E-02	0.00000E+00
40.0	0.204	0.651	0.000	0.037	0.00000E+00	0.53913E-02	0.00000E+00
50.0	0.204	0.651	0.000	0.035	0.00000E+00	0.39527E-02	0.00000E+00
60.0	0.204	0.651	0.000	0.033	0.00000E+00	0.33834E-02	0.00000E+00
70.0	0.204	0.651	0.000	0.032	0.00000E+00	0.31371E-02	0.00000E+00
80.0	0.204	0.651	0.000	0.030	0.00000E+00	0.30259E-02	0.00000E+00
90.0	0.204	0.651	0.000	0.029	0.00000E+00	0.29747E-02	0.00000E+00
100.0	0.204	0.651	0.000	0.028	0.00000E+00	0.29508E-02	0.00000E+00
110.0	0.204	0.651	0.000	0.027	0.00000E+00	0.29396E-02	0.00000E+00
120.0	0.204	0.651	0.000	0.025	0.00000E+00	0.29344E-02	0.00000E+00
130.0	0.204	0.651	0.000	0.024	0.00000E+00	0.29320E-02	0.00000E+00
140.0	0.204	0.651	0.000	0.023	0.00000E+00	0.29308E-02	0.00000E+00
150.0	0.204	0.651	0.000	0.021	0.00000E+00	0.29303E-02	0.00000E+00
160.0	0.204	0.651	0.000	0.020	0.00000E+00	0.29300E-02	0.00000E+00
170.0	0.204	0.651	0.000	0.019	0.00000E+00	0.29299E-02	0.00000E+00
180.0	0.204	0.651	0.000	0.018	0.00000E+00	0.29299E-02	0.00000E+00
190.0	0.204	0.651	0.000	0.016	0.00000E+00	0.29298E-02	0.00000E+00
200.0	0.204	0.651	0.000	0.015	0.00000E+00	0.29298E-02	0.00000E+00
210.0	0.204	0.651	0.000	0.014	0.00000E+00	0.29298E-02	0.00000E+00
220.0	0.204	0.651	0.000	0.013	0.00000E+00	0.29298E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.55232E-03
WETWELL POOL= 0.10186E+01

Case 6.

```
ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
0,,,,,,,,,,,,,,,
ENTER INITIAL MOLTEN MASS IN RPV(KG)
,,
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GRAVITY DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.060	0.056	2.537	0.010	0.23945E+00	0.00000E+00	0.00000E+00
2.0	0.065	0.067	2.518	0.022	0.28427E+00	0.00000E+00	0.00000E+00
3.0	0.071	0.079	2.495	0.036	0.33235E+00	0.00000E+00	0.00000E+00
4.0	0.076	0.092	2.470	0.052	0.38343E+00	0.00000E+00	0.00000E+00
5.0	0.082	0.105	2.440	0.070	0.43724E+00	0.00000E+00	0.00000E+00
6.0	0.087	0.119	2.406	0.080	0.49345E+00	0.15156E+00	0.00000E+00
7.0	0.092	0.134	2.368	0.094	0.55168E+00	0.25282E+00	0.00000E+00
8.0	0.098	0.150	2.326	0.105	0.61153E+00	0.35409E+00	0.00000E+00
9.0	0.103	0.167	2.280	0.115	0.67253E+00	0.45014E+00	0.00000E+00
10.0	0.108	0.184	2.228	0.124	0.73417E+00	0.53950E+00	0.00000E+00
11.0	0.113	0.202	2.172	0.132	0.79588E+00	0.62253E+00	0.00000E+00
12.0	0.118	0.220	2.111	0.138	0.85705E+00	0.70020E+00	0.00000E+00
13.0	0.123	0.239	2.044	0.145	0.91702E+00	0.77342E+00	0.00000E+00
14.0	0.128	0.259	1.973	0.151	0.97503E+00	0.84279E+00	0.00000E+00
15.0	0.133	0.279	1.896	0.156	0.10303E+01	0.90854E+00	0.00000E+00
16.0	0.138	0.299	1.813	0.161	0.10819E+01	0.97056E+00	0.00000E+00
17.0	0.143	0.319	1.725	0.165	0.11290E+01	0.10284E+01	0.00000E+00
18.0	0.147	0.340	1.631	0.169	0.11703E+01	0.10815E+01	0.00000E+00
19.0	0.151	0.360	1.530	0.173	0.12048E+01	0.11290E+01	0.00000E+00
20.0	0.156	0.381	1.423	0.176	0.12310E+01	0.11698E+01	0.00000E+00
21.0	0.160	0.401	1.309	0.178	0.12473E+01	0.12028E+01	0.00000E+00

t	R_{BRK}	A_{BRK}	$H_{[illegible]}$	H_{CRD}	Q_{BRK}	Q_{DOWNC}	Q_{SILL}
22.0	0.164	0.421	1.187	0.179	0.12517E+01	0.12265E+01	0.00000E+00
23.0	0.168	0.441	1.057	0.179	0.12419E+01	0.12392E+01	0.00000E+00
24.0	0.171	0.460	0.916	0.179	0.12143E+01	0.12387E+01	0.00000E+00
25.0	0.174	0.478	0.763	0.177	0.11643E+01	0.12224E+01	0.00000E+00
26.0	0.177	0.494	0.594	0.173	0.10840E+01	0.11863E+01	0.00000E+00
27.0	0.180	0.509	0.401	0.166	0.95804E+00	0.11245E+01	0.00000E+00
28.0	0.182	0.521	0.159	0.156	0.74502E+00	0.10252E+01	0.00000E+00
29.0	0.183	0.526	0.000	0.128	0.00000E+00	0.85493E+00	0.00000E+00
30.0	0.183	0.526	0.000	0.103	0.00000E+00	0.51308E+00	0.00000E+00
40.0	0.183	0.526	0.000	0.046	0.00000E+00	0.35696E-01	0.00000E+00
50.0	0.183	0.526	0.000	0.038	0.00000E+00	0.10777E-01	0.00000E+00
60.0	0.183	0.526	0.000	0.034	0.00000E+00	0.56827E-02	0.00000E+00
70.0	0.183	0.526	0.000	0.032	0.00000E+00	0.40605E-02	0.00000E+00
80.0	0.183	0.526	0.000	0.031	0.00000E+00	0.34286E-02	0.00000E+00
90.0	0.183	0.526	0.000	0.029	0.00000E+00	0.31572E-02	0.00000E+00
100.0	0.183	0.526	0.000	0.028	0.00000E+00	0.30351E-02	0.00000E+00
110.0	0.183	0.526	0.000	0.027	0.00000E+00	0.29789E-02	0.00000E+00
120.0	0.183	0.526	0.000	0.025	0.00000E+00	0.29528E-02	0.00000E+00
130.0	0.183	0.526	0.000	0.024	0.00000E+00	0.29406E-02	0.00000E+00
140.0	0.183	0.526	0.000	0.023	0.00000E+00	0.29349E-02	0.00000E+00
150.0	0.183	0.526	0.000	0.021	0.00000E+00	0.29322E-02	0.00000E+00
160.0	0.183	0.526	0.000	0.020	0.00000E+00	0.29309E-02	0.00000E+00
170.0	0.183	0.526	0.000	0.019	0.00000E+00	0.29303E-02	0.00000E+00
180.0	0.183	0.526	0.000	0.018	0.00000E+00	0.29301E-02	0.00000E+00
190.0	0.183	0.526	0.000	0.016	0.00000E+00	0.29299E-02	0.00000E+00
200.0	0.183	0.526	0.000	0.015	0.00000E+00	0.29299E-02	0.00000E+00
210.0	0.183	0.526	0.000	0.014	0.00000E+00	0.29298E-02	0.00000E+00
220.0	0.183	0.526	0.000	0.013	0.00000E+00	0.29298E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.00000E+00
WETWELL POOL= 0.82247E+00

Case 7

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ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
1, 0.5, 50., 0., 300., 10, 0.05
ENTER INITIAL MOLTEN MASS IN RPV(KG)
2.5E+5
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PRESSURIZED DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.081	0.205	2.349	0.176	0.49104E+01	0.00000E+00	0.00000E+00
2.0	0.111	0.390	1.913	0.432	0.93571E+01	0.29200E+01	0.00000E+00
3.0	0.142	0.634	1.046	0.711	0.15225E+02	0.88599E+01	0.00000E+00
4.0	0.157	0.779	0.000	0.315	0.00000E+00	0.57621E+01	0.00000E+00
5.0	0.157	0.779	0.000	0.198	0.00000E+00	0.21148E+01	0.00000E+00
6.0	0.157	0.779	0.000	0.144	0.00000E+00	0.10481E+01	0.00000E+00
7.0	0.157	0.779	0.000	0.114	0.00000E+00	0.60570E+00	0.00000E+00
8.0	0.157	0.779	0.000	0.095	0.00000E+00	0.38525E+00	0.00000E+00
9.0	0.157	0.779	0.000	0.082	0.00000E+00	0.26188E+00	0.00000E+00
10.0	0.157	0.779	0.000	0.074	0.00000E+00	0.18699E+00	0.00000E+00
11.0	0.157	0.779	0.000	0.067	0.00000E+00	0.13871E+00	0.00000E+00
12.0	0.157	0.779	0.000	0.062	0.00000E+00	0.10608E+00	0.00000E+00
13.0	0.157	0.779	0.000	0.058	0.00000E+00	0.83189E-01	0.00000E+00
14.0	0.157	0.779	0.000	0.055	0.00000E+00	0.66630E-01	0.00000E+00
15.0	0.157	0.779	0.000	0.053	0.00000E+00	0.54340E-01	0.00000E+00
16.0	0.157	0.779	0.000	0.051	0.00000E+00	0.45017E-01	0.00000E+00
17.0	0.157	0.779	0.000	0.049	0.00000E+00	0.37812E-01	0.00000E+00
18.0	0.157	0.779	0.000	0.048	0.00000E+00	0.32151E-01	0.00000E+00
19.0	0.157	0.779	0.000	0.046	0.00000E+00	0.27640E-01	0.00000E+00
20.0	0.157	0.779	0.000	0.045	0.00000E+00	0.24000E-01	0.00000E+00
21.0	0.157	0.779	0.000	0.044	0.00000E+00	0.21029E-01	0.00000E+00
22.0	0.157	0.779	0.000	0.044	0.00000E+00	0.18580E-01	0.00000E+00

t	R_{BRK}	A_{BRK}		H_{CRD}	Q_{BRK}	Q_{DOWNC}	Q_{SILL}
23.0	0.157	0.779	0.000	0.043	0.00000E+00	0.16542E-01	0.00000E+00
24.0	0.157	0.779	0.000	0.042	0.00000E+00	0.14833E-01	0.00000E+00
25.0	0.157	0.779	0.000	0.042	0.00000E+00	0.13388E-01	0.00000E+00
26.0	0.157	0.779	0.000	0.041	0.00000E+00	0.12159E-01	0.00000E+00
27.0	0.157	0.779	0.000	0.041	0.00000E+00	0.11107E-01	0.00000E+00
28.0	0.157	0.779	0.000	0.040	0.00000E+00	0.10200E-01	0.00000E+00
29.0	0.157	0.779	0.000	0.040	0.00000E+00	0.94158E-02	0.00000E+00
30.0	0.157	0.779	0.000	0.039	0.00000E+00	0.87333E-02	0.00000E+00
40.0	0.157	0.779	0.000	0.037	0.00000E+00	0.50811E-02	0.00000E+00
50.0	0.157	0.779	0.000	0.035	0.00000E+00	0.38354E-02	0.00000E+00
60.0	0.157	0.779	0.000	0.033	0.00000E+00	0.33338E-02	0.00000E+00
70.0	0.157	0.779	0.000	0.032	0.00000E+00	0.31150E-02	0.00000E+00
80.0	0.157	0.779	0.000	0.030	0.00000E+00	0.30157E-02	0.00000E+00
90.0	0.157	0.779	0.000	0.029	0.00000E+00	0.29699E-02	0.00000E+00
100.0	0.157	0.779	0.000	0.028	0.00000E+00	0.29486E-02	0.00000E+00
110.0	0.157	0.779	0.000	0.027	0.00000E+00	0.29386E-02	0.00000E+00
120.0	0.157	0.779	0.000	0.025	0.00000E+00	0.29339E-02	0.00000E+00
130.0	0.157	0.779	0.000	0.024	0.00000E+00	0.29317E-02	0.00000E+00
140.0	0.157	0.779	0.000	0.023	0.00000E+00	0.29307E-02	0.00000E+00
150.0	0.157	0.779	0.000	0.021	0.00000E+00	0.29302E-02	0.00000E+00
160.0	0.157	0.779	0.000	0.020	0.00000E+00	0.29300E-02	0.00000E+00
170.0	0.157	0.779	0.000	0.019	0.00000E+00	0.29299E-02	0.00000E+00
180.0	0.157	0.779	0.000	0.018	0.00000E+00	0.29299E-02	0.00000E+00
190.0	0.157	0.779	0.000	0.016	0.00000E+00	0.29298E-02	0.00000E+00
200.0	0.157	0.779	0.000	0.015	0.00000E+00	0.29298E-02	0.00000E+00
210.0	0.157	0.779	0.000	0.014	0.00000E+00	0.29298E-02	0.00000E+00
220.0	0.157	0.779	0.000	0.013	0.00000E+00	0.29298E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.99300E-03
WETWELL POOL= 0.87961E+00

Case 8

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ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
0,,,,,,,,,,,,,,
ENTER INITIAL MOLTEN MASS IN RPV(KG)
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GRAVITY DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.060	0.113	2.521	0.020	0.47830E+00	0.00000E+00	0.00000E+00
2.0	0.065	0.135	2.483	0.043	0.56555E+00	0.00000E+00	0.00000E+00
3.0	0.071	0.158	2.439	0.071	0.65775E+00	0.00000E+00	0.00000E+00
4.0	0.076	0.183	2.387	0.102	0.75393E+00	0.00000E+00	0.00000E+00
5.0	0.082	0.209	2.329	0.120	0.85296E+00	0.81938E+00	0.00000E+00
6.0	0.087	0.237	2.263	0.135	0.95357E+00	0.62014E+00	0.00000E+00
7.0	0.092	0.266	2.189	0.148	0.10543E+01	0.76868E+00	0.00000E+00
8.0	0.097	0.296	2.107	0.159	0.11536E+01	0.90487E+00	0.00000E+00
9.0	0.102	0.327	2.017	0.169	0.12496E+01	0.10300E+01	0.00000E+00
10.0	0.107	0.359	1.918	0.178	0.13404E+01	0.11453E+01	0.00000E+00
11.0	0.112	0.392	1.810	0.186	0.14236E+01	0.12512E+01	0.00000E+00
12.0	0.116	0.425	1.693	0.192	0.14968E+01	0.13472E+01	0.00000E+00
13.0	0.121	0.459	1.566	0.198	0.15569E+01	0.14320E+01	0.00000E+00
14.0	0.125	0.492	1.428	0.202	0.16008E+01	0.15034E+01	0.00000E+00
15.0	0.129	0.525	1.279	0.206	0.16241E+01	0.15588E+01	0.00000E+00
16.0	0.133	0.557	1.117	0.207	0.16218E+01	0.15945E+01	0.00000E+00
17.0	0.137	0.588	0.941	0.207	0.15865E+01	0.16061E+01	0.00000E+00
18.0	0.140	0.617	0.745	0.204	0.15074E+01	0.15875E+01	0.00000E+00
19.0	0.143	0.643	0.521	0.198	0.13647E+01	0.15296E+01	0.00000E+00
20.0	0.146	0.666	0.249	0.187	0.11103E+01	0.14165E+01	0.00000E+00
21.0	0.147	0.681	0.000	0.161	0.47266E+00	0.12077E+01	0.00000E+00
22.0	0.147	0.681	0.000	0.123	0.00000E+00	0.76048E+00	0.00000E+00
23.0	0.147	0.681	0.000	0.100	0.00000E+00	0.46586E+00	0.00000E+00
24.0	0.147	0.681	0.000	0.085	0.00000E+00	0.30835E+00	0.00000E+00

t	R_{BRK}	$\hat{A}_{BRK}$	H_{RPV}	H_{CRD}	Q_{BRK}	Q_{DOWNC}	Q_{SILL}
25.0	0.147	0.681	0.000	0.075	0.00000E+00	0.21580E+00	0.00000E+00
26.0	0.147	0.681	0.000	0.068	0.00000E+00	0.15758E+00	0.00000E+00
27.0	0.147	0.681	0.000	0.062	0.00000E+00	0.11899E+00	0.00000E+00
28.0	0.147	0.681	0.000	0.058	0.00000E+00	0.92342E-01	0.00000E+00
29.0	0.147	0.681	0.000	0.055	0.00000E+00	0.73307E-01	0.00000E+00
30.0	0.147	0.681	0.000	0.052	0.00000E+00	0.59330E-01	0.00000E+00
40.0	0.147	0.681	0.000	0.040	0.00000E+00	0.13996E-01	0.00000E+00
50.0	0.147	0.681	0.000	0.036	0.00000E+00	0.65376E-02	0.00000E+00
60.0	0.147	0.681	0.000	0.034	0.00000E+00	0.43650E-02	0.00000E+00
70.0	0.147	0.681	0.000	0.032	0.00000E+00	0.35536E-02	0.00000E+00
80.0	0.147	0.681	0.000	0.031	0.00000E+00	0.32123E-02	0.00000E+00
90.0	0.147	0.681	0.000	0.029	0.00000E+00	0.30602E-02	0.00000E+00
100.0	0.147	0.681	0.000	0.028	0.00000E+00	0.29905E-02	0.00000E+00
110.0	0.147	0.681	0.000	0.027	0.00000E+00	0.29582E-02	0.00000E+00
120.0	0.147	0.681	0.000	0.025	0.00000E+00	0.29431E-02	0.00000E+00
130.0	0.147	0.681	0.000	0.024	0.00000E+00	0.29360E-02	0.00000E+00
140.0	0.147	0.681	0.000	0.023	0.00000E+00	0.29327E-02	0.00000E+00
150.0	0.147	0.681	0.000	0.021	0.00000E+00	0.29312E-02	0.00000E+00
160.0	0.147	0.681	0.000	0.020	0.00000E+00	0.29304E-02	0.00000E+00
170.0	0.147	0.681	0.000	0.019	0.00000E+00	0.29301E-02	0.00000E+00
180.0	0.147	0.681	0.000	0.018	0.00000E+00	0.29299E-02	0.00000E+00
190.0	0.147	0.681	0.000	0.016	0.00000E+00	0.29299E-02	0.00000E+00
200.0	0.147	0.681	0.000	0.015	0.00000E+00	0.29298E-02	0.00000E+00
210.0	0.147	0.681	0.000	0.014	0.00000E+00	0.29298E-02	0.00000E+00
220.0	0.147	0.681	0.000	0.013	0.00000E+00	0.29298E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.00000E+00
WETWELL POOL= 0.83220E+00

Case 9.

```
ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
1, 0.5, 50., 0., 300., 1, 0.05
ENTER INITIAL MOLTEN MASS IN RPV(KG)
2.5E+5
```

PRESSURIZED DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.081	0.020	2.532	0.018	0.49104E+00	0.00000E+00	0.00000E+00
2.0	0.111	0.039	2.489	0.053	0.93571E+00	0.00000E+00	0.00000E+00
3.0	0.142	0.063	2.413	0.112	0.15225E+01	0.00000E+00	0.00000E+00
4.0	0.173	0.094	2.297	0.175	0.22514E+01	0.12131E+01	0.00000E+00
5.0	0.204	0.130	2.132	0.241	0.31225E+01	0.15887E+01	0.00000E+00
6.0	0.234	0.172	1.906	0.307	0.41357E+01	0.26050E+01	0.00000E+00
7.0	0.265	0.220	1.599	0.373	0.52910E+01	0.37704E+01	0.00000E+00
8.0	0.296	0.275	1.165	0.439	0.65884E+01	0.50648E+01	0.00000E+00
9.0	0.326	0.334	0.328	0.505	0.80280E+01	0.64872E+01	0.00000E+00
10.0	0.342	0.367	0.000	0.263	0.00000E+00	0.39353E+01	0.00000E+00
11.0	0.342	0.367	0.000	0.175	0.00000E+00	0.16410E+01	0.00000E+00
12.0	0.342	0.367	0.000	0.131	0.00000E+00	0.86361E+00	0.00000E+00
13.0	0.342	0.367	0.000	0.106	0.00000E+00	0.51735E+00	0.00000E+00
14.0	0.342	0.367	0.000	0.090	0.00000E+00	0.33717E+00	0.00000E+00
15.0	0.342	0.367	0.000	0.078	0.00000E+00	0.23329E+00	0.00000E+00
16.0	0.342	0.367	0.000	0.071	0.00000E+00	0.16885E+00	0.00000E+00
17.0	0.342	0.367	0.000	0.065	0.00000E+00	0.12660E+00	0.00000E+00
18.0	0.342	0.367	0.000	0.060	0.00000E+00	0.97677E-01	0.00000E+00
19.0	0.342	0.367	0.000	0.057	0.00000E+00	0.77164E-01	0.00000E+00
20.0	0.342	0.367	0.000	0.054	0.00000E+00	0.62191E-01	0.00000E+00
21.0	0.342	0.367	0.000	0.051	0.00000E+00	0.50994E-01	0.00000E+00
22.0	0.342	0.367	0.000	0.050	0.00000E+00	0.42445E-01	0.00000E+00
23.0	0.342	0.367	0.000	0.048	0.00000E+00	0.35801E-01	0.00000E+00

t	R_{BRK}	A_{BRK}		H_{CRD}	Q_{BRK}	Q_{DWNC}	Q_{SILL}
24.0	0.342	0.367	0.000	0.047	0.00000E+00	0.30556E-01	0.00000E+00
25.0	0.342	0.367	0.000	0.045	0.00000E+00	0.26358E-01	0.00000E+00
26.0	0.342	0.367	0.000	0.044	0.00000E+00	0.22957E-01	0.00000E+00
27.0	0.342	0.367	0.000	0.043	0.00000E+00	0.20172E-01	0.00000E+00
28.0	0.342	0.367	0.000	0.043	0.00000E+00	0.17869E-01	0.00000E+00
29.0	0.342	0.367	0.000	0.042	0.00000E+00	0.15947E-01	0.00000E+00
30.0	0.342	0.367	0.000	0.041	0.00000E+00	0.14331E-01	0.00000E+00
40.0	0.342	0.367	0.000	0.037	0.00000E+00	0.66212E-02	0.00000E+00
50.0	0.342	0.367	0.000	0.035	0.00000E+00	0.43939E-02	0.00000E+00
60.0	0.342	0.367	0.000	0.033	0.00000E+00	0.35653E-02	0.00000E+00
70.0	0.342	0.367	0.000	0.032	0.00000E+00	0.32174E-02	0.00000E+00
80.0	0.342	0.367	0.000	0.030	0.00000E+00	0.30625E-02	0.00000E+00
90.0	0.342	0.367	0.000	0.029	0.00000E+00	0.29916E-02	0.00000E+00
100.0	0.342	0.367	0.000	0.028	0.00000E+00	0.29587E-02	0.00000E+00
110.0	0.342	0.367	0.000	0.027	0.00000E+00	0.29433E-02	0.00000E+00
120.0	0.342	0.367	0.000	0.025	0.00000E+00	0.29361E-02	0.00000E+00
130.0	0.342	0.367	0.000	0.024	0.00000E+00	0.29328E-02	0.00000E+00
140.0	0.342	0.367	0.000	0.023	0.00000E+00	0.29312E-02	0.00000E+00
150.0	0.342	0.367	0.000	0.021	0.00000E+00	0.29305E-02	0.00000E+00
160.0	0.342	0.367	0.000	0.020	0.00000E+00	0.29301E-02	0.00000E+00
170.0	0.342	0.367	0.000	0.019	0.00000E+00	0.29299E-02	0.00000E+00
180.0	0.342	0.367	0.000	0.018	0.00000E+00	0.29299E-02	0.00000E+00
190.0	0.342	0.367	0.000	0.016	0.00000E+00	0.29298E-02	0.00000E+00
200.0	0.342	0.367	0.000	0.015	0.00000E+00	0.29298E-02	0.00000E+00
210.0	0.342	0.367	0.000	0.014	0.00000E+00	0.29298E-02	0.00000E+00
220.0	0.342	0.367	0.000	0.013	0.00000E+00	0.29298E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.00000E+00
WETWELL POOL= 0.10332E+01

Case 10.

```
ENTER IPRESS,DT,TMSUMP,TMCOVR,TMSILL,NBRKS, RBRKO
0,,,,,,,,,,,,
ENTER INITIAL MOLTEN MASS IN RPV(KG)
,,
```

GRAVITY DISCHARGE CASE

T(SEC)	RBRK(M)	ABRK(M*M)	HRPV(M)	HCRD(M)	QBRK(M**3/SEC)	QDWNC	QSILL
1.0	0.060	0.011	2.550	0.002	0.47776E-01	0.00000E+00	0.00000E+00
2.0	0.065	0.013	2.546	0.004	0.56917E-01	0.00000E+00	0.00000E+00
3.0	0.071	0.016	2.541	0.007	0.66834E-01	0.00000E+00	0.00000E+00
4.0	0.076	0.018	2.536	0.010	0.77518E-01	0.00000E+00	0.00000E+00
5.0	0.082	0.021	2.530	0.014	0.88956E-01	0.00000E+00	0.00000E+00
6.0	0.087	0.024	2.523	0.018	0.10114E+00	0.00000E+00	0.00000E+00
7.0	0.093	0.027	2.516	0.023	0.11404E+00	0.00000E+00	0.00000E+00
8.0	0.098	0.030	2.507	0.028	0.12766E+00	0.00000E+00	0.00000E+00
9.0	0.104	0.034	2.497	0.034	0.14196E+00	0.00000E+00	0.00000E+00
10.0	0.109	0.037	2.487	0.041	0.15693E+00	0.53209E-03	0.00000E+00
11.0	0.115	0.041	2.475	0.048	0.17255E+00	0.15438E-01	0.00000E+00
12.0	0.120	0.045	2.462	0.054	0.18878E+00	0.39896E-01	0.00000E+00
13.0	0.125	0.049	2.448	0.060	0.20560E+00	0.69009E-01	0.00000E+00
14.0	0.131	0.054	2.432	0.066	0.22298E+00	0.99916E-01	0.00000E+00
15.0	0.136	0.058	2.416	0.071	0.24088E+00	0.13085E+00	0.00000E+00
16.0	0.141	0.063	2.398	0.075	0.25926E+00	0.16085E+00	0.00000E+00
17.0	0.147	0.068	2.379	0.079	0.27809E+00	0.18950E+00	0.00000E+00
18.0	0.152	0.073	2.358	0.083	0.29732E+00	0.21675E+00	0.00000E+00
19.0	0.157	0.078	2.336	0.086	0.31691E+00	0.24273E+00	0.00000E+00
20.0	0.163	0.083	2.312	0.089	0.33681E+00	0.26766E+00	0.00000E+00
21.0	0.168	0.089	2.287	0.092	0.35696E+00	0.29175E+00	0.00000E+00
22.0	0.173	0.094	2.261	0.094	0.37731E+00	0.31521E+00	0.00000E+00
23.0	0.178	0.100	2.233	0.097	0.39781E+00	0.33821E+00	0.00000E+00
24.0	0.183	0.106	2.203	0.100	0.41838E+00	0.36085E+00	0.00000E+00

t	R_{BRK}	A_{BRK}		H_{CRD}	Q_{BRK}	Q_{DOWNC}	Q_{SILL}
25.0	0.189	0.112	2.172	0.102	0.43896E+00	0.38323E+00	0.00000E+00
26.0	0.194	0.118	2.139	0.104	0.45949E+00	0.40538E+00	0.00000E+00
27.0	0.199	0.124	2.104	0.107	0.47988E+00	0.42732E+00	0.00000E+00
28.0	0.204	0.130	2.068	0.109	0.50008E+00	0.44903E+00	0.00000E+00
29.0	0.209	0.137	2.029	0.111	0.51998E+00	0.47049E+00	0.00000E+00
30.0	0.213	0.143	1.989	0.113	0.53951E+00	0.49163E+00	0.00000E+00
40.0	0.259	0.211	1.486	0.129	0.69163E+00	0.66790E+00	0.00000E+00
50.0	0.297	0.276	0.738	0.128	0.65022E+00	0.68578E+00	0.00000E+00
60.0	0.310	0.302	0.000	0.059	0.00000E+00	0.12278E+00	0.00000E+00
70.0	0.310	0.302	0.000	0.038	0.00000E+00	0.19889E-01	0.00000E+00
80.0	0.310	0.302	0.000	0.033	0.00000E+00	0.78971E-02	0.00000E+00
90.0	0.310	0.302	0.000	0.030	0.00000E+00	0.48182E-02	0.00000E+00
100.0	0.310	0.302	0.000	0.028	0.00000E+00	0.37338E-02	0.00000E+00
110.0	0.310	0.302	0.000	0.027	0.00000E+00	0.32904E-02	0.00000E+00
120.0	0.310	0.302	0.000	0.025	0.00000E+00	0.30955E-02	0.00000E+00
130.0	0.310	0.302	0.000	0.024	0.00000E+00	0.30068E-02	0.00000E+00
140.0	0.310	0.302	0.000	0.023	0.00000E+00	0.29658E-02	0.00000E+00
150.0	0.310	0.302	0.000	0.022	0.00000E+00	0.29466E-02	0.00000E+00
160.0	0.310	0.302	0.000	0.020	0.00000E+00	0.29377E-02	0.00000E+00
170.0	0.310	0.302	0.000	0.019	0.00000E+00	0.29335E-02	0.00000E+00
180.0	0.310	0.302	0.000	0.018	0.00000E+00	0.29315E-02	0.00000E+00
190.0	0.310	0.302	0.000	0.016	0.00000E+00	0.29306E-02	0.00000E+00
200.0	0.310	0.302	0.000	0.015	0.00000E+00	0.29302E-02	0.00000E+00
210.0	0.310	0.302	0.000	0.014	0.00000E+00	0.29300E-02	0.00000E+00
220.0	0.310	0.302	0.000	0.013	0.00000E+00	0.29299E-02	0.00000E+00

ACCUMULATED FRACTION OF DEBRIS IN:
DRYWELL(OUTSIDE CRD ROOM)= 0.00000E+00
WETWELL POOL= 0.81493E+00

APPENDIX C

APPENDIX C

A SELECTED REVIEW OF THE PROBABILISTIC RISK ASSESSMENT PREPARED FOR THE SHOREHAM NUCLEAR POWER STATION

Prepared for
Future Resources Associates, Inc.
Berkeley, California

by
P. R. Davis

August 1982

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A SELECTED REVIEW OF THE PROBABILISTIC RISK ASSESSMENT PREPARED FOR THE SHOREHAM NUCLEAR POWER STATION

I. INTRODUCTION

This report presents the results of an effort to perform a limited review of selected areas of the Shoreham Nuclear Power Station Probabilistic Risk Assessment (PRA). The effort was undertaken in June 1982 based on a services agreement between the author and Future Resources Associates (FRA). FRA had previously entered into a contract with Suffolk County, New York (the site of the Shoreham station), and this effort is part of a broader task involving other contractors and consultants to fulfill the Suffolk County Contract.

Due to time limitations, only ten man-days were available for the review and report preparation. This is insufficient to provide a comprehensive review of all important areas of the PRA. However, based on the authors experience,[1] it was concluded that a meaningful, though limited, review could be

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[1]The author has performed some nine reviews of Probabilistic Risk Assessments for various clients. Included in this nine have been four reviews of BWR PRAs similar to Shoreham. In addition, the author has performed core melt probability assessments of two BWRs.

accomplished. The review was limited primarily to a consideration of core melt probability and initial containment response.

The review proceeded basically in three phases. The first phase consisted of comparing the core damage probability estimated by the Shoreham PRA with similar estimates from four other PRA studies performed for boiling water reactors. The purpose of this comparison was to determine if the Shoreham PRA result was consistent with similar studies involving different sponsoring organizations and contractors.

The second phase involved developing a list of questions, based on a preliminary review, which was transmitted to the PRA performing contractor (Science Applications, Inc.) and a response obtained.

The third and final phase consisted of identifying any remaining issues and considerations which were judged to have the potential for influencing either the probability or consequences of core damage accidents as assessed in the Shoreham PRA.

The report is organized around these three phases, with Sections two through four covering the phases. A final section, Section five, provides the conclusions of the review.

It should be noted at the outset that performing probabilistic risk assessments for nuclear power plants involves a considerable amount of subjective judgement and interpretation. While significant advancements have been made in both methodology and data processing, the PRA field is of relatively recent origin, and a substantial body of ongoing research is providing new insights on various issues. As a result, there is a considerable difference of opinion among PRA practitioners regarding the validity and uncertainty of various PRA methods. In many cases, these differences can be shown to have a negligible effect on risk. In any event, most scientists and engineers familiar with PRA can find areas of disagreement and criticism with any PRA study. This does not necessarily mean the results are invalid, it merely reflects the subjective nature of many PRA methods and procedures.

Table II-1 - PROBABILISTIC RISK ASSESSMENTS CURRENTLY AVAILABLE

Plant	NSS Vendor	Utility	Type	A-E	Commercial Operation	Power Level	Publication	Sponsor	Performer	Remarks
Browns Ferry 1	GE	TVA	BWR4/Mark I	Utility	1973	1065	Aug. 1982	NRC	EG&G	IREP Plant
Grand Gulf 1	GE	Mississippi Power & Light	BWR6/Mark 3	Bechtel	1982	1250	Oct. 1981	NRC	Sandia	RSSMAP Plant
Peach Bottom 2	GE	Philadelphia Elec.	BWR4/Mark I	Bechtel	1973	1065	Oct. 1975	NRC	NRC-various cont.	RSS plant
Limerick	GE	Philadelphia Elec.	BWR4/Mark 2	Bechtel	1985	1065	1982	Utility	SAI/PLG	
Shoreham	GE	LILCO	BWR4/Mark 2	S&W	1983	819	1982	Utility	SAI/PLG	Draft only available
GESSAR II	GE	(NA)	BWR6/Mark 3	(NA)	(NA)	1220	March 1982	GE	GE	Contains proprietary information
Sequoyah	W	TVA	PWR-4 loop (Ice Cond.)	Utility	1982	1148	Feb. 1981	NRC	Sandia	RSSMAP Plant
Indian Point 2	W	Consolidated Ed.	PWR-4 loop	UE&C	7/74	873	1982	Utility	PLG	
Indian Point 3	W	PASNY	PWR-4 loop	UE&C	8/76	965	1982	Utility	PLG	
Zion 1 & 2	W	Commonwealth Ed.	PWR-4 loop	S&L	1973	1040	1981	Utility	PLG	
Surry	W	Va.Elect. & Power	PWR-3 loop	S&W	12/72	775	Oct. 1975	NRC	NRC-various cont.	RSS Plant
Crystal River 3	B&W	Florida Power	PWR	Gilbert	3/77	825	Dec. 1981	NRC	SAI	IREP Plant
Oconee 3	B&W	Duke Power	PWR	Utility/ Bechtel	12/74	886	Jan. 1981	NRC	Sandia	RSSMAP Plant

Table II-2 - COMPARISON OF CORE DAMAGE PROBABILITIES FOR BOILING WATER REACTORS

Plant	Type	PRA Sponsor	Principal Contractor	Damage
Browns Ferry	BWR 4/Mark I	Nuclear Regulatory Commission	EG&G	2.0 X 10^{-4}
Grand Gulf	BWR 6/Mark III	NRC	Sandia Labs	3.7 X 10^{-5}
Peach Bottom	BWR 4/Mark I	NRC	(1)	2.9 X 10^{-5}
Limerick	BWR 4/Mark II	Philadelphia Electric Co.	SAI	1.5 X 10^{-5}
Shoreham	BWR 4/Mark II	Long Island Lighting Co.	SAI	4.4 X 10^{-5} (2)

[1] Various contractors involved, under direct NRC management

[2] This value is actually a "core vulnerable" condition, as described in the Shoreham PRA. A small factor is applied to each core vulnerable probability to estimate the core damage probability. None of the other PRAs listed used this additional factor. Thus, the Shoreham "core vulnerable" probability is comparable to the other values in the table for core damage. (See Section IV for further discussion of this approach.)

II. COMPARISON WITH OTHER PRA RESULTS

This section presents the results of the first phase of the Shoreham PRA review; a comparison of the Shoreham core damage probability estimate with that computed by other studies. To date some 13 PRA studies have been completed and published. These studies are shown in Table II-1 along with pertinent information relative to the plant design. As shown in Table II-1, a total of six BWR PRAs have been published. However, the GESSAR II PRA contains proprietary information, and will not be considered further in this comparison. The remaining five BWR PRAs are compared in Table II-2. The second column (Type) indicates firts the primary system design classification (BWR 4 or BWR 6) and the containment design (Mark I, Mark II, or Mark III). For purposes of core damage probability estimates, the primary system design, along with associated safety systems, is more important than containment design.

The third column indicates the organization which sponsored (funded) the PRA effort, and the fourth column indicates the principal contractor who performed the work. The last column is the overall result in terms of computed probability of core damage accidents per reactor year for the plant being considered. As indicated by the last column in Table II-2, the Shoreham result falls in the middle of the core damage probability distribution, being about a factor of three above the lowest (Limerick) and about a factor of three below the highest (Browns Ferry).

It should be noted that the Shoreham PRA uses an additional factor which reduces somewhat the Table II-2 computed probability. This factor is applied to the "core vulnerable" condition, which is the value in the result column of Table II-2. None of the other PRAs use this approach. The use of and validity of this factor is explored further in Section IV.

III. PRELIMINARY REVIEW-QUESTIONS AND ANSWERS

Following a preliminary review, a list of seven questions was formulated which were transmitted to the Shoreham PRA contractor (Science Applications, Inc.). It should be noted that many more than seven questions were derived from this review, but all but seven were eliminated based on their resolution as a result of other information found in the PRA, or were found to not have a significant potential influence on either the probability or consequences of core damage. (Further potentially important issues were identified as the review proceeded. These are discussed in the following section. Insufficient time was available to discuss and attempt to resolve these issues with the contractor. However, the concluding section of this report attempts to evaluate the potential impact of these issues.)

The seven questions formulated in the preliminary review are considered separately. A brief explanation of the potential significance follows each question, followed by the SAI response, and concluding with a discussion of the adequacy of the response.

1.A.-Question -Will the high drywell temperatures calculated to occur during degraded core cooling accidents cause rapid degradation of the concrete wall and a pressure buildup inside the drywell?

1.B.-Explanation-Concrete is not a high temperature structural material. Degradation, in terms of hydrating and decomposition, can start as low as 200°F depending on age and composition of the concrete. While drywell temperatures are not displayed in Appendix C (MARCH calculation results), a drywell temperature of 3000°F is indicated on page C-73 (although this value is said to be unrealistic).

1.C.-SAI Response-No. Upon concrete degradation due to high temperature, H_2O is released from the concrete between 200°F and 700°F, and CO_2 is released between 1200°F and 1700°F. In order to increase the containment pressure by 1 psi, conservatively assuming no heat loss through the passive heat sinks and no migration in the concrete, approximately 105 cubic feet of concrete at an elevated temperature of 200-700°F would have to release its free water. At more elevated temperatures (greater than 1200°F) the decomposition of approximately 60 cubic feet of concrete would increase containment pressure about 1 psi. In terms of the unlined concrete walls, the thermal bonding layer above 1200°F should be approximately 6 inches deep for pressures to exceed the containment ultimate pressure capacity during periods of high containment temperatures. The particular sequence for which high temperatures are significant before vessel heat failure is Class 3. The thermal transient duration for this sequence, however, is very short. Therefore, the concrete wall is not expected to reach the decomposition temperatures of the carbonates to a depth of 1/2 foot.

1.D.-Evaluation of Response-The SAI response seems adequate. However, since drywell temperatures and the resulting concrete thermal response is not provided in the report, it is not possible to confirm that the potential for high temperature drywell degradation is insignificant. A further examination of this issue (see Section IV), tends to indicate that it may not be of concern.

2.A.-Question-How much effect on suppression pool decontamination factors does the normal MARCH code calculation of very little heat transfer between the hot noncondensible gases have?

2.B.-Discussion-Generally, the MARCH code calculates very little heat transfer between the hot non-condensible gases and the suppression pool through which the gases would flow following a degraded core cooling accident. This can produce unrealistically low suppression pool temperatures. The decontamination factor (DF) for radionuclides in the suppression pool is thought to be degraded as the suppression pool temperature approaches saturation conditions. Neglecting gas to pool heat transfer may therefore result in more optimistic DFs than should realistically be expected.

2.C.-SAI RESPONSE-The sensible energy of the hot noncondensible gases flowing from the primary system or drywell into the wetwell through the S/RV or vents, respectively, are considered in the containment response analysis of Shoreham. These gases are conservatively assumed to leave the suppression pool at suppression pool temperature, effectively cooling the gases down. The suppression pool temperature correspondingly increases due to this energy addition. The decontamination factor that was used accounted for this additional heat up of the wetwell pool due to the noncondensible gases.

2.D.-Evaluation of Response-The SAI response appears adequate and the issue is considered resolved. Furthermore, the assumed SAI DFs (Table 3. 7. 4, pp. 3-160) seem quite pessimistic (too low) based on recent information. This issue is discussed further in Section IV.

3.A.-Question-Have high temperature penetration failures and pre-existing containment leaks been considered?

3.B.-Discussion-Relatively recent evaluations have shown a significant number of instances when reactor containment building penetrations have been found leaking. Further, penetration seals are not designed to withstand the severe degraded core accidents analyzed in the PRA.

3.C.-SAI Response-Plant response and potential containment failures due to high temperature degradation of penetration seals were not specifically analyzed for plant consequences. These were considered implicitly in the assigned probability of containment leakage which precludes overpressurization as shown in the containment event trees.

3.D.-Evaluation of Response-The SAI response is not considered adequate to resolve this question. However, it appears that direct pool scrubbing of most radionuclides (by release through safety and relief valves) will occur for the most probable accident sequences (Table 1, pg. 3). In these cases, containment leakage is of lesser consequence. Furthermore, the Reactor Building Standby Ventilation System can process and filter significant amounts of containment leakage (see following question), especially if pool scrubbing has previously occurred to remove aerosols and fission products.

4.A.-Question-With degradation of the Reactor Building Standby Ventilation System (RBSVS) from aerosol loading, high fission product loading, and adverse environment, can the system keep up with modest leakage?

4.B.-Discussion-The RBSVS is designed to control the minor radionuclide releases which can result from normal operation, in addition to those releases predicted to occur from design basis accidents. It is not designed, however, to operate under degraded core cooling accidents when excessive pressures, temperatures, and aerosol/radionuclide filter loadings may challenge the system far beyond its design capabilities.

4.C.-Response-The RBSVS ventilation system consists of three parallel 45000 cfm exhaust fans (each having 100% capacity when used for recirculation during abnormal conditions) and two 1585 cfm 100% capacity parallel filter trains. Since each fan is 100% capacity for the RBSVS mode, only one filter train fan is required to operate, and the redundant filter trains may be shut down. The potential filtration of fission products was considered in the PRA to the extent that the RBSVS could handle design leakage rates and partial containment failure leakage rates which do not overpressurize the reactor building. At these flow rates, the dilution factors of these gases containing fission products and aerosols are on the order of 1500 to 1. Additionally, for these accident scenarios, it is known that aerosol plugging of the leakage paths can occur, which is expected to reduce the fission product and aerosol concentration of the gases escaping to the secondary enclosure. Therefore, failure of the RBSVS was considered a low probability event (i.e. 0.01 for C_1 and C_3 accident sequence classes). However, degradation of the RBSVS performance was addressed in the analysis of reduced decontamination factors for filter efficiencies.

4.D.-Evaluation of Response-The SAI response is considered marginally adequate to resolve the issue. The PRA assessment of RBSVS design capability and potential failure modes is insufficient to determine if a rigorous evaluation of RBSVS availability has been made. However, considering that the most probable accident sequences appear to be those which discharge most radionuclides directly to the suppression pool, and that the suppression pool DFs selected by SAI appear to be pessimistic, this issue seems to have little potential impact on risk.

5.A.-Question-Appendix H of the Shoreham PRA assumes that the hot debris bed which can be formed from core materials is always coolable if sufficient water can be supplied. What is the basis for this assumption, and has the effect of enhanced radionuclide release with attendant steam pressure excursion on containment integrity when this cooling occurs been considered?

5.B.-Discussion-During the progression of a core meltdown accident, it is likely that following penetration of the reactor vessel, the core material may form a debris bed which could be coolable providing water can be supplied.

5.C.-SAI Response-The Appendix H assumption of debris bed coolability, given that a sufficient water supply is available, is based upon previous analyses of coolable debris bed formation (references provided) and engineering judgment.

These analyses indicate that the dryout heat flux of a debris bed is a function of the bed height, the porosity, and the average particle size of the debris. Medium-sized particles resulting from vessel head failure or from mild core-water interaction may form a coolable debris bed if the decay heat level is low, and if the debris bed is not isolated from the water. The bubbling of gases, which are generated from concrete attack, through the melt could agitate the molten core material allowing water to penetrate and requench the melt into coolable particles.

The actual geometric configuration of the Shoreham containment, specifically the drywell floor and the pedestal region, was considered in characterizing the debris bed formation, the depth of the debris bed, and the probable medium-sized particles that can form from a mild interaction of the core material with the water.

Analysis of the disposition of the core debris upon bottom head failure indicates that the molten core debris would flow out of the pedestal region. The debris could spread out into the drywell floor, and a portion could also flow into the wetwell pool through the downcomers located underneath the RPV in the pedestal region. The height of the core material that can accumulate on the drywell floor is limited by the downcomer configuration to a maximum height of six inches.

The downcomer lip which extends about 6 inches above the floor could melt due to the interaction with the high temperature core debris. If this occurs, the debris bed height would be determined by the height of the flange which is only about 2½ inches. Therefore, it was concluded that a very shallow bed would form, assuming long term cooling to the water overburdern, given sufficient coolant injection.

5.D.-Evaluation of Response-The SAI response to the debris bed coolability issue is reasonable, and valid arguments are given supporting the potential for coolability. However, since the geometry and particle size of the debris bed is unknown, assessments of coolability must be considered speculative. SAI did not respond to the part of the question dealing with steam pressure spikes and fission product release.

Upon further consideration of this issue, it is considered not to be significant in terms of the potential for risk increase for the following reasons:

1. If containment failure has not occurred at the time debris bed cooling occurs, most of the fission products (which are released during the initial meltdown phase) will be securely trapped in the suppression pool water. Thus, the possibility of containment failure from a steam pressure surge upon debris bed cooling would not cause a large fission product release.

2. If containment failure occurs before debris bed cooling, the major consequences of the accident would be underway, and the added fission product release would likely not be significant.

5. A steam pressure surge sufficient to challenge containment integrity requires a large amount of water delivered to the bed and intimate mixing. It is not likely that a high volume water source would be available since most such sources within the containment must previously have been assumed failed or degraded to cause the accident to progress through core meltdown.

<u>6.A.-Question</u>-Have the recent Sandia experiments showing the potential for energetic molten core-water reactions been considered in the Shoreham PRA? Also, what is the basis for the particle size assumption of ½" (diameter) in the vessel and 2" in the reactor cavity?

<u>6.B.-Discussion</u>-This question is related to the previous one, except that the concern here is that energetic molten core water reactions may occur within the vessel or upon vessel failure if water already exists at these location. These events could cause early containment failure if the steam pressure surges were sufficient.

<u>6.C.-SAI Response</u>-The Sandia experiments on molten core-water interaction were considered in the Shoreham PRA. For very energetic molten core-water interaction, the mode of melt delivery into a pool of water and mixing with the water can affect the break up and interaction of the fuel and water. It is also known that the initial conditions can suppress efficient core-water fragmentation. In the Shoreham PRA, this in-vessel interaction was considered probabilistically. The source term and release characteristics for the base case were extrapolated from WASH-1400 analysis. However, the specific accident

sequence and the conditions in the vessel at the time of potential core-water interaction were considered in assigning the conditional probability. That is, for a high pressure transient accident sequence, the probability of in-vessel steam explosion was considered lower than that for a large LOCA sequence in which the primary system pressure is at approximately the containment pressure.

The containment response was analyzed for an ex-vessel core-water interaction in the drywell. Ex-vessel fuel-coolant mixing was considered more likely if water was available in the drywell. The analysis of the efficiency of the fragmentation, however, considered the geometric configuration of the floor. The shallow water pool could inhibit a very efficient mixing of coolant with the molten core. It was assessed that this would retard the rapid and efficient fuel fragmentation that is needed to form more heat transfer area for explosion propagation. An overall mean particle size of 2 inches was judged to approximate the particle size distribution for rapid steam formation during this period. Additionally, a sensitivity analysis was performed to assess the effect of small particle size formation. Such formation resulted in containment failure immediately following vessel heat failure. This sequence was accounted for in the containment event tree and included in release category AA.

6.D.-Evaluation of Response-The SAI response seems somewhat confusing, and is considered only partially adequate. However, a further evaluation of this issue indicates that the potential for large energetic reactions is small. Since extensive and complex structures exist in a BWR below the core, it is unlikely that the molten core migration could proceed in a coherent manner. Large, energetic molten core-water reactions require that rather large coherent masses of molten material and water mix in a short time. Furthermore, these below core structures contain a significant amount of stored heat and also provide extensive heat conduction paths from the core region to the lower plenum. It is therefore likely that most or all of the lower plenum water would be boiled off before the arrival of significant amounts of molten material.

In the reactor cavity below the vessel, vertical open pipes exist, connected to the suppression pool. Thus, it is not likely that any significant amounts of water could exist in this region to interact with molten material upon vessel failure (which in not likely to be coherent due to the many penetrations in the lower head).

<u>7.A.-Question</u>-What is the basis for assuming (pg. c-23) no zircaloy-water reaction when the zircaloy becomes molten?

<u>7.B.-Discussion</u>-The reaction between zircalon and water can be important because it increases the heat generation rate in the core and produces hydrogen. The resulting hydrogen can burn, adding additional heat load to the containment, or it can decrease the D.F. of the suppression pool if it mixes and flows with radionuclides to the pool.

<u>7.C.-SAI Response</u>-The statement on p. C-23 is as follows: "Metal-water reaction in a melted node was not allowed." During the postulated core degradation and meltdown accident

scenario, the core heat up and meltdown model assumes a formation of a coherent molten fuel pool supported by a crust at the solid fuel-melt interface. This crust and molten layer causes the steam blockage which prevents water or steam from flowing through the channels with molten fuel. Therefore, metal-water reaction in a melted node (or ZR) was not allowed.

7.D.-Response Evaluation-The SAI response is considered adequate in resolving this question.

IV. ADDITIONAL ISSUES CONSIDERED IN THE REVIEW

This section considers additional issues which were identified as potentially important during the review and assesses their potential significance. These issues have not been discussed with the PRA contractor since time was not available. The issues are as follows:

1. MSIV Leakage - During essentially all accidents considered in the Shoreham PRA, the Main Steam Isolation Valves are assumed to close based on the MSIV closure logic incorporated into the plant design. MSIV closure isolates the primary system from the remainder of the power conversion system (turbine, condensor, feedwater pumps, etc.). For many of the higher probability accidents, the effluent from the primary system (which contains radionuclides released from the overheated core) is released to the suppression pool through the safety and relief valves. If the MSIV seal is effective for these accidents, most of the fission products are scrubbed out by the pool. However, past experience with MSIVs has shown that leakage occasionally occurs. If such leakage occurred during the accidents being considered, a potential path through the valves and to the atmosphere could exist. This path could be even more likely when it is recognized that the MSIVs are not designed for the high temperature radiation environment produced by the accidents considered. The Shoreham PRA does not appear to provide an adequate consideration of this issue.

Two questions are involved in resolving this issue: What is the likelihood of significant MSIV leakage during important accidents, and what is the consequence of such leakage in terms of off-site release.

Due to time constraints, it was not possible to explore either of these questions in depth. However, it has been recognized that MSIV leakage does occassionally occur, and design and maintenance improvements are underway to provide additional assurance of leak integrity. Presumably, the Shoreham Plant will take advantage of these improvements.

With respect to the second question, MSIV leakage does not automatically mean release to the atmosphere. In fact, the system downstream of the MSIV is basically a closed system during operation, and it is designed to preclude radioactivity release. While potential paths to the atmosphere may exist under some conditions (such as condensor air ejectors) such leakage is expected to be automatically isolated and can also be manually isolated.

The potential for leakage through the MSIVs to the atmosphere depends on system design and operating procedures. However, based on this very perfunctory consideration of the matter, it does not appear to be a significant potential contributor to risk.

2. <u>Control System Failure</u> - All BWRs have rather complex control systems which monitor the flow of energy from the core to the turbine, the core power, the feedwater flow,

and various other parameters. The system automatically adjusts some of these parameters to compensate for changing conditions and to keep the plant running smoothly. It has been recognized rather recently that this same system can malfunction and cause the plant to experience a serious transient which could lead to a degraded core condition. This potential event does not appear to be adequately considered in the Shoreham PRA, or any other PRA. While it has not been conclusively demonstrated that such events have the potential for causing severe core damage which would significantly contribute to risk, some control system malfunctions have occurred which have caused rather severe disturbances to the normal operation of the plant. Until more work is completed (currently underway within NRC and its contractors), it is not possible to determine the significance of this potential for Shoreham.

3. Dependent Failures - One particularly difficult issue with all PRAs has been the treatment of dependent failure, wherein the failure of one system causes or influences the failure of another. The Shoreham PRA is considered deficient with respect to the description of how dependent failures were identified and quantified. It is not possible to tell, based on the information provided, if an adequate assessment has been given to dependent failures.

4. Decontamination Factors - The Shoreham PRA assumes suppression pool decontamination factors ranging from 1 to 100 (pp. 3-160). Recent research in this area, including experiments supported by analysis, shows that DFs two or more

orders of magnitude higher would likely be more realistic. While such increases have not been universally accepted, the potential for significant increase appears strong. This would markedly decrease the source term for many accident sequences.

5. Derivation of "Conditional Frequency of Release" - The Shoreham PRA assigns (Table 4.2) values ranging from .07 to 1.0 to accident sequences. These values are multiplied by the calculated probability of a core vulnerable condition to arrive at the actual probability of a given accident sequence. The factor (pp. 3-142, 3-177) is apparently to account for the possibility that the core may not melt under the conditions assumed, or by some undefined mechanism, sufficient coolant is re-introduced into the system to terminate core heat-up. While this is a novel approach which appears to have merit, an adequate description of how the "Conditional Frequency of Release" values were derived could not be found. The factors do not appear, however, to have an overly significant influence on computed risk since they are not very small.

6. Conflict with Recent Degraded Core Probabilistic Assessment - Recently, an NRC sponsored study (NUREG/CR-2497) has suggested that core damage probabilities may be significantly higher than assessed in the Shoreham (and other) PRAs. While this report is under review (the author is involved in one such review), preliminary indications are that the results are based on invalid extrapolations, include questionable statistical procedures, and do not consider systems and procedures which can be effective in mitigating the progression of core meltdown accidents.

V. CONCLUSIONS

Based on this limited and selective review of the Shoreham PRA, the following conclusions are derived:

* The study appears to be a comprehensive and competent assessment employing state of the art methodology and generally adequate data sources.
* The results were found to be comparable, in terms of degraded core condition probability, to other PRA studies of similar plants.
* While some areas of the study were considered to be deficient, in no case was a deficiency found which appeared to have a significant potential for increasing the risk. (The omission of external event accident initiators could be an exception, but review of this issue was beyond the scope of the effort.)
* One area (suppression pool decontamination factors) was found which appears to have a significant potential for decreasing the computed risks from Shoreham. Recent evidence strongly suggests that much higher DFs may be appropriate for suppression pools. This could substantially reduce the Shoreham radionuclide release.

Appendix D

INTERNAL FLOOD ANALYSIS

by

HOWARD E. LAMBERT

Appendix D

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APPENDIX D

INTERNAL FLOODING ANALYSIS

(SAI's Section 3.4.4.1 and Appendix G)

D.1 Introduction

This appendix addresses the possibility that portions of emergency core cooling systems could be disassembled during maintenance (e.g. a pump impeller replacement, valve stem replacement, valve seat adjustments). If during this disassembly, human error or set of human errors occur which deisolate the component undergoing maintenance, such as opening a MOV, then release of water through the opened valve can occur. This has already occurred at Peachbottom.

As described below, we believe that SAI's computation of the accident frequency caused by internal flooding is incorrect and non-conservative for the two reasons described below.

- The approach used by SAI in quantification of these sequences does not produce units with accident frequency. One key event in the flooding sequence is the operator inadvertantly opening a valve during maintenance. The probability estimate for this event is given on a per maintenance act basis. SAI's calculations do not reflect this, which leads to a factor of about 100 in underestimating the internal flood frequency.

- SAI used human error probability estimates from Swain and Guttmann's work that do not reflect a highly stressful situation. We believe that a highly stressful situation does exist when internal flooding occurs and that all calculations should reflect this.

D.2 Description of the Internal Flooding Sequence

SAI identified 8 initiator types which would lead to flooding at level 8 in the Reactor Building where several ECCS and residual heat removal components are located. In particular, SAI identified 3 initiator types, 2, 3 and 4 as described below which could quickly (within 30 to 40 Minutes) flood and disable the components described in SAI's Table G.6 if the operator fails to reclose the isolation valve in time.

INITIATOR	WATER SOURCE	SYSTEMS INVOLVED *
2	Containment Storage Tank	HPCI, RCIC and CS
3 & 4	Screenwell	RBCLCW & RHR

In this case, it should be noted that internal flooding does not result in loss of the power conversion system, a high pressure injection system normally on-line when the reactor is producing power. While flooding occurs, if the operator erroneously isolates the power conversion system (contrary to normal operating procedures), then an accident sequence is initiated as described by the bold line on the event tree in figure 3.4.23.

* HPCI: High Pressure Coolant Injection
RCIC: Reactor Core Isolation Cooling
CS: Containment Spray
RBCLCW: Reactor Building Closed Loop Cooling Water
RHR: Residual Heat Removal

Table G.6

SUMMARY OF VITAL EQUIPMENT ASSOCIATED WITH SAFETY SYSTEMS LOCATED IN THE ELEVATION 8 COMPARTMENT AND THE POSTULATED HEIGHT AT WHICH VITAL EQUIPMENT COULD DISABLE THE SAFETY SYSTEM

SYSTEM	ASSOCIATED VITAL EQUIPMENT	POSTULATED DISABLED HEIGHT
HPCI	HPCI PUMPS	6'-0"
	HPCI TURBINE	8'-0"
	HPCI INST. RACK A,B	*3'-0"
RCIC	RCIC PUMP	5'-0"
	RCIC TURBINE	4'-0"
	RCIC INST. RACK A,B	*6'-0"
LPCI	RHR PUMPS	6'-0"
	RHR INST. RACK A,B	*6'-0"
CORE SPRAY	CORE SPRAY PUMP	6'-0"
	CORE SPRAY INST. RACK A,B	*6'-0"

* Based on the assumption that a short will cause a shutdown of the system.

** The estimated figures indicate that the suppression pool inventory may or may not be enough to effect vital equipment. However, if this inventory is combined with other sources such as the reactor primary system (during ADS sequence) it will without question disable the system.

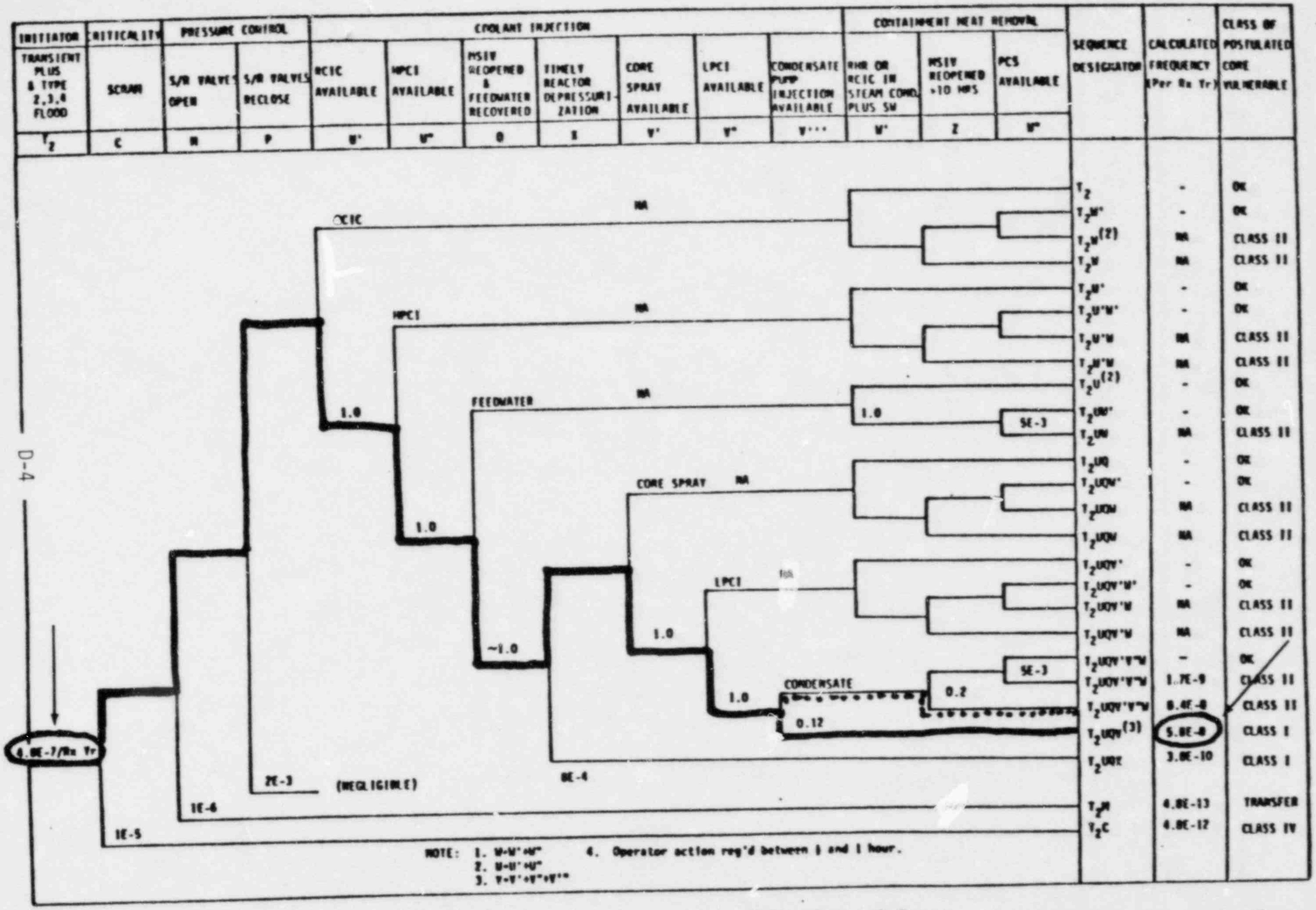

Figure 3.4.23 Event Tree Diagram for Sequences Following a Release of Water into Elevation 8 of the Type 2,3,4 Initiator

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SAI developed fault trees which describe the following sequence of events involving operator error that lead to flooding at level 8 concurrent with loss of the power conversion system.

- Event A: On-line maintenance of any of the following systems occurs
  1) HPCI
  2) RCIC
  3) Core Spray
  4) RBCLCW HX
  5) RHR HX

- Event B: System is disassembled for maintenance.

- Event C: Operator inadvertantly opens an isolation valve during maintenance causing flooding to start.

- Event D: Operator fails to reclose the isolation valve within 40 minutes which results in flooding to the six foot level.

- Event E: Operator erroneously isolates power conversion system during flooding.

A typical fault tree development for the sequence described above is shown in figure G.6. It must be noted that SAI combined events A and B on the fault tree.

D.3 Units of Frequency Issue

As described in the previous section, event C is the initiating event which causes flooding. The occurrence of events A and B defines a vulnerable system state that permits event C to initiate flooding when event C occurs. Furthermore, the occurrence of events A and B and C defines another vulnerable system state that permits event E to initiate the accident sequence when event E occurs. Because event C is an initiating event, we must compute the frequency of occurrence of this event.

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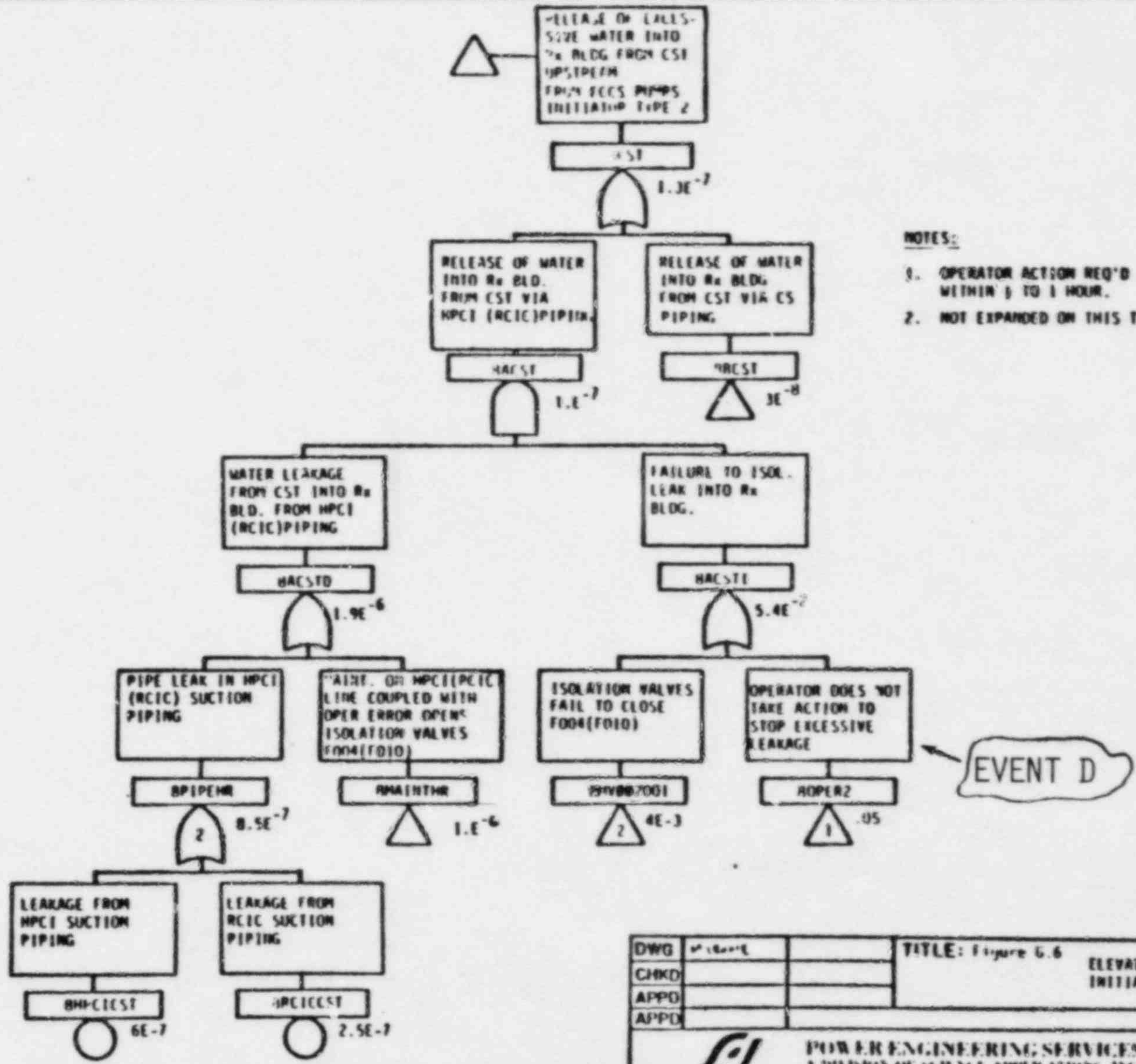

NOTES:

1. OPERATOR ACTION REQ'D WITHIN ½ TO 1 HOUR.
2. NOT EXPANDED ON THIS TREE.

DWG			TITLE: Figure 6.6 ELEVATION 8 FLOOD INITIATOR TYPE 2
CHKD			
APPD			
APPD			Sh 1 of 2 SCALE –
SAI POWER ENGINEERING SERVICES A DIVISION OF SCIENCE APPLICATIONS, INC.			DWG NO / REV

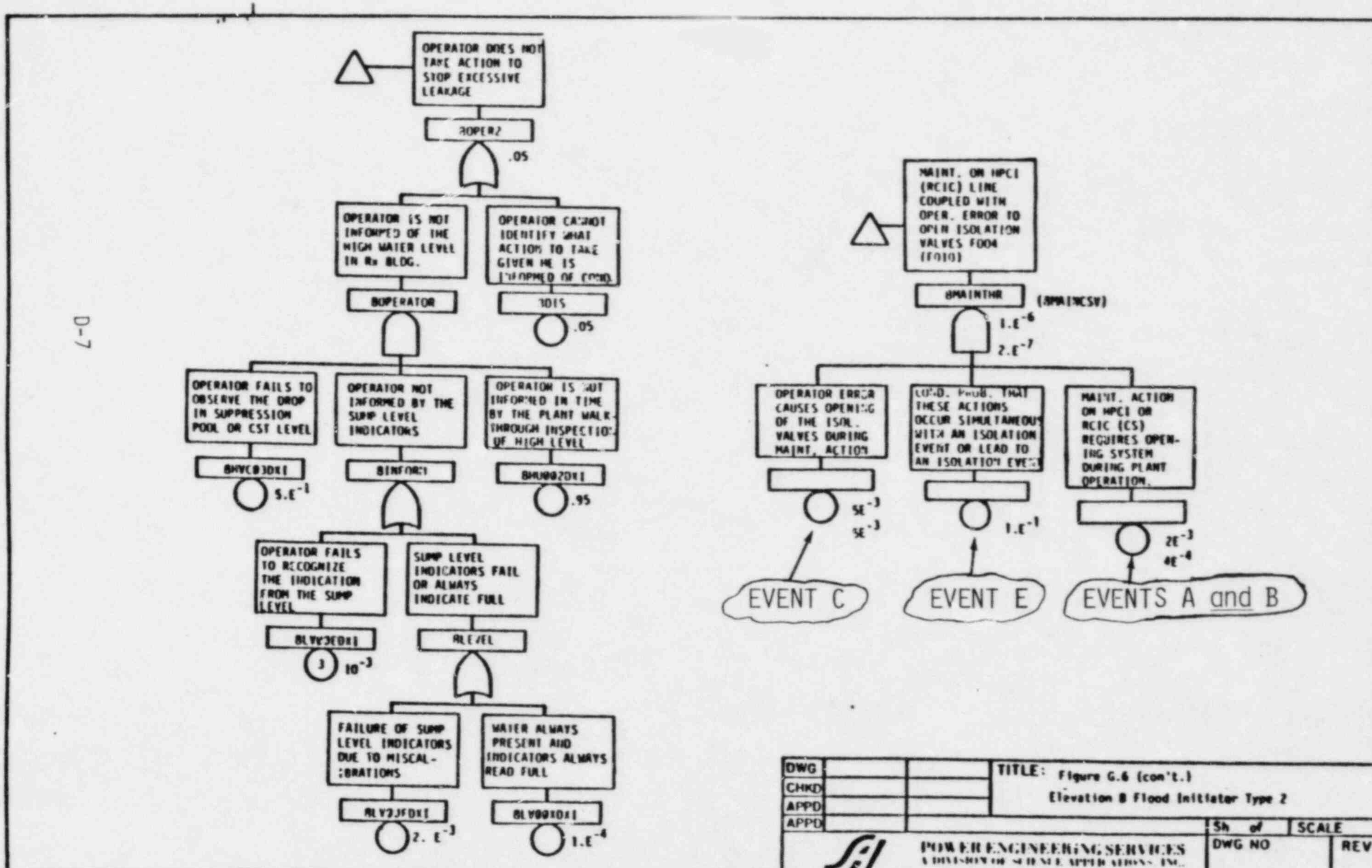

Figure G.6 (con't.)
Elevation B Flood Initiator Type 2

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Events C and D describe operator error and SAI used Swain and Guttmann's estimates, reference 1, to assign probabilities to these events. Specifically for event C, SAI on page G-38 of the draft Shoreham PRA said "The conditional probability that the operator opens the isolation valve to the system while it is undergoing maintenance is determined from Swain and Guttmann. The value is determined to $5x10^{-3}$/valve operation. This error rate is higher than normal plant operator error rates since it has been shown that operator error under maintenance conditions may be higher than that found performing normal plant function".

Using the above probability, we must compute the number of valve operations per year to obtain accident frequency, i.e. the expected number of flooding events per reactor year. The number of valve operations relates directly to the number of maintenances performed per year since the isolation valve must be closed and reopened once during maintenance.

Hence the units on event A should be the expected number of on-line maintenances per year, <u>not</u> system unavailability as computed by SAI. Computation with system unavailability results in an accident frequency about 100 times smaller than with the expected number of maintenance acts per year. When this point was raised at FRA's July 16 meeting with SAI personnel, SAI answered with a response described in the attachment. This attachment uses HPCI as an example.

The HPCI system unavailability is 10^{-2} (units without dimensions) and its maintenance frequency is

$$\frac{.09 \text{ acts}}{\text{month}} \quad * \quad \frac{12 \text{ months}}{\text{year}} \quad = \quad 1.08 \text{ acts/reactor year}$$

We claim that the expression for accident flood frequency above the six feet elevation concurrent with loss of the power conversion system is

[expected no. of on-line maintenances per year]	[probability that the system is disassembled given maintenance]	[probability that the operator opens the isolation valve during maintenance]	[probability operator fails to reclose the isolation valve]	[probability that the operator erroneously isolates the power conversion system during flooding]
$E[N_A(\text{one year})]$	$P(B/A)$	$P(C/A)$	$P(D/A \cap B \cap C)$	$P(E/A \cap B \cap C)$

Where $\cap$ denotes logical intersection and

$$E\,[N_A\ (\text{one year})] = \int_{t}^{t+\Delta t} w_A(t)dt$$

Where w_A = maintenance frequency, maintenance acts per unit time

Δt = one year

t = time

Again, the reason for the above functional form is that $P(C/A)$ is given on a probability per maintenance basis. (In section D.6, we develop a probabilistic expression in which system unavailability is a valid term.)

Swain and Guttmann give a best estimate of 5×10^{-3}, as mentioned above, but estimate the 90% upper confidence level of $P(C/A)$ as 2×10^{-2}/maintenance outage. Becuase the long maintenance outage of 3.5 days (see attachment) results in several shift changes, it is conceivable that the upper bound estimate may be applicable, resulting in an estimate that would be a factor of 4 higher for the flooding frequency.

D.4 Level of Stress Issue

In addition SAI's estimate of the probability of failing to recover, $P(D/A \cap B \cap C)$, is 0.05 which corresponds to the non-stressful situation in figure 17-2, Swain and Guttmann. However, if stressful conditions exist, then the estimate is a factor of 5 higher, i.e. 0.25.

Also, one can see from the event tree in figure 3.4.23, that SAI assigned a probability of 0.12 to failing to restore the condensate system. Considering the level of stress involved, a probability of 0.25 or higher could be a more reasonable estimate.

D.5 Observations

Examining the event tree in figure 3.4.23, one can see that SAI calculated a frequency of 4.8×10^{-7}/reactor year for flooding to the six foot level (T2 initiator) and 5.8×10^{-8}/reactor year for a class I core vulnerable accident.

However, following our arguments given above, the estimate for the T2 initiator could be as high as 1×10^{-3}/reactor year for flooding to the six foot level and 1×10^{-4}/reactor year (or higher) for the class I core vulnerable accident--factors of about 2000 difference. These class I accident figures are for the core vulnerable states to be reached: the core melt frequency would be somewhat smaller, depending on an analysis we have not done.

We recommend that SAI recompute the internal flooding frequency considering that event A should be given in units of frequency. Furthermore, because the internal flooding sequence is dominated by a chain of human error events, we recommend that an organization expert in human factors carefully analyze and compute these accident frequencies associated with internal flooding. This recommendation applies to the entire flooding event-tree sequence involving human action, i.e.

- erroneously removing the power converison system during flooding
- failing to provide coolant makeup with the condensate pumps
- failing to provide containment heat removal by opening the MSIV's within 10 hours.

from reference 1

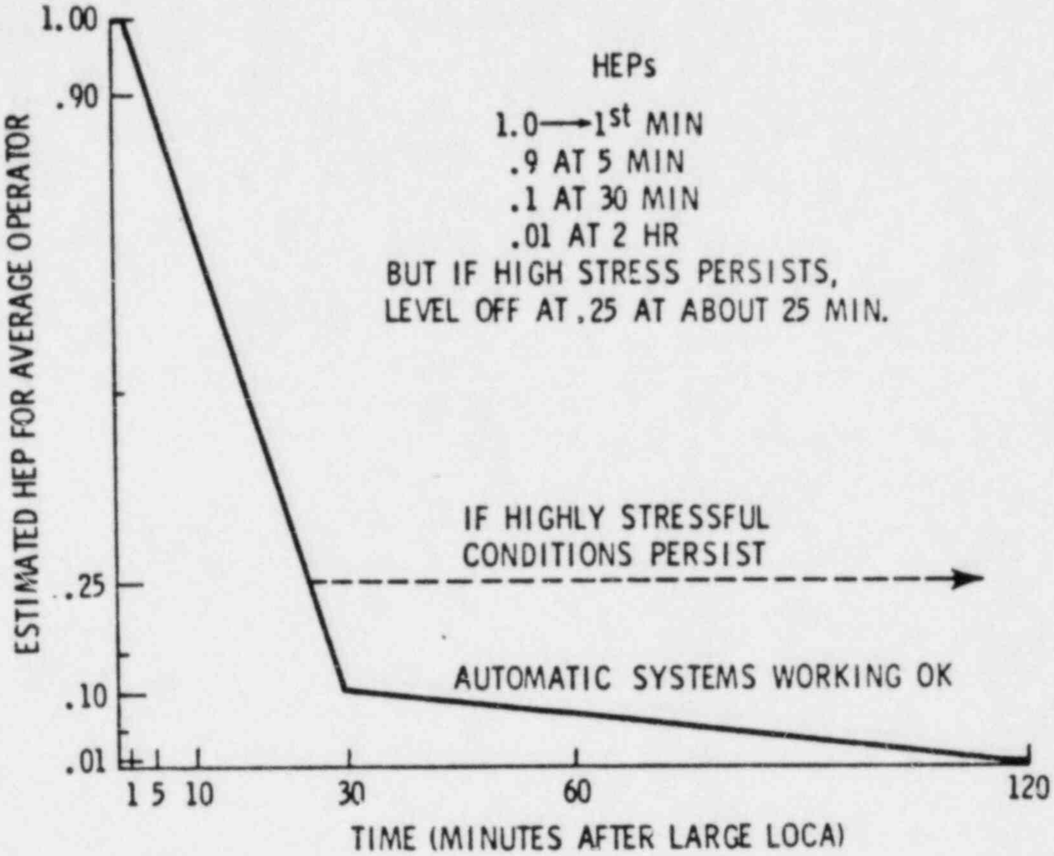

Figure 17-2. Estimated Human Performance after a Large LOCA

Also, as described in the main body of the report, an important assumption made by SAI is that flooding to the six foot level will not result in automatic closure of the MSIV's. (SAI does assume however that reactor trip will occur). It is important to verify that the assumption regarding automatic MSIV closure is true--otherwise the power conversion system is lost and the only normally available coolant makeup system is the condensate system. In this case the accident frequency caused by flooding would increase by an additional factor of about 10 and design changes may be necessary to mitigate the effects of the internal flooding accident sequence (e.g. elevating the ECCS pumps to a higher level).

D.6 Alternate Probability Expression

We can develop an equivalent expression for flooding frequency in which system unavailability is a valid term. In this case we must know the conditional probability per unit time that the operator inadvertantly opens the valve during maintenance, defined as $\lambda_{C/A}$ below.

For notation let Q denote system unavailability
P probability
λ frequency, probability of occurrence per unit time
$\cap$ logical intersection

The accidental flood frequency above the six foot level per year concurrent with loss of the power conversion system is

$$\int_{t}^{t+\Delta t} Q_A\ P(B/A)\ P\ (D/A\cap B\cap C)\ P(E/A\cap B\cap C)\ \lambda_{C/A} dt$$

where Δt = 1 reactor year

The term Q_A P(B/A) P(D/A$\cap$ B$\cap$ C) $\lambda_{C/A}$ is probability per unit time that flooding will occur above the six foot level. The term Q_A P(B/A) defines the fraction of time the system is vulnerable to flooding. We can calculate $\lambda_{C/A}$ as

$$\lambda_{C/A} = \left[\frac{\text{Probability operator inadvertantly opens valve}}{\text{maintenance outage}}\right] \left[\frac{\text{Maintenance outage}}{\text{time}}\right]$$

$$= \frac{5 \times 10^{-3}}{\text{maintenance outage}} \cdot \frac{\text{maintenance outage}}{3.5 \text{ days}} \quad \frac{1 \text{ day}}{24 \text{ hours}}$$

$$= 6.0 \times 10^{-5} \text{ / hr}$$

It must be noted however that the approach outlined in this section would still result in a factor of about 100 higher estimate than SAI's estimate for the flooding frequency.

REFERENCES

1. A.D. Swain and H.E. Guttmann, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, p. 13-4, April 1980

Question #1

Attachment

In the evaluation of the frequency of internal flooding events which could lead to potential core vulnerable conditions, the relationship of the frequency of internal flooding events to major maintenance acts on certain systems is required. Explain this relationship.

Response:

On-line maintenance of systems located in the reactor building which can release water into the the reactor building has been evaluated since it may be the dominant contributor to potential flooding. The following method is used to quantify the frequency of potential maintenance errors leading to the release of water into the reactor building.

Calculate the conditional probability of the system being subject to the release of water ~~to~~ the reactor building as a function of the length of time it is exposed*:

$$\left(\begin{array}{c}\text{Maintenance}\\ \text{Unavailability}\\ \text{Per System}\end{array}\right) \times \left(\begin{array}{l}\text{Fraction of the Unavailability for}\\ \text{Which System is Opened up and}\\ \text{Subject to Release of Water if}\\ \text{Additional Human Errors Occur}\end{array}\right)$$

(Table 6.8 SNPS DRAFT PRA)

This can be viewed in terms of a demand probability in the following context — those events for which maintenance requires opening the system up for major repairs in all likelihood requires much more than the mean time of repairs associated with all maintenance acts.** Because of the significant effort involved in disassembling safety systems, the mean time for these acts is estimated to be approximately 3.5 days (i.e., 1/2 the technical specification limit for on-line maintenance).

* The frequency of such events is found to be less than annual, which is to be expected.

** WASH-1400 evaluated the mean time of all maintenance acts to be approximately 19 hours.

Based upon this calculation and the total unavailability of the system, it is found that for HPCI such acts occur with a frequency of approximately:

$$\text{Frequency}\left(\frac{\text{Acts}}{\text{Mo.}}/\text{System}\right) = \frac{\frac{10^{-2}}{\text{system (HPCI)}}}{3.5\ \text{days}/\text{major system outage}}$$

$$\text{Frequency}\left(\frac{\text{Acts}}{\text{Mo.}}/\text{System}\right) = .09\ \text{acts}/\text{mo.} - \text{System (HPCI)}$$

An implicit assumption in this evaluation is that the frequency of such major on-line maintenance is approximately once per year, and therefore the "risk" associated with this on-line maintenance is directly proportional to the length of the outage, i.e. unavailability. If, on the other hand, the actual frequency of such major on-line work efforts is significantly higher than the frequency used here, there could be additional contribution to the likelihood of a release due to the higher frequency of such events. Specifically, if there is a substantially higher frequency of major on-line maintenance of the identified safety systems, then the chance of misoperation during the process of returning the system to service could produce a non-negligible contribution. However, it is judged that the probability of a release of water to a given elevation is directly proportional to the length of time that the system is opened up and subject to the postulated failure mode.

DRAFT - PRELIMINARY

The purpose of this section is to define the best estimate values used to quantify the fault trees for evaluating the conditional probability of maintenance operations leading to the release of large quantities of water into the Elevation 8 compartment.

1. The probability that the ECCS components in Elevation 8 may be disassembled during plant operations. This conditional probability must be less than the overall on-line maintenance unavailability taken from Appendix A.4 and shown in Table G.8.

Table G.8

MAINTENANCE UNAVAILABILITY

SYSTEM	TOTAL SYSTEM UNAVAILABILITY (APPENDIX A.4)	CONDITIONAL PROBABILITY OF SYSTEM OPENED DURING PLANT OPERATION (ESTIMATED)
Core Spray		
Loop A	2×10^{-3}	2×10^{-4}
Loop B	2×10^{-3}	2×10^{-4}
LPCI		
Pump Leg A, Pump Leg C	4×10^{-3}	8×10^{-4}*
Pump Leg B, Pump Leg D	4×10^{-3}	8×10^{-4}*
HPCI	10^{-2}	1×10^{-3}
RCIC	1.1×10^{-2}	1.1×10^{-3}
TOTAL	NA	3.3×10^{-3}

* A higher fraction of probability of maintenance is included here to account for heat exchanger maintenance on tubes.

The conditional probability of the system being opened is based upon the following considerations:

- Only a small fraction of the maintenance operation involve opening of the system to the Elevation 8 atmosphere; therefore, for most system maintenance operations (90%), the system is not subjected to the failure mode of interest, i.e. internal flooding of the Elevation compartment.

- A portion of the maintenance operation is assumed to be involved in disassembling and assembling the components; therefore, the system is not opened during this time to the Elevation 8 and also does not contribute to the potential for water release.

2. The conditional probability that the operator opens the isolation valve to the system while it is undergoing maintenance is determined from Swain and Guttman (G.2). The value is determined to be 5×10^{-3}/valve operation. This error rate is higher than normal plant operator error rates since it has been shown that operator error under maintenance conditions may higher than that found performing normal plant function.

G.5.3 Operator Failure to Take Appropriate Mitigating Action

In addition to these input quantities, there is another vital aspect of the quantification, the time dependent operator intervention error rate. Figure G.17 displays the error rate used to characterize the control room operator response to a rising water level in the sump tanks and subsequently in the Elevation 8 compartment.

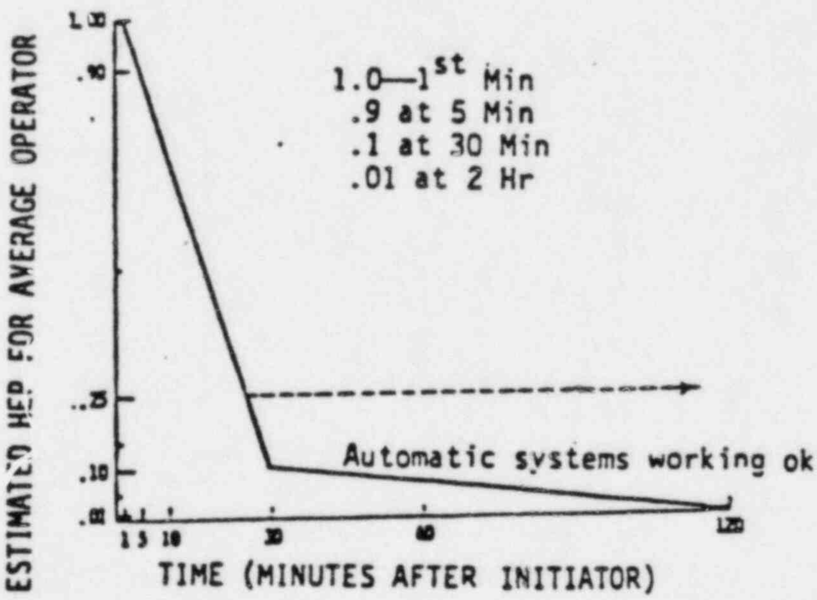

Figure G.17 Time Dependent Operator Error Rate Applicable to the Operator Response for Internal Flooding in the Elevation 8 Compartment.